Nuclear Fuels Storage and Transportation Requirements Document

Fuel Cycle Research & Development

Prepared for
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Nuclear Fuels Storage and Transportation
Planning Project
NFST Requirements Team

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DISCLAIMER

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This document was prepared by Alexcia Delley with contributions from the following Nuclear Fuel Storage and Transportation Team Subject Matter Experts.

NAME	ORGANIZATION
Randy Belles	ORNL
Bruce Bevard	ORNL
Syed Bokhari	DOE
Joe Carter	SRNL
Alexcia Delley	SRNL
Matt Feldman	ORNL
James George	DOE
Robby Joseph	ORNL
Rob Howard	ORNL
Mark Nutt	ANL
Tom Severynse	SRNL
Jack Wheeler	DOE
Jeff Williams	DOE

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ACRONYMS

AAR	Association of American Railroads
BRC	Blue Ribbon Commission
CFR	Code of Federal Regulations
CMS	Cask Maintenance System
CRWMS	Civilian Radioactive Waste Management System
DOE	Department of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
DOT	Department of Transportation
DS	Disposal System
ESC	Existing Size Cask/Canister
FCRD	Fuel Cycle Research and Development
F&R	Function and Requirement
FMS	Facility Maintenance System
GTCC	Greater-Than-Class C
HLW	High Level Waste
ISF	Interim Storage Facility
LLW	Low Level Waste
MDO	Management and Disposal Organization
MOU	Memorandum of Understanding
MTU	Metric Tons of Uranium
MTHM	Metric Tons of Heavy Metal
NEI	Nuclear Energy Institute
NFST	Nuclear Fuels Storage and Transportation Planning Project
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Technical Report
NWPA	Nuclear Waste Policy Act of 1982, as amended
OCRWM	Office of Civilian Radioactive Waste Management
R&D	Research and Development
RD&D	Research, Development, and Demonstration
RW	Radioactive Waste
SS	Storage System
SSC	System, Structures, and Components
SNF	Spent Nuclear Fuel (used interchangeably with "Used Nuclear Fuel")
TBD	To Be Determined
TBV	To Be Verified
TS	Transportation System
UFD	Used Fuel Disposition
UNF	Used Nuclear Fuel
U. S.	United States
WMS	Waste Management System
WP	Waste Package
WPS	Waste Packaging System
WPSC	Waste Package Compatible Size Container

REVISION HISTORY

Revision	Date	Changes
0	September, 2013	Initial Issue
1	June 30, 2014	The entire document was revised to expand the top level functions and requirements into lower level, more detailed requirements.
2	February 16, 2016	Updated information, standardized format for consistency, added appendix for consolidated NFST reference list, and added appendix for detailed NRC requirements for nuclear fuel storage.

WASTE MANAGEMENT SYSTEM[a]

1. Waste Management System Introduction

The *Nuclear Waste Policy Act of 1982, as amended*, (NWPA) established the federal government's responsibility to accept Spent Nuclear Fuel (SNF) [b] and high-level radioactive waste (HLW) from waste owners and generators for ultimate disposition. SNF generated by the current fleet of commercial nuclear reactors is being stored at the reactor sites, both in fuel pools and in dry Independent Spent Fuel Storage Installations (ISFSIs), with a limited amount being stored in away-from-reactor ISFSIs. Greater-Than-Class-C (GTCC) low level waste (LLW) is also currently stored at shutdown and operating nuclear power plant sites.[c] The federal government-generated and managed SNF and HLW is or will, when generated, be stored at federal sites. The Department of Energy, Office of Nuclear Energy (DOE-NE) is developing a suite of options and set of supporting analyses that will enable future informed choices about how best to manage these materials.

The benefits of consolidating these materials at an Interim Storage Facility (ISF) within an integrated nuclear waste management system have been recognized by both the Blue Ribbon Commission for America's Nuclear Future[d] and in the Administration's Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste ("Strategy").[e] Consolidated interim storage would:

- Allow for the removal of SNF from shutdown reactor sites
- Enable the federal government to begin meeting waste acceptance obligations
- Support the operation of a future deep geologic repository
- Provide for increased flexibility in the overall nuclear waste management system.

1.1 Background

DOE-NE is pursuing activities that can be conducted within the constraints of existing law and will facilitate the development of an ISF, a geologic repository, and supporting transportation infrastructure. The activities

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[a] This report was prepared to advance DOE-NE's objective of developing a suite of options and set of supporting analyses that will enable future informed choices about how best to manage SNF and HLW. As such, this report should not be interpreted as a statement of DOE policy as to how it intends to fulfill its acceptance obligation under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste, 10 CFR Part 961. To the extent the discussions or recommendations in this report conflict with the provisions of the Standard Contract, the Standard Contract provisions prevail. (DOE recognizes that the dates in the *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste* are no longer considered absolute, however the dates are placeholders for planning purposes.)

[b] The terms "used nuclear fuel" or UNF and "spent nuclear fuel" or SNF were used interchangeably in this document.

[c] A Federal Circuit Court panel ruled that for purposes of determining damages in the spent nuclear fuel litigation, GTCC LLW waste is considered HLW under the terms of DOE's Standard Contract (Yankee Atomic Electric Co. v. U.S., 536 F. 3d 1268 (Fed. Cir. 2008) and Pacific Gas & Electric Co. v. U.S., 536 F. 3d 1282 (Fed. Cir. 2008)).

[d] *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy,* January 2012.

[e] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste,* January 2013.

being conducted can be transferred to a new nuclear waste management organization when established and will not constrain its options. These include initiating planning for a large scale transportation program; evaluating operational options for interim storage; furthering the design of a generic ISF; and developing plans for initiating a consent-based siting process.

The mission is to lay the groundwork for implementing interim storage, including associated transportation, per the Administration's Strategy. Activities will support and maintain confidence in the safety and sustainability of nuclear energy by demonstrating responsible actions to ensure the safe, secure and effective management and disposition of used nuclear fuels.

The NFST is developing and beginning the implementation of an integrated management plan to (1) implement interim storage; (2) improve the overall integration of storage as a planned part of the waste management system; and (3) prepare for the large-scale transportation of SNF and HLW with an initial focus on removing SNF from the shutdown reactor sites. Within existing authorizations, the NFST is planning for and implementing the storage and transportation aspects of the Administration's Strategy. The NFST activities are prioritized and executed such that they will provide a foundation for a new nuclear waste management organization, if authorized by Congress, or proceed with full implementation if the program remains within DOE.

1.2 Purpose

This document:

- Establishes an initial set of Functions and Requirements (F&Rs) for storage and transportation portions of the waste management system
- Provides bases for planning future activities (e.g., alternative analyses)
- Identifies interfaces between the Storage and Transportation Systems.

This document may be updated, as appropriate, in the future to incorporate specific changes in technical scope or performance requirements that may have significant program implications. Such changes may include changes to the program mission, operational capability, and stakeholder issues.

1.3 System Overview

1.3.1 Administration's Strategy

DOE-NE is developing a suite of options and set of supporting analyses that will enable future informed choices about how best to manage the used fuel from commercial nuclear power reactors in the United States (U. S.). The NWPA established the U.S. Government's responsibility to safely manage commercial SNF and HLW.

In January 2013, the Administration issued its *Strategy for* implementation *of* the Blue Ribbon Commission recommendations. The key elements of this strategy are captured in Figure 1.

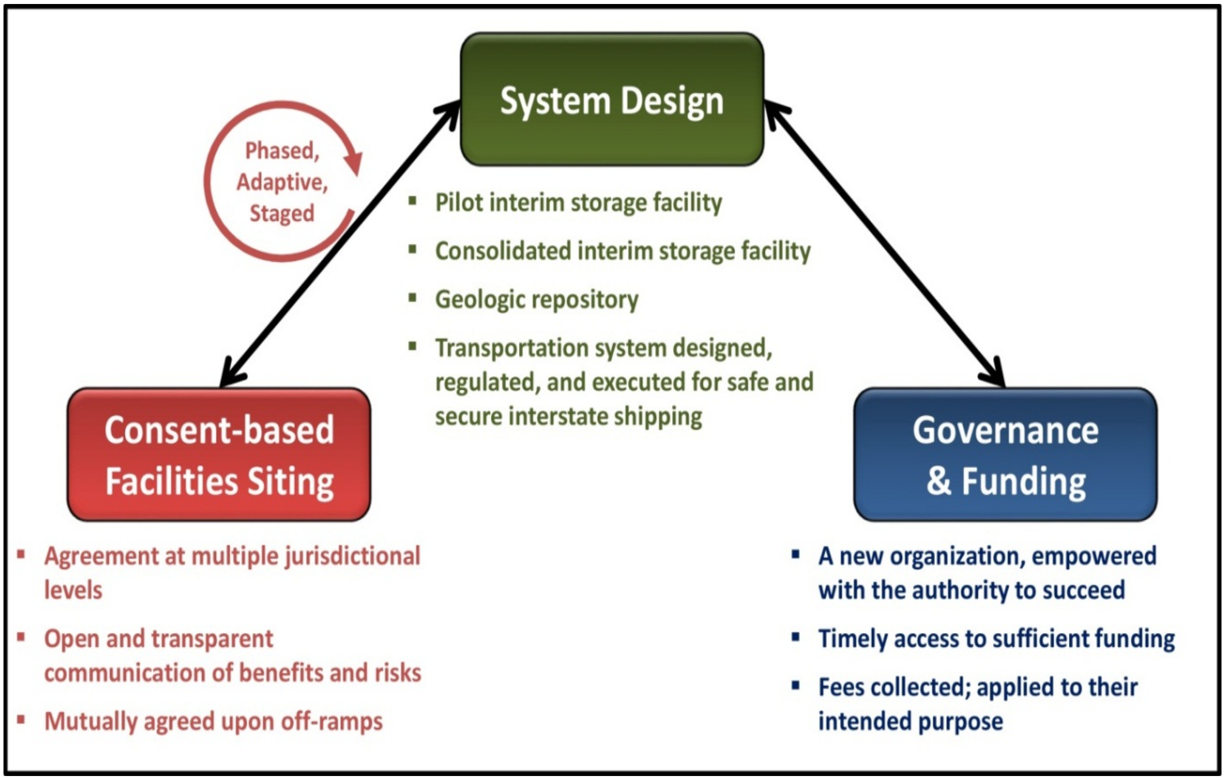

Figure 1- Key Strategy Elements

As stated,

> "This Strategy includes a phased, adaptive, and consent-based approach to siting and implementing a comprehensive management and disposal system. At its core, this strategy endorses a waste management system containing a pilot interim storage facility: a larger, full-scale interim storage facility (ISF) and a geologic repository in a timeframe that demonstrates the federal commitment to addressing the nuclear waste issue, builds capability to implement a program to meet that commitment, and prioritizes the acceptance of fuel from shutdown reactors. A consent-based siting process could result in more than one storage facility and/or repository, depending on the outcome of discussions with host communities; the *Nuclear Waste Policy Act of 1982* (NWPA) envisaged the need for multiple repositories as a matter of equity between regions of the country. As a starting place, this Strategy is focused on just one of each facility.[f]
>
> With the appropriate authorizations from Congress, the Administration currently plans to implement a program over the next 10 years that:
>
> - Sites, designs and licenses, constructs and begins operations of a pilot [ISF] by 2021 **[TBV]** with an initial focus on accepting used nuclear fuel from shut-down reactor sites;

[f] Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste, January 2013, p 1, paragraph 4.

- Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 **[TBV]** that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities; and
- Makes demonstrable progress on the siting and characterization of repository site(s) to facilitate the availability of a geologic repository by 2048 **[TBV]**."[g]

The Strategy is a framework for moving toward a sustainable program to deploy an integrated system capable of transporting, storing, and disposing of SNF and HLW from civilian nuclear power generators, defense, national security and other activities.[h]

1.3.2 Integrated Waste Management System

The Strategy states that a future Waste Management System (WMS) should be comprised of consolidated interim storage, a deep geologic repository, supporting transportation infrastructure, and waste management activities both at the civilian nuclear power generators and at federal sites. In addition, the nuclear waste management concept may include a research and development facility, possibly co-located with an interim storage facility. Currently, the WMS is comprised of SNF and HLW storage at generator sites. Integrated facility and infrastructure system analyses and trade studies that evaluate a range of facility and infrastructure waste management system architectures are being conducted to inform future decisions to implement the Strategy.

A high-level logistic framework describing the different options/alternatives and disposition pathways between storage at the generator sites through ultimate disposition in a geologic repository is shown in Figure 2. This framework shows the potential pathways for SNF and HLW stored at generator sites, both wet and dry, to reach ultimate disposition in a disposal facility, potentially involving passage through interim storage, and/or requiring packaging for disposal.

Waste Management System Architecture Evaluation efforts provide information regarding the various alternatives for managing SNF generated by the current fleet of light water reactors operating in the U.S. and federal government owned SNF and HLW. The objectives of these efforts are to:

- Provide quantitative information with respect to a broad range of nuclear waste management alternatives and considerations
- Develop an integrated approach for evaluating storage, transportation, and disposal options, with emphasis on flexibility in each activity
- Evaluate impacts of storage choices on SNF storage, handling, and disposal options
- Identify alternative strategies and evaluate these strategies with respect to cost and flexibility
- Consider a broad range of factors including repository emplacement capability, thermal constraints, packaging needs, storage and transportation alternatives, and other potential system impacts.

Initial analyses of an integrated system capable of transporting, storing, and disposing of SNF from civilian nuclear power generators has already been completed.[ij] These initial analyses did not include other SNF and

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[g] Ibid, p 2, paragraph 2.

[h] Ibid, p 1, paragraph 1.

[i] Used Fuel Management System Architecture Evaluation, Fiscal Year 2012, FCRD-NFST-2013-000020, Rev. 0, October 31, 2012.

[j] Used Fuel Management System Architecture Evaluation, Fiscal Year 2013, FCRD-NFST-2013-000377, Rev. 1 DRAFT(?), October 31, 2013.

HLW generators (e.g., defense, national security, etc.). These and future integrated system analyses and trade studies will support establishment and refinement of the functional requirements presented herein

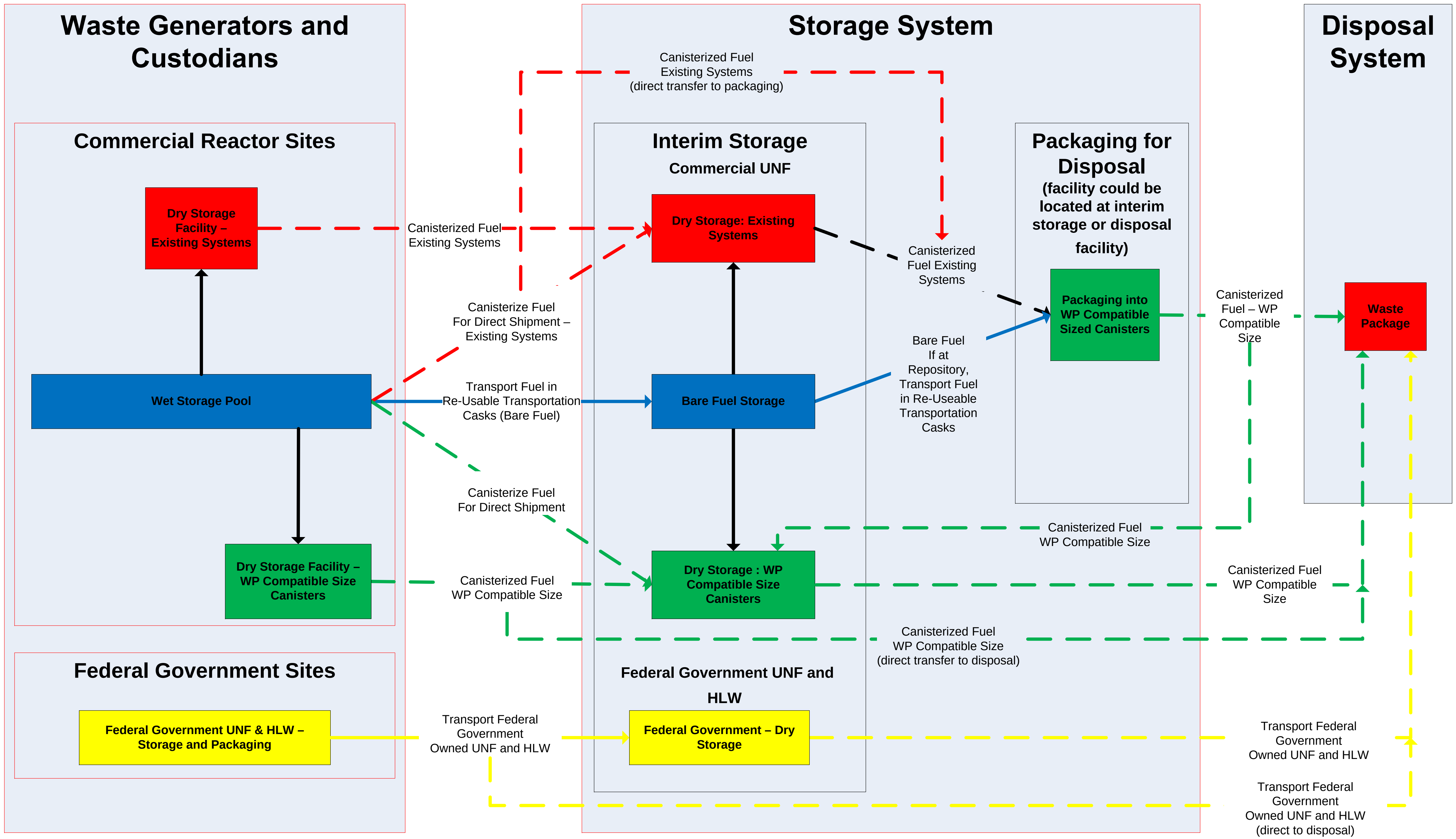

Figure 2 – Overall Options and Alternatives for Managing UNF and HLW.

1.3.3 Waste Management System Concept for Function and Requirements Development

It is recognized that the entire nuclear waste management system comprises the suite of facilities, infrastructure, and functions between storage at the generator sites through ultimate disposition in a geologic repository. However, the functional requirements presented herein are focusing on the transportation and storage aspects of the Administration's Strategy. This delineation is shown in Figure 3 and supports the development of capabilities needed to pursue activities that can be conducted within the constraints of existing law and facilitate the development of an ISF and the supporting transportation infrastructure.

Nuclear waste management activities at the generator sites, either commercial nuclear reactor sites or federal government sites, establish the boundary conditions for the storage and transportation portions of the waste management system. Therefore, generator waste management is considered as an interface.

The ultimate disposal of waste including SNF and HLW is also treated as an interface. Functional requirements for the Disposal System will be established later as the siting and development process for a geologic repository progresses. However, it is important to consider this interface when establishing functional requirements for the transportation and storage components of the waste management system.

The transportation aspect of the waste management system has been allocated to a Transportation System and the storage aspect has been allocated to a Storage System as shown in Figure 3. These systems will work in conjunction with each other to fulfill a variety of functional and performance requirements for the implementation of the transportation and storage aspects of the Administration's Strategy. The responsibilities included in these systems have been summarized below and are based on Program Guidance provided in Section 3 and Assumptions provided in Section 4. Interfaces with the generator sites and the Disposal System are discussed in Section 6.

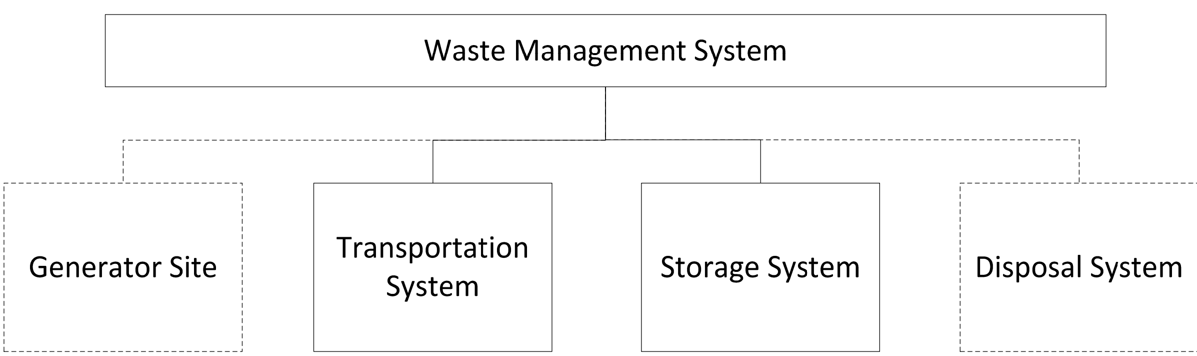

Figure 3- Waste Management System Architecture

1.3.3.1 Transportation System

The Transportation System (TS) responsibilities include design, acquisition, construction, operation, and maintenance of the systems, structures, and components needed to *transport* the following material from their present location to storage and disposal:

- Commercial SNF
- Government owned SNF
- Government owned HLW
- Greater-Than-Class C Low-Level Waste (GTCC – LLW) generated from decommissioning nuclear power reactors.

Transportation includes waste acceptance, transport of the materials defined above to an ISF, and transport of the materials defined above to a disposal facility.

1.3.3.2 Storage System

Storage System (SS) responsibilities include the design, acquisition, construction, operation, and maintenance of the systems, structures, and components needed to *store* the following material:

- Commercial SNF
- Government owned SNF
- Government owned HLW
- GTCC - LLW generated from decommissioning nuclear power reactors.

Storage includes both the pilot ISF and the larger ISF. The larger ISF shall provide the capability to store all of the materials defined above. The pilot ISF shall have limited capacity focused on the shutdown reactor sites as directed by the Strategy. The pilot ISF shall provide the capability to store the following:

- Commercial SNF stored in dry storage casks currently located at shutdown reactor sites
- GTCC- LLW in dry storage canisters generated from decommissioning nuclear power reactors.

2. WMS Resource Documents

The following reports were utilized to develop the functions and requirements (F&Rs) documented in this report:

- Since the passage of the NWPA in 1982, DOE and others have completed many studies directly and indirectly relevant to the role that consolidated storage could play in the back-end of the fuel cycle. In August, 2011, studies related to both the Monitored Retrievable Storage (MRS) facility and the surface facility proposed for the Yucca Mountain Repository were examined to identify prior studies which could be applied to future scoping, design, and cost studies. The results of the examination were documented in the report titled, *Consolidated Storage Lessons Learned and Background Information.* [k] Appendix A of the referenced report provides a list of prior studies judged as most important to informing future studies.

  While the present SNF and consolidated storage missions have changed considerably from the prior studies, the core functions to receive, store, package, continually monitor, and then ship nuclear fuel for ultimate disposal remain the same. Many of the design attributes of the previous studies remain valid technical solutions for managing the back-end of the fuel cycle.

- The report, *Civilian Radioactive Waste Management System Requirements Document*, A00000000 – 00811-1708 – 00003, Rev. 3, November 1996, was selected as a resource document because it specified the following top-level requirements for the Civilian Radioactive Waste Management System (CRWMS): 1) Accept and Transport Waste, 2) Store Waste (if approved), and 3) Emplace and Isolate Waste.

[k] Carter, J., Delley, A., Cotton, T. *Consolidated Storage Lessons Learned and Background Information,* FCRD-USED-2011-000345, Rev. 0, September 13, 2011.

The "store" requirements were evaluated. The requirements that were applicable to the current Administration's Strategy were used as a starting point for developing the F&Rs documented in this report.

- The report, *Civilian Radioactive Waste Management System Requirements Document*, DOE/RW-0406, Rev. 8, September 12, 2007, was selected as a resource document because it specified the following top-level requirements for the Civilian Radioactive Waste Management System (CRWMS): 1) Waste Acceptance, 2) Transportation, and 3) Monitored Geologic Repository.

  The "accept" and "transport" requirements were evaluated. The requirements that were applicable to the current Administration's Strategy were used as a starting point for developing the F&Rs documented in this report.

- The report, *Dry Storage of Used Fuel Transition to Transport*, FCRD-UFD-2012-000253, Rev. 0, August 2012, was selected as a resource document because it provided details of dry storage cask systems and contents in use in the U. S. for commercial light water reactor fuel dry storage.

- In September 2012, DOE contracted with three teams to prepare design concept studies to investigate SNF storage and transportation. DOE was seeking alternatives to support an evaluation and possible future selection of a concept that could be developed as an option for interim storage of commercial SNF. The three teams were headed by AREVA Federal Services, Energy*Solutions*, and Shaw Environmental and Infrastructure (now CB&I). The three teams completed their design concept studies and issued reports.

  The design concepts reports were prepared by the three independent teams without system or design requirements provided by the contract in order to solicit unbiased opinions. While the three reports provided some overlapping materials, each report also contained original concepts. Taken together the reports are comprehensive regarding transportation and storage concepts needed.

- The Fuel Cycle Research and Development report, *Used Fuel Management System Architecture Evaluation*, *Fiscal Year 2012*, FRDC-NFST-2013-000020, Rev. 0, October 31, 2012, summarized system-level analyses of the overall interface between the generators' sites, interim storage and ultimate disposition of SNF along with development of supporting logistic simulation tools. A range of at-reactor SNF acceptance methods were evaluated with respect to at-reactor SNF management, ISF SNF management, and SNF packaging.

- *Used Fuel Management System Architecture Evaluation*, *Fiscal Year 2013*, FCRD-NFST-2013-000377, Rev. 1 Draft October 21, 2013, presents the results of 1) further investigations into different alternatives for accepting SNF from the commercial reactor fleet, 2) an initial evaluations of the effects of repository thermal constraints on ISF operations, 3) an evaluation of used fuel pool configurations, 4) process flow configuration and description development, and 5) an assessment of research and development needs for future SNF packaging.

- *Project Concept for Nuclear Fuels Storage and Transportation,* FCRD-NFST-2013-000132, Rev. 1, June 15, 2013, used work performed by industry and national laboratories to develop a partial list of facility design concepts that could potentially be deployed to meet the requirements of the *Strategy* for further evaluation. An initial set of F&Rs were developed in this report.

  The F&Rs provided in the *NFST Requirements Document* (this document) supersedes the F&Rs provided in the *Project Concept for Nuclear Fuels Storage and Transportation* and will be used to provide guidance for future stages of design development.

- *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263, Rev. 1, March 2014, provided the commercial SNF, Government (DOE) owned SNF and Government (DOE) HLW currently in inventory and forecast for the life of the ISFs. This inventory projection will be used to ensure consistency between the various program elements.

- *Preliminary Site Factors and Considerations for Interim Used Fuel Storage Facilities*, FCRD-NFST-2013-000370, Rev. 0, September 23, 2013, presented siting factors and additional considerations for siting the pilot ISF and the larger consolidated ISF.

3. WMS Program Guidance

In addition to the Resource Documents described in Section 2, F&Rs were developed based on the following guidance from the *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste,* January 2013.

A. "The Administration endorses the key principles that underpin the BRC's recommendations. The BRC's report and recommendations provide a starting point for this Strategy, which translates many of the BRC's principles into an actionable framework within which the Administration and Congress can build a national program for the management and disposal of the nation's used nuclear fuel and high-level radioactive waste." [l]

B. "This Strategy includes a phased, adaptive, and consent-based approach to siting and implementing a comprehensive management and disposal system." [m]

C. "At its core, this Strategy endorses a waste management system containing a pilot interim storage facility; a larger, full-scale interim storage facility; and a geologic repository in a timeframe that demonstrates the federal commitment to addressing the nuclear waste issue, builds capability to implement a program to meet that commitment, and prioritizes the acceptance of fuel from shut-down reactors." [n]

D. "A consent-based siting process could result in more than one storage facility and/or repository, depending on the outcome of discussions with host communities; the Nuclear Waste Policy Act of 1982 (NWPA) envisaged the need for multiple repositories as a matter of equity between regions of the country. As a starting place, this Strategy is focused on just one of each facility." [o]

E. "Sites, designs and licenses, constructs and begins operations of a pilot interim storage facility by 2021 **[TBV]** with an initial focus on accepting used nuclear fuel from shut-down reactor sites." [p]

F. "Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities." [q]

G. "Makes demonstrable progress on the siting and characterization of repository site(s) to facilitate the availability of a geologic repository by 2048 **[TBV]**." [r]

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[l] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste,* January 2013, p 1, paragraph 3.

[m] Ibid, p 1, paragraph 4.

[n] Ibid, p 1, paragraph 4.

[o] Ibid, p 2, paragraph 1.

[p] Ibid, p 2, paragraph 2.

[q] Ibid, p 2, paragraph 4.

[r] Ibid, p 2, paragraph 5.

H. "A pilot interim storage facility with limited capacity capable of accepting used nuclear fuel and high-level radioactive waste and initially focused on serving shut-down reactor sites." [s]

I. "A larger, consolidated interim storage facility, potentially co-located with the pilot facility and/or with a geological repository, that provides the needed flexibility in the waste management system and allows for important near-term progress in implementing the federal commitment." [t]

J. "Following these initial efforts, capacity will be developed to enable the acceptance and transportation of used nuclear fuel at rates greater than that at which utilities are currently discharging it in order to gradually work off the current inventory." [u]

K. "The Administration remains committed to addressing the Cold War legacy; and, in addition to ongoing efforts, will consider transportation and interim storage of government-owned used nuclear fuel and high-level radioactive waste at interim storage facilities." [v]

L. "Depending on the outcome of a consent-based process, [the larger ISF] could have a capacity of 20,000 MTHM or greater, and could be co-located with the pilot facility or the eventual geological repository." [w]

M. "The Administration will undertake the transportation planning and acquisition activities necessary to initiate this process with the intent to transfer them to a separate organizational entity if and when it is authorized by Congress and in operation. Outreach and communication, route analysis, and emergency response planning activities consistent with existing NWPA requirements would be conducted during this time." [x]

4. WMS Assumptions

The following assumptions were developed in order to bridge the gap between the Administration's Program Guidance provided in Section 3.0 and the performance requirements listed in Section 8.0.

A. The pilot ISF will receive and store commercial SNF stored in dry storage canisters from shutdown reactor sites.[y]

Basis: The pilot ISF will only receive SNF stored in dry storage canisters to facilitate the design, licensing, and initial operations of the pilot.
Verification Method: Implementing Decision

[s] Ibid, p 4, paragraph 2.
[t] Ibid, p 4, paragraph 3.
[u] Ibid, p 5, paragraph 1.
[v] Ibid, p 5, paragraph 1.
[w] Ibid, p 6, paragraph 2.
[x] Ibid, p 7, paragraph 1.
[y] Under the Standard Contract (10 CFR 961.11), DOE is obligated to accept only bare spent nuclear fuel. Acceptance of canistered fuel would require a mutual agreement to modify the Standard Contract.

Basis: The Strategy states the Pilot will focus on shutdown reactor sites. Fuel at the initial nine shutdown sites (10 reactors) is stored in dry canisters. Fuel at other shutdown reactors is expected to be transferred to dry storage but the schedule is uncertain.

Verification Method: Implementing Decision

B. The pilot ISF and larger ISF will accept GTCC-LLW generated from decommissioning nuclear power plants.[z]

Basis: Per the latest revision of the *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263, accepting GTCC-LLW from decommissioned nuclear power plants will completely de-inventory shut-down reactor sites.

Verification Method: Not Applicable

C. The larger ISF will accept government owned SNF and HLW.

Basis: The Administration's Strategy states, "The Administration remains committed to addressing the Cold War legacy; and, in addition to ongoing efforts, will consider transportation and interim storage of government-owned used nuclear fuel and high-level radioactive waste at interim storage facilities."

Verification Method: Implementing Decision

D. The pilot ISF and larger ISF could potentially be co-located.

Basis: The Administration's Strategy states , "A consent-based siting process could result in more than one storage facility and/or repository, depending on the outcome of discussions with host communities; the *Nuclear Waste Policy Act of 1982* (NWPA) envisaged the need for multiple repositories as a matter of equity between regions of the country. As a starting place, this Strategy is focused on just one of each facility."

The Administration's Strategy includes "[a] larger, consolidated interim storage facility, potentially co-located with the pilot facility and/or geological repository that provides the needed flexibility in the waste management system and allows for important near-term progress in implementing the federal commitment."

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[z] DOE-NE recognizes that the DOE, Office of Environmental Management (EM), is preparing an Environmental Impact Statement (EIS) for disposal of Greater-Than-Class C Low-Level Radioactive Waste (GTCC LLRW). The EIS evaluates potential alternatives involving various disposal methods for application at six federally owned sites and generic commercial sites. The DOE-NE is considering alternatives for the interim storage of GTCC from decommissioned commercial nuclear power plants only.

If the pilot ISF and larger ISF were built as independent facilities, many of the systems, structures and/or components would be duplicated at additional permitting, capital, and operating cost.

Verification Method: Implementing Decision

E. A modular design and operating concept will be used to design and operate the pilot ISF and the larger ISF. Modular concepts may be used to either expand the capacity of the integrated facility and/or expand the functional capabilities of the facility. Modules will be deployed individually as needed to support changes in the facility mission.

Basis: The Administration's Strategy includes "a phased, adaptive, and consent-based approach to siting and implementing a comprehensive management and disposal system." Module design and operating concept are effective and efficient alternatives for satisfying this guidance.

Verification Method: Implementing Decision

F. The geologic repository (i.e., disposal facility) is not included in this version of the F&Rs.

Basis: The Administration's Strategy calls for "demonstrable progress on the siting and characterization of repository sites to facilitate the availability of a geologic repository by 2048 **[TBV]**." The development of the geologic repository F&Rs has been deferred to a later date.

Verification Method: Implementing Decision

G. The WMS shall provide a platform for ongoing R&D to better understand how the storage system will perform over time **[TBV].**

Basis: Per the *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy*, January, 2012,[aa] "A federal facility with spent fuel receipt, handling and storage capabilities can support other valuable activities that would benefit the waste management system. These include long-term monitoring and periodic inspection of dry storage systems and work on improved storage methods. Many current dry cask systems lack instrumentation to measure key parameters such as gas pressure, the release of volatile fission products, and moisture. Some of this work can be done in laboratories, but key aspects require the ability to handle and open loaded spent fuel storage containers and examine the fuel. A consolidated storage facility with laboratory and hot cell facilities and access to a substantial quantity and variety of spent fuel would provide an excellent platform for ongoing research and development to better understand how the storage systems currently in use at both commercial and DOE sites perform over time."

Verification Method: Implementing Decision

[aa] *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy*, January, 2012, p 39, paragraph 4.

H. SNF may require packaging prior to disposal.

Basis: If the current storage canisters do not meet the disposal facility acceptance criteria, then packaging will be required prior to disposal. This packaging could be implemented at the ISF or Repository.

Verification Method: Implementing Decision

5. WMS Regulatory Requirements

This section identifies the primary requirements of the WMS as established by, or derived from, key Federal laws and regulations.

A. WMS shall comply with the applicable provisions of 42 U.S.C. 10101 et seq., the *Nuclear Waste Policy Act of 1982*, *as* amended.

B. WMS shall comply with the applicable provisions of 10 CFR Part 71, *Packaging and Transportation of Radioactive Material*.

C. WMS shall comply with the applicable provisions of Department of Transportation (DOT) regulations as documented in Title 49 of the Code of Federal Regulations.

D. WMS shall comply with the applicable provisions of 10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel*, *High Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*. See Appendix B for details of 10 CFR Part 72, NUREG-1567, Regulatory Guide 3.48, Regulatory Guide 3.60, Regulatory Guide 3.50 (Rev 2), Regulatory Guide 3.53, and Regulatory Guide 3.73.

E. WMS shall comply with applicable provisions of 10 CFR Part 75, *Safeguards on Nuclear Material-Implementation of US/IAEA Agreement*.

F. WMS shall comply with the applicable provisions of 10 CFR Part 73, *Physical Protection of Plants and Materials*.

G. WMS shall comply with the applicable provisions of 29 CFR Part 1910, *Occupational Safety and Health Standards*.

H. WMS shall comply with the applicable provisions of 10 CFR Part 20, *Standards for Protection against Radiation*.

I. WMS shall comply with the applicable provisions of 10 CFR Part 851, *Worker Safety and Health Program*.

J. Handling of fuel must meet the requirements of ANSI/ANS 8.1-2014, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*.

6. WMS Programmatic Interface Requirements

The Programmatic Interface Requirements provide the administrative activities that must be performed to transport SNF, HLW, and GTCC-LLW between the WMS storage systems and the WMS external interface systems (i.e., between: 1) Generator Sites and the SS, and 2) SS and the Disposal System).

In general, implementation of these requirements should only generate documentation such as contracts, memoranda of understanding, waste acceptance criteria, response plans, communication plans, routing plans, shipping papers, titles, etc. Unlike performance requirements, implementation of these requirements should not generate systems, structures, or components (SSCs). These requirements also help ensure consistency and promote clear communications between the purchaser and the custodian. In some cases regulations are included with the requirements as bases.

A. Waste acceptance criteria shall be established to ensure that the waste can be received by the SS.

B. Waste acceptance criteria for disposal shall be established to ensure compatibility with the Disposal System.

C. Technical assistance and funding shall be provided to states and Tribes affected by shipment of SNF, HLW, and GTCC-LLW through their jurisdictions for training local public safety officials on procedures for safe routine transport and response to emergency situations in accordance with Section 180(c) of the NWPA. Effective emergency response is necessary to mitigate potential impacts should an accident occur during transport.

D. Appropriate routing requirements shall be established per 49 CFR Part 172, *Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, Training Requirements, and Security Plans*, Subpart I, *Safety and Security Plans*.

E. SNF, HLW, and GTCC-LLW must be protected while in transit. Security is required to prevent radiological sabotage or theft of these materials, and to coordinate response to potential threats.

F. The WMS will interface with the following entities when accepting and sending SNF, HLW and GTCC- LLW:

- Purchasers - Non-Federal entities that have entered into a contractual agreement with DOE.
- Custodians - Government entities possessing SNF and HLW considered candidate for disposal.

H. The WMS shall validate title and/or transfer of responsibility and custody documentation from the purchasers and custodians in accordance with applicable law.

I. As required by 10 CFR Part 961, accurate description of the package contents is required to assure safe handling of packages during and after transport.

J. Per 10 CFR Part 961 and 49 CFR Part 173, the loaded transport package must be appropriately marked and labeled for shipment of SNF, HLW, or GTCC-LLW.

K. Records of capacity, quantities, location and characteristics must be maintained in accordance with 10 CFR Part 961, *Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste* ("*Standard Contract*").

7. WMS Functional Hierarchy and Flow Diagrams

The functional hierarchy diagram illustrated in Figure 4 provides the hierarchical relationship between the functions to be performed by the WMS. The top-level function was broken down into sub-functions. The sub-functions are the actions or capabilities necessary to perform the top-level function. In order to ensure consistency and promote clear communications during the functional and requirements analyses process, the top level function "Manage SNF, HLW, and GTCC-LLW" was allocated to the Waste Management System; the sub-function "Transport SNF, HLW and GTCC-LLW" was allocated to the Transportation System; and the sub-function "Store SNF, HLW and GTCC-LLW" was allocated to the Storage System. The functions, requirements and bases statements will hereafter reference these systems.

The functional flow diagram illustrated in Figure 5 shows a logical interrelationship between the functions to be performed.

Figure 6 combines the material flows for the various SNF and HLW management operations shown in Figure 2 and the functional flow shown in Figure 5. The material flow is sequential, but there are some variations in the sequential flow through the functional steps that need to be understood. These variations are due to uncertainties in how the waste management system would be configured and operated, as shown schematically in Figure 2.

While it is recognized that the feasibility of directly disposing large dry storage canisters in a geologic repository is being evaluated, the packaging of SNF into disposal canisters is assumed to be required until such feasibility can be demonstrated. Where a packaging facility would be located, either co-located with the interim storage facility or co-located with the disposal facility, is not known. The location of the packaging facility affects the functional steps between discharge from an interim storage facility and the Disposal System. In addition, how the waste management system would be operated once a disposal facility is available is not known. For example, as shown in Figure 6 once the disposal facility is available, shipments to the interim storage facility may cease with all shipments from the generators going directly to the disposal facility, provided such shipments meet disposal requirements.

The WMS includes a Packaging for Disposal function. This function has been allocated to the storage system, regardless of where the facility would be located, as shown in Figure 4. It can be seen in Figure 6 that the location of the Packaging for Disposal function affects when in the process that SNF must pass through the transportation system between discharge from interim storage and receipt at disposal (either before or after the Packaging for Disposal function).

Figure 6 also recognizes that disposable canisters may be stored at the ISF. The SNF in such canisters would not have to be packaged for disposal when discharged from interim storage.

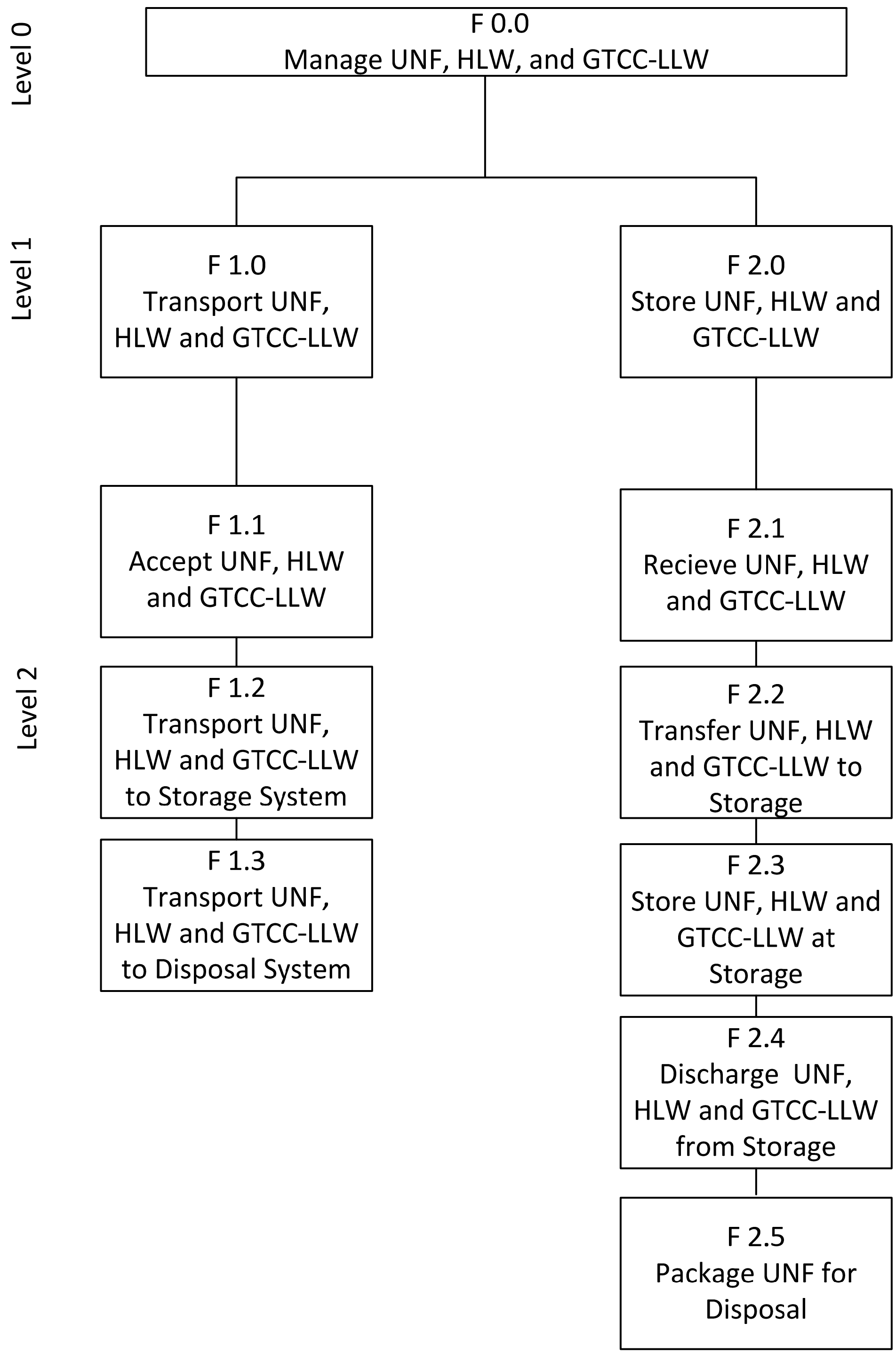

Figure 4 – Waste Management System: Functional Hierarchy Diagram

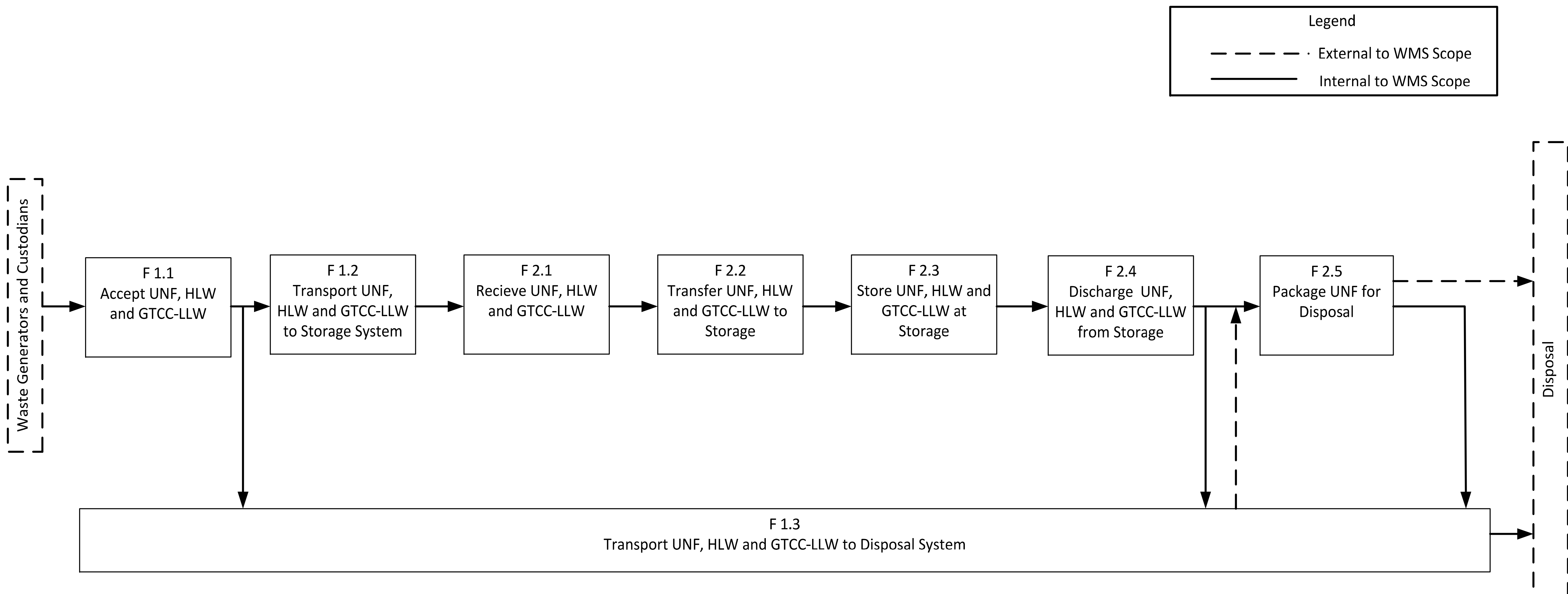

Figure 5 – Waste Management System - Functional Flow Diagram

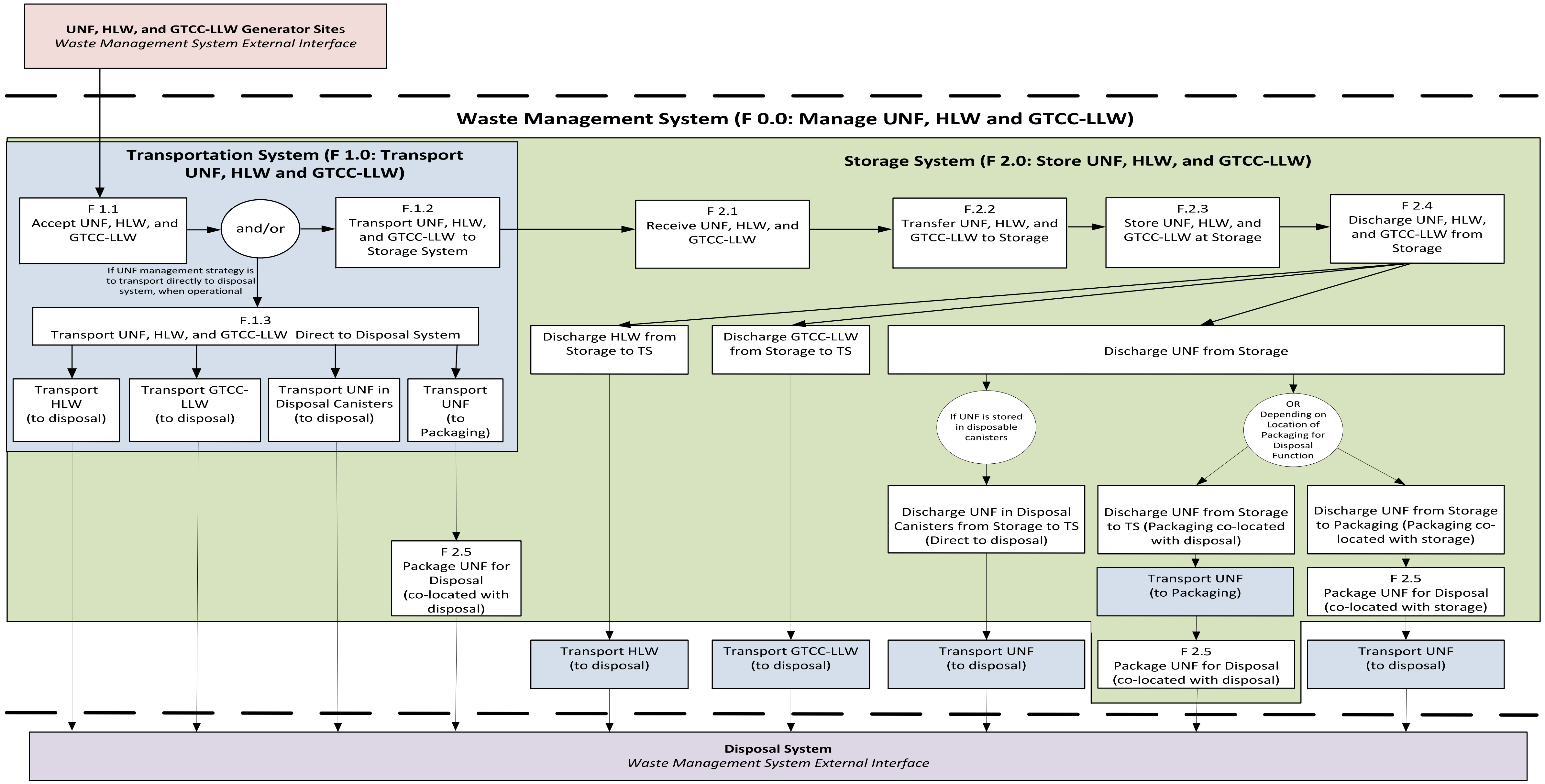

Figure 6 – Waste Management System – Functional and Material Flow Diagram

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8. WMS Functions and Performance Requirements

This section provides the functions and requirements for the WMS. The functional hierarchy is illustrated in Figure 4 and the functional flow is illustrated in Figure 5. In order to ensure consistency and promote clear communications during the functional and requirements analyses process, the top level function "Manage SNF, HLW, and GTCC-LLW" was allocated to the WMS.

If applicable, the assumptions provided in Section 4 are provided below as performance requirements. Once the assumptions are validated, they will be deleted from the assumption section. If the assumptions are found to be invalid, they will be removed from the assumptions section and the performance requirements will be rewritten.

F 0.0 Manage SNF, HLW and GTCC-LLW

R 0.1 The WMS shall be designed to accommodate the following materials as described in the latest revision of *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263.

- Commercial SNF
- Government owned SNF
- Government owned HLW
- GTCC-LLW generated from decommissioning nuclear power reactors

Basis: The WMS shall accommodate the materials listed in R 0.1 based on the following guidance extracted from the *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste,* January 2013.

- "Sites, designs and licenses, constructs and begins operations of a pilot ISF by 2021 **[TBV]** with an initial focus on accepting used nuclear fuel from shutdown reactor sites."
- "Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 **[TBV]** that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities."
- "The Administration remains committed to addressing the Cold War legacy, and in addition to ongoing efforts, will consider transportation and interim storage of government-owned used nuclear fuel and high-level radioactive waste at interim storage facilities."

R 0.1.1 The pilot ISF will receive and store commercial SNF stored in dry storage canisters from shutdown reactor sites.

Basis: The pilot ISF will only receive SNF stored in dry storage canisters to facilitate the design, licensing, and initial operations of the pilot facility.

Basis: The Strategy states the Pilot ISF will focus on shutdown reactor sites.

R 0.1.2 The larger ISF will accept commercial SNF from reactor sites.

Basis: The Administration's Strategy states "Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 **[TBV]** that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities."

R 0.1.3 The larger ISF will accept government owned SNF and HLW.

Basis: The Administration's Strategy states "The Administration remains committed to addressing the Cold War legacy, and will consider transportation and interim storage of government-owned used nuclear fuel and high-level radioactive waste at interim storage facilities."

R 0.1.4 The pilot ISF and larger ISF will accept GTCC-LLW generated from the decommissioning nuclear power plants.

Basis: Per the latest revision of the *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263, accepting GTCC-LLW generated from decommissioning nuclear power plant will completely de-inventory shut-down reactor sites.

R 0.2 The WMS shall be capable of handling canisters in use by the commercial nuclear industry and the federal government as currently defined in the latest revision of *Nuclear Fuel Storage and Transportation Planning Project Inventory Basis,* FCRD-NFST-2013-000263**.**

Basis Policy to accept existing materials as packaged **[TBD].**

R 0.3 The WMS shall provide certified transportation casks for transporting the canisters defined in R 0.2.

Basis: Per 10 CFR 961.11, Article IV.B.2., "DOE shall arrange for, and provide, a cask(s) and all necessary transportation of the SNF and/or HLW from the purchaser's site to the DOE facility."

R 0.4 The WMS shall be capable of accommodating SNF, HLW and GTCC- LLW at the annual acceptance rates specified in Table 1. **[TBV**]

Table 1: WMS Annual Acceptance Rate

Year	Pilot ISF		Larger ISF			
	Shut Down Reactor Sites		Operating and Future Commercial Reactor Sites		Government Owned	
	UNF	GTCC- LLW	UNF	GTCC- LLW	UNF[bb]	HLW
1	500 MTHM/yr. **[TBV]**	TBD				
2	1,000 MTHM/yr **[TBV]**	TBD				
3	1,500 MTHM/yr **[TBV]**	TBD				
4	1,500 MTHM/yr **[TBV]**	TBD				
5			2,000 MTHM/yr **[TBV]**	TBD	TBD	TBD
6			2,500 MTHM/yr **[TBV]**	TBD	TBD	TBD
7			3,000 MTHM/yr **[TBV]**	TBD	TBD	TBD

[bb] The terms “used nuclear fuel” or UNF and “spent nuclear fuel” or SNF re used interchangeably in this document.

Basis: The actual operational load is a function of the numbers, types and sizes of casks and canisters in which the SNF, HLW and GTCC- LLW are accepted from the point of origin. Since these specific numbers are not known and schedules have not been established, the required rates provided are estimated in terms of desired WMS system level acceptance rates to allow timely removal of material from shut down sites, and steadily reducing the inventory at operating sites.

R 0.4.1 The pilot ISF acceptance rate shall be ramped up over the first three years of operation to 1,500 MTHM/yr. **[TBV]**

Basis*:* The total 1,500 MTHM/yr **[TBV]** receipt rate was selected to be 50% of the larger facility acceptance rate. This acceptance rate will allow the Pilot to demonstrate key aspects of the transportation, fuel receipt and storage at sufficient rate to provide lessons learned for the larger ISF operations.

R 0.4.2 The larger ISF acceptance rate shall be ramped up over the first three years of operation to 3,000 MTHM/yr. **[TBV]**

Basis: The NFST is investigating a range (> 3,000 MTHM/yr) of acceptance rates in an effort to determine the impact on the expected federal liabilities. The lower end of this range (3,000 MTHM/yr **[TBV]**) is based on 150% of the current quantity of fuel being discharged annually[cc] in order to limit and then end the liability costs.[dd] 3,000 MTHM/yr **[TBV]** is also the historical repository program acceptance rate. Higher acceptance rates will limit the liability sooner but will also result in an expansion of the larger facility capacity and at a greater annual operating cost to support the increased acceptance rate.

R 0.5 The WMS shall have the capability to package SNF into disposal canisters prior to disposal.

Basis: If the current storage canisters do not meet the disposal facility acceptance criteria, and the disposal facility does not have packaging capability, then packaging will be required prior to shipment from the ISF.

R 0.6 WMS shall provide a platform for ongoing R&D to better understand how the storage system will perform over time **[TBV].**

Basis: Per the *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy*, January 2012,[ee] "A federal facility with spent fuel receipt, handling and storage capabilities can support other valuable activities that would benefit the waste management system. These include long-term monitoring and periodic inspection of dry storage systems and work on improved storage methods. Many current dry cask systems lack instrumentation to measure key parameters such as gas pressure, the release of volatile fission products, and moisture. Some of this work can be done in laboratories, but key aspects require the ability to handle and

[cc] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste,* p. 3, states "Currently more than 68,000 metric tons heavy metal (MTHM) of used nuclear fuel are stored at 72 commercial power plants around the country with approximately 2,000 MTHM added to that amount every year."

[dd] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*, p. 3, "outlines a strategy that is intended to limit, and then end, liability costs by making it possible for the government to begin performing on its contractual obligations."

[ee] *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy*, January, 2012, p 39, paragraph 4.

open loaded spent fuel storage containers and examine the fuel. A consolidated storage facility with laboratory and hot cell facilities and access to a substantial quantity and variety of spent fuel would provide an excellent platform for ongoing research and development to better understand how the storage systems currently in use at both commercial and DOE sites perform over time."

R 0.7 The pilot ISF shall begin operations in 2021. **[TBV]**

Basis: The Administration's Strategy states, "Sites, designs and licenses, constructs and begins operations of a pilot ISF by 2021 **[TBV]** with an initial focus on accepting used nuclear fuel from shutdown reactor sites."

R 0.8 The larger ISF shall begin operations in 2025. **[TBV]**

Basis: The Administration's Strategy states, "Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 **[TBV]** that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities."

R 0.9 The Transportation System infrastructure shall have a design life of at least 40 years **[TBV].**

Basis: Good engineering practice.

R 0.10 The Transportation System rolling stock shall have a design life of at least 40 years**.**

Basis: Railcars will be developed using off-the-shelf components. Maintenance will be performed on the railcar fleet at regular intervals such that those railcars will be more preventative maintenance that common rolling stock. Rail rolling stock can be maintained for 40 years on average.

R 0.11 The Storage System shall have a design life of at least 80 years.

Basis: Assuming the Repository receives at its historical design rate of 3,000 MT/yr[ff] and the total SNF inventory is approximately 140,000 MT[gg] , the repository operations will require approximately 47 years. The strategy indicates the Repository will open in 2048 **[TBV]** and the Pilot ISF in 2021 **[TBV]**. Therefore, the Pilot ISF will open 27 years (2048 – 2021 = 27 years) in advance of the Repository. Based on this data, the SS design life should be at least 74 years (47 years + 27 years).

10 CFR 72.42 provides that a license term for an ISFSI or MRS not exceed 40 years from the date of issuance. Licenses for either type of installation may be renewed by the NRC at the expiration of the license term upon application by the licensee for a period not to exceed 40 years and under the requirements of 10 CFR Part 72.

Establishing an 80 year design life for the Storage System covers one initial license term and anticipates one license renewal. A requirement for an 80 year design life

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[ff] Office of Civilian Radioactive Waste Management (OCRWM), *Civilian Radioactive Waste Management System Requirements Document*, DOE/RW-0406, Rev. 8, September 12, 2007**. [TBV]**

[gg] Carter, J. T. and Leduc, D. R, *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263, latest revision, Savannah River National Laboratory, Aiken, SC, August 30, 2013

will support the development of design bases to include time-dependent aging analyses (TLAAs) and aging management plans sufficient to cover the minimum duration (74 years) necessary to support repository operations to receive 140,000 MT of SNF.

R 0.12 Packaging for disposal function shall have a design life of 50 years.

Basis: Assuming the Repository receives at its historical design rate of 3,000 MT/yr[hh] and the total SNF inventory is approximately 140,000 MT[ii] , the repository operations will require approximately 47 years.

R 0.13 Research and Development (R&D) SSCs shall have a design life of 80 years **[TBV].**

Basis: The R&D system is assumed to be a subsystem of the SS. Therefore, the same basis for the SS 80 year design life is also applicable to the R&D Systems. Refer R 0.11 basis.

[hh] Office of Civilian Radioactive Waste Management (OCRWM), *Civilian Radioactive Waste Management System Requirements Document*, DOE/RW-0406, Rev. 8. September 12, 2007**. [TBV]**

[ii] Carter, J. T. and Leduc, D. R, *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263, latest revision, Savannah River National Laboratory, Aiken, SC, August 30, 2013

TRANSPORTATION SYSTEM

9. Transportation System (TS)

This section establishes the initial F&Rs for the Transportation System (TS)

9.1 TS Assumptions

The following assumptions were developed to bridge the gap between the Program Guidance provided in Section 3 and the TS requirements.

A. The WMS assumptions provided in Section 4 are applicable to the TS.

B. The majority of the shipments will be done by rail.

Basis: In April 2004, DOE selected the mostly rail scenario analyzed in the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (FEIS) [jj] as the transportation mode[kk]. In July 2005, DOE issued a policy stating that dedicated trains would be the usual mode of rail transportation for SNF and HLW to a repository. The National Academy of Sciences' (NAS) National Research Council Committee on Transportation of Radioactive Waste reviewed the practices and record of DOE shipments of radioactive materials, particularly SNF and TRU. In its report, *Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States* (National Research Council, 2006)[ll], the NAS concluded, that using mostly rail as the transportation mode and the use of dedicated trains is would be advantageous. Additionally, the majority of the dual purpose storage system (storage and transportation) canisters deployed at nuclear utilities exceeds the legal weight truck limits (see Table 2, Nominal Characteristics of Used Nuclear Fuel Transportation Casks) and would require heavy haul transport permitting if transported over the road.

Verification Method: Implementing Decision

C. WMS railcars (individual cars and consists) will be certified by the Association of American Railroads under AAR S-2043, *Performance Specification for Trains Used to Carry High-Level Radioactive Material*.

Basis: DOE and DOD entered a settlement agreement with Union Pacific Railroad Company in 2004 and with the BNSF Railway Company in 2012. The settlement agreement "encompasses transportation in freight cars that are not now in service, some of which have not yet been designed or built. Safe transportation of those cars may require operating,

[jj] TBD

[kk] *Record of Decision on Mode of Transportation and Nevada Rail Corridor for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, NV*, Federal Register, Vol. 69, No. 68, Thursday, April 8 2004.

[ll] TBD

train placement, route, or other restrictions and may not be possible in regular train service."

The settlement agreements further state that, "All cars supplied by the Government Shipper shall be designed and maintained suitable for interchange service and will comply with AAR Construction Standards at the time built and AAR Interchange Rules in effect at the time of the movement," M-1001, *Design, Fabrication, and Construction of Freight Cars*, is contained in Section C, Part II of the AAR MSRP while AAR Standard S-2043 is contained in MSRP Section C, *Car Construction–Fundamentals and Details* (AAR 2011). Thus, the settlement agreement appears to encompass invokes Standard S-2043.

Verification Method: Implementing Decision

9.2 TS Functions and Performance Requirements

The functions' hierarchy is illustrated in Figure 4 and the functional flow is illustrated in Figure 5. In order to ensure consistency and promote clear communications during the functional and requirements analyses process, the top level TS function "Transport SNF, HLW and GTCC-LLW" was broken down into the following sub-functions:

- F 1.1 "Accept SNF, HLW and GTCC-LLW "
- F 1.2 "Transport SNF, HLW and GTCC-LLW to the Storage System"
- F 1.3 "Transport SNF HLW, and GTCC-LLW to the Disposal System"

If applicable, the assumptions provided in Section 9.1 are provided below as performance requirements. Once the functions are validated, they will be deleted from the assumption section. If the assumptions are invalid, they will be removed from the assumption section and the performance requirements will be rewritten.

F 1.0 Transport SNF, HLW and GTCC-LLW

R 1.0.1 The TS shall provide the capability needed to transport the casks defined in Table 2 and Table 3 by waterway vessel or heavy-haul vehicle.

Basis: Waterway vessel or heavy-haul vehicle generator will be needed to transport casks from Generator Sites without rail access to an intermodal transfer point.

R 1.0.2 The TS shall provide the capability needed to transport the transportation casks defined in Table 2 by truck, as necessary.

Basis: Truck transport may provide cost and schedule advantages for transportation of SNF and HLW.

R 1.0.3 The TS shall provide the capability needed to transport the casks defined in Table 3 by rail.

Basis: If the weight of the loaded transportation casks exceeds the limits for truck transport, the casks will need to be transported by rail.

A. The majority of the shipments will be done by rail.

Basis: In April 2004, DOE selected the mostly rail scenario analyzed in the *Final Environmental Impact Statement for a Geologic Repository* *for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (FEIS) [mm] as the transportation mode[nn]. In July 2005, DOE issued a policy stating that dedicated trains would be the usual mode of rail transportation for SNF and HLW to a repository. The National Academy of Sciences' (NAS) National Research Council Committee on Transportation of Radioactive Waste reviewed the practices and record of DOE shipments of radioactive materials, particularly SNF and TRU. In its report, *Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States* (National Research Council, 2006)[oo], the NAS concluded, that using mostly rail as the transportation mode and the use of dedicated trains is would be advantageous. Additionally, the majority of the dual purpose storage system (storage and transportation) canisters deployed at nuclear utilities exceeds the legal weight truck limits (see Table 2, Nominal Characteristics of Used Nuclear Fuel Transportation Casks) and would require heavy haul transport permitting if transported over the road.

B. The WMS shall meet the acceptance rates provided in R 0.4 by providing [TBD] quantities of rolling stock components.

Basis: Same as R 0.4 acceptance rate.

C. The WMS railcars (individual cars and consists) will be certified by the American Association of Railroads under AAR S-2043, *Performance Specification for Trains Used to Carry High-Level Radioactive Material*.

***Basis*:** In April 2004, DOE selected the mostly rail scenario analyzed in the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (FEIS) [pp] as the transportation mode[qq]. In July 2005, DOE issued a policy stating that dedicated trains would be the usual mode of rail transportation for SNF and HLW to a repository. The National Academy of Sciences' (NAS) National Research Council Committee on Transportation of Radioactive Waste reviewed the practices and record of DOE shipments of radioactive materials, particularly SNF and TRU. In its report, *Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the*

[mm] TBD

[nn] *Record of Decision on Mode of Transportation and Nevada Rail Corridor for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, NV*, Federal Register, Vol. 69, No. 68, Thursday, April 8 2004.

[oo] TBD

[pp] TBD

[qq] *Record of Decision on Mode of Transportation and Nevada Rail Corridor for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, NV*, Federal Register, Vol. 69, No. 68, Thursday, April 8 2004.

United States (National Research Council, 2006)[rr], the NAS concluded, that using mostly rail as the transportation mode and the use of dedicated trains is would be advantageous. Additionally, the majority of the dual purpose storage system (storage and transportation) canisters deployed at nuclear utilities exceeds the legal weight truck limits (see Table 2, Nominal Characteristics of Used Nuclear Fuel Transportation Casks) and would require heavy haul transport permitting if transported over the road.

The settlement agreements state that "All cars supplied by the Government Shipper shall be designed and maintained suitable for interchange service and will comply with AAR Construction Standards at the time built and AAR Interchange Rules in effect at the time of the movement," M-1001, *Design, Fabrication, and Construction of Freight Cars*, is contained in Section C, Part II of the AAR MSRP while AAR Standard S-2043 is contained in MSRP Section C, *Car Construction–Fundamentals and Details* (AAR 2011). Thus, The Settlement Agreement reference thus would appear to encompass invokes Standard S-2043.

[rr] TBD

Table 2: Truck Casks

Manufacturer and Model Docket No. CoC Revision	Length without Impact Limiters (in.)	Length with Impact Limiters (in.)	Diameter without Impact Limiters (in.)	Diameter with Impact Limiters (in.)	Empty Weight with Impact Limiters (lb.)	Loaded Weight with Impact Limiters (lb.)	Attributes
NAC International							
NAC-LWT (71-9225, Rev. 59)	199.8	231.9	44.24	65.25	48,000	52,000	Transport, HBU, MOX, PWR, BWR, Bare Fuel, no Canistered Fuel, GTCC
Transnuclear							
TN-LC (71-9358, Rev. 2)	197.5	230.0	38.5	66	43,900	51,000	Transport, HBU, MOX, PWR, BWR, Bare Fuel, no Canistered Fuel, no GTCC
TN FSV (71-9253, Rev. 12)	TBD	247	31	78	TBD	47,000	Transport, HBU, no MOX, graphite fuel, PWR, no BWR, Bare Fuel, Canistered Fuel, no GTCC
General Atomics							
GA-4 (71-9226, Rev. 4)	187.76	233.75	41.81 (across corners) 48.31 (across trunnions)	90.0	48,352	55,000	Transport, no HBU, no MOX, PWR, BWR, Bare Fuel, no Canistered Fuel, no GTCC

Attributes:
BWR= Boiling water reactor
PWR= Pressurized water reactor
HBU= High burnup
MOX= Mixed oxide
GTCC= Greater-than-Class C low-level radioactive waste

Table 3: Rail Casks

Manufacturer and Model Docket No. CoC Revision	Length without Impact Limiters (in.)	Length with Impact Limiters (in.)	Diameter without Impact Limiters (in.)	Diameter with Impact Limiters (in.)	Empty Weight with Impact Limiters (lb.)	Loaded Weight with Impact Limiters (lb.)	Attributes
NAC International							
NAC-STC (71-9235, Rev. 12)	193.0	257.0	99.0	124.0	188,767-194,560	249,290–254,588	Transport, no HBU, no MOX PWR, BWR, Bare Fuel, Canistered Fuel, GTCC
NAC-UMS UTC (71-9270, Rev. 4)	209.3	273.3	92.9	124.0	178,798	248,373–255,022	Transport, HBU, no MOX, PWR, BWR, no Bare Fuel, Canistered Fuel, GTCC
MAGNATRAN 71-9356	211.4	322	110	128	208,000	312,000	Transportation CoC pending PWR, BWR
Holtec							
HI-STAR 100 (71-9261, Rev. 9)	203.1	305.88	96.0	128.0	179,710	279,893–272,622	Transport, Storage, no HBU, MOX, PWR, BWR, no Bare Fuel, Canistered Fuel, no GTCC
HI-STAR HB (71-9261, Rev. 9)	128.0	230.8	96.0	128.0[a]	--[b]	187,200	Transport, Storage, no HBU, no MOX, no Bare Fuel, no PWR, BWR (Humboldt Bay only), no GTCC
HI-STAR 180 (71-9325, Rev. 0)	174.37	285.04	106.30	128.0	< 308,647	308,647	Transport, HBU, MOX, PWR, no BWR, Bare Fuel, no Canistered Fuel, no GTCC
HI-STAR 60 (71-9336, Rev. 0)	158.94	274.37	75.75	128.0[a]	<164,000	164,000	Transport, no HBU, no MOX, PWR, no BWR, Bare Fuel, no Canistered, no GTCC
Transnuclear							
MP187 (71-9255, Rev. 12)	201.50	308.00	92.50	126.75	190,200	265,100–271,300	Transport, no HBU, MOX, PWR, no BWR, no Bare Fuel, Canistered Fuel, no GTCC

Manufacturer and Model Docket No. CoC Revision	Length without Impact Limiters (in.)	Length with Impact Limiters (in.)	Diameter without Impact Limiters (in.)	Diameter with Impact Limiters (in.)	Empty Weight with Impact Limiters (lb.)	Loaded Weight with Impact Limiters (lb.)	Attributes
MP197 (71-9302, Rev. 5)	208.0	281.25	91.5	122.00	176,710	265,100	Transport, no HBU, no MOX, PWR, BWR, no Bare Fuel, Canistered Fuel, no GTCC
MP197HB (71-9302, Rev. 5)	210.25	271.25	97.75	126.00	179,000	267,390	Transport, no HBU **[TBV],** no MOX, PWR, BWR, no Bare Fuel, Canistered Fuel, GTCC
TN-40 (71-9313, Rev. 0)	175.0	261.0	99.52	144.00	--[c]	271,500	Transport, Storage, no HBU, no MOX, PWR (Prairie Island only), no BWR, Bare Fuel, no Canistered Fuel, no GTCC
TN40HT	170.0	261.0[a]	101.00	144[a]	--[c]	242,343	No transportation CoC PWR
TN-68 (71-9293, Rev. 3)	197.25	271.00	98.00	144.00	<272,000	272,000	Transport, Storage, no HBU, no MOX, no PWR, BWR, Bare Fuel, no Canistered Fuel, no GTCC
Energy Solutions							
TS125 (71-9276, Rev. 4)	210.4	342.4	94.2	143.5	196,118	285,000	Transport, HBU, MOX, PWR, BWR, no Bare Fuel, Canistered Fuel, no GTCC

Source: Greene et al. (2013).
a. Estimated
b. HI-STAR HB transportation casks are already loaded, so they will not be shipped empty.
c. TN-40 transportation casks are authorized for single use shipments and would not be shipped empty. TN40HT transportation casks are also assumed to be authorized for single use shipments and would not be shipped empty.
Attributes:
HBU= High burnup
MOX= Mixed oxide
GTCC= Greater-than-Class C low-level radioactive waste
BWR= Boiling water reactor assemblies
PWR= Pressurized water reactor assemblies

R 1.0.4 Capability is needed to decontaminate transportation casks.
Basis: Cask must be decontaminated prior to performing maintenance activities.

R 1.0.5 Capability is needed to maintain transportation casks.
Basis: Maintenance is required to ensure operability and integrity of equipment.

R 1.0.6 Capability is needed for maintenance of fleet vehicles (heavy haul vehicle and rail) and ancillary equipment.
Basis: Maintenance is required to ensure operability and integrity of equipment.

F 1.1 Accept SNF, HLW and GTCC-LLW[ss]

R 1.1.1 The TS shall accept the material defined in R 0.1.
Basis: Same as R 0.1.

R 1.1.2 The TS shall accept the canisters defined in R 0.2.
Basis: Same as R 0.2.

R 1.1.3 The TS shall accept the materials per the rates defined in R 0.4.
Basis: Same as R 0.4.

F 1.2 Transport SNF, HLW and GTCC-LLW to SS

R 1.2.1 The TS shall transport the materials per requirements R 0.1 through R 0.4.
Basis: Same as the bases for requirements R 0.1 through R 0.4.

R 1.2.2 The TS shall transport components per requirements R 1.0.1 through R 1.0.3.
Basis: Same as the basis for requirements R 1.0.1 through R 1.0.3.

F 1.3 Transport SNFSNF**, HLW and GTCC-LLW from WMS to Disposal System**

R 1.3.1 The TS shall transport HLW from storage to the DS.
Basis: Requisite Step

R 1.3.2 The TS shall transport GTCC-LLW from storage to the DS.
Basis: Requisite Step

R 1.3.3 The TS shall transport SNF in disposal containers from storage to the DS.

[ss] In general, implementation of these requirements should only generate documentation such as contracts, memorandum of understandings, waste acceptance criteria, response plans, communication plans, routing plans, shipping papers, and titles, etc. Unlike performance requirements, implementation of these requirements should not generate systems, structures, or components (SSCs). These requirements also help ensure consistency and promote clear communications between the purchaser and the custodian.

Basis: Requisite Step

R 1.3.4 The TS shall transport SNF from storage to packaging.
Basis: [**TBD** – Trade studies will be conducted to evaluate alternatives]

R 1.3.5 The TS shall **t**ransport SNF from R&D to disposal.
Basis: Requisite Step

STORAGE SYSTEM – INTERIM STORAGE FACILITY

10. Storage System – Interim Storage Facility

The section establishes the initial F&Rs for the Storage System (SS) (i.e., Interim Storage Facility) technical baseline.

10.1 ISF Assumptions

The WMS assumptions provided in Section 4 were developed to bridge the gap between the Program Guidance provided in Section 3 and the ISF requirements.

A. Canistered SNF will be stored dry at the pilot ISF in the same canister it was received.
Basis: The pilot will only accept SNFSNF stored in dry storage canisters to facilitate the design, licensing, and initial operations of the pilot**.**
Verification Method: Implementing Decision

B. Canistered SNF and HLW will be stored dry at the larger ISF in the same canister it was received.
Basis: Packaging will not be implemented prior to storage at the ISF **[TBV].**
Verification Method: Implementing Decision

C. Packaging for disposal will be co-located with the Storage System (larger ISF) and/or Disposal System.
Basis: **[TBD-** Trade studies will be conducted to evaluate alternatives.]
Verification Method: Implementing Decision

D. Disposal acceptance criteria will be established by the time the WMS needs to discharge material to Disposal System
Basis: SNF discharged from WMS must meet the receiving facility acceptance criteria.
Verification Method: Implementing Decision

E. The R&D system will be co-located with the ISF**.**
Basis: If the ISF and R&D system are independent; many of the systems, structures, and/or components would be duplicated at additional capital cost and operating cost.
Verification Method: Implementing Decision

F. R&D functions will be conducted via dry mechanisms. **[TBV]**
Basis: R&D functions will require the removal of used fuel assemblies from dry storage systems. Per the *Gap Analysis to Support Extended Storage of Used Nuclear Fuel*[tt] this is the preferred process.
Verification Method: Implementing Decision
Basis: **[TBD -** Trade studies will be conducted to evaluate alternatives]

[tt] *The Gap Analysis to Support Extended Storage of Used Nuclear Fuel, page TBD, paragraph TBD*

10.2 ISF Functions and Performance Requirements

The functions' hierarchy is illustrated in Figure 4 and the functional flow is illustrated in Figure 5. The SS top-level function (F 2.0) "Store SNF, HLW and GTCC-LLW" has been allocated to the SS, which includes both the pilot ISF and the larger ISF. In order to ensure consistency and promote clear communications during the functional and requirements analyses process, the top level SS function was broken down into the following sub-functions:

- F 2.1 "Receive SNF HLW and GTCC-LLW"
- F 2.2 "Transfer SNF HLW and GTCC-LLW to the SS"
- F 2.3 "Store SNF HLW, and GTCC-LLW at Storage Facility"
- F 2.4 "Discharge SNF HLW, and GTCC-LLW from the SS"
- F 2.5 "Package SNF for Disposal"

If applicable, the assumptions provided in Section 10.1 are provided below as performance requirements. Once the functions are validated, they will be deleted from the assumption section. If the assumptions are invalid, they will be removed from the assumption section and the performance requirements will be rewritten.

F 2.0 Store SNF, HLW and GTCC- LLW

R 2.0.1 The pilot ISF and larger ISF could potentially be co-located **[TBV]**

Basis: The Administration's Strategy states, "A consent-based siting process could result in more than one storage facility and/or repository, depending on the outcome of discussions with host communities; the Nuclear Waste Policy Act of 1982 (NWPA) envisaged the need for multiple repositories as a matter of equity between regions of the country. As a starting place, this Strategy is focused on just one of each facility."

Basis: The Administration's Strategy states, "A larger, consolidated interim storage facility, potentially co-located with the pilot facility and/or geological repository that provides the needed flexibility in the waste management system and allows for important near-term progress in implementing the federal commitments."

Basis: If the pilot ISF and larger ISF are independent facilities, many of the systems, structures and/or components would be duplicated at additional licensing cost, capital cost and operating cost.

R 2.0.2 A modular design and operating concept will be used to design and operate the pilot ISF and the larger ISF. Modular concepts may be used to either expand the capacity of the integrated facility and/or expand the functional capabilities of the facility. Modules will be deployed individually as needed to support changes in the facility mission. [**TBV]**

Basis: The Administration's Strategy states "A phased, adaptive, and consent-based approach to siting and implementing a comprehensive management and disposal system shall be used." A module design and operating concepts is an effective and efficient alternative for satisfying this guidance.

R 2.0.3 The SS shall be designed for the capacities specified in Table 4. [**TBV]**

Table 4: Storage System Capacities

Type	Pilot ISF	Larger ISF
Commercial UNF	5,000 to 10,000 MTHM **[TBV]**	70,000 MTHM (nominal) **[TBV]**
Government-Owned UNF	0 **[TBV]**	**[TBD]**
Government-Owned HLW	0 **[TBV]**	**[TBD]**
Commercial GTCC -LLW	15 to 27 Casks[uu] **[TBV]**	**[TBD]**

Basis Nuclear Fuels Storage and Transportation Planning Project (NFST) is considering a range between 5,000 and 10,000 metric tons uranium (MTU) for the pilot and a nominal larger ISF capacity up to 70,000 MTU.

Pilot ISF

The pilot ISF capacity range under consideration is large to reflect the discussion and dialog within the nuclear industry and congressional stakeholders. The Blue Ribbon Commission on America Nuclear Future (BRC) focused their discussion on the spent nuclear fuel (SNF) inventory from 10 reactors on 9 sites that have ceased all nuclear power operations. For reactors shutdown prior to 2000, the remaining fuel has been moved to dry storage to support reactor decommissioning, which is complete at several of the sites (e.g. Big Rock Point). These sites are often referred to as "stranded sites" since after reactor decommissioning, the only nuclear operations support SNF storage. The inventory at these sites totals 7,649 assemblies containing 2,813 MTU (see details in the latest revision of *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*, FCRD-NFST-2013-000263).[vv]

During the first six months of 2013 nuclear power plant operators ceased operations at one additional reactor (Kewaunee) and announced that restart efforts at 3 reactors (Crystal River, and San Onofre units 2 and 3) were being stopped. This group of reactors is often referred to as "early shutdown" reactors since operations were ceased prior to reaching the 60 year operating lifetime allowed by the Nuclear Regulatory Commission (NRC).[ww] Vermont Yankee permanently shut down in December 2014. Oyster Creek has announced an early shutdown date of 2019. The early shutdown reactors sites are similar to those considered stranded in that there will be no other nuclear operations on the sites following pool de-inventory and reactor decommissioning (which is not immediately planned for these sites). This fuel is expected to be moved to dry storage although the timetable

[uu] GTCC from 9 stranded sites + assumption of 2 casks per reactor from the remaining shutdowns.

[vv] TBD

[ww] The NRC initially licenses power reactors for a 40 year period and allows a 20 year extension for a total of 60 years.

for movement is uncertain. The inventory at these sites is estimated as 11,230 assemblies containing 3,582 MTU (see details in the latest revision of *Nuclear Fuels Storage and Transportation Planning Project Inventory Basis,* FCRD-NFST-2013-000263). [xx]

In the future these categories could be combined. These additional plants will bring the total stranded inventory (15 reactors on 12 sites) to about 18,900 assemblies containing approximately 6,400 MTU.

The *Nuclear Waste Administration Act of 2013* (not enacted), introduced in the Senate authorizes a "pilot facility for priority waste"[yy] and defines priority waste as "spent nuclear fuel removed from a civilian nuclear reactor that has been permanently shut down."[zz] Three additional reactors are permanently shut down on sites with continued nuclear power operations. These reactors shutdown prior to 2000 have an inventory of 3,933 assemblies containing approximately 646.8 MTU.

As currently defined by the *Nuclear Waste Administration Act of 2013* (not enacted), the total priority fuel anticipated at the start of the pilot operations in 2021tt could be nearly 23,000 assemblies containing over 7,000MTU.

The NFST is considering a slightly larger range (up to 10,000MTU) to allow for uncertainty in the number of shut down reactors prior to 2021 **[TBV]**.

Larger ISF

The NFST is considering a larger ISF capacity of a nominal 70,000 MTU during full operations. This capacity is based on the policy dates established for the start of the larger ISF in 2025 **[TBV]**[aaa] and the repository in 2048 **[TBV]**.[bbb] This 23-year difference multiplied by the acceptance rate provides the capacity. Currently NFST is assuming a nominal acceptance rate of 3,000 MTU/yr. Therefore: the capacity is 23 years times 3,000 MTU/yr = 69,000 MTU or a nominal 70,000 MTU.

The NFST is investigating a range (> 3,000 MTU/yr) of acceptance rates in an effort to determine the impact on the "expected federal liabilities."[ccc] The lower end of this range (3,000 MTU/yr) is based on

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[xx] TBD

[yy] *Waste Administration Act of 2013* (not enacted), Sec 303 (1)

[zz] *Waste Administration Act of 2013* (not enacted), Sec 103 (22)

[aaa] "*Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*," p. 2, outlines a policy which "Advances toward the siting and licensing of a larger interim storage facility to be available by 2025."

[bbb] "*Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*," p. 2, outlines a policy which "Makes demonstrable progress on the siting and characterization of repository sites to facilitate the availability of a geologic repository by 2048 **[TBV]** that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities."

[ccc] Ibid

150% of the current quantity of fuel being discharged annually.[ddd] In order to limit and then end the liability costs.[eee] 3,000 MTU/yr is also the historical repository program acceptance rate.[fff] Higher acceptance rates would limit the liability sooner but would also result in an expansion of the larger facility capacity and a greater annual operating cost to support the increased acceptance rate.

The larger ISF capacity will be built over the entire operating lifetime. The initial capacity will likely be that required for the first 2 to 3 years of operations or 6,000 to 9,000 MTU. The remaining capacity will be built as needed. In a similar fashion the receipt facilities can be built in phases or modules. For example 1,500 MTU/yr receipt facilities would allow a sequential built-out of 1,500, 3,000 and 4,500 MTU/yr if required. Each phase could also be tailored to a specific SNF package form such as dry storage canister (initially) and bare fuel (later). Receipt of bare fuel could reduce the liability sooner versus continued acceptance of only canistered dry fuel. [ggg]

NFST is currently assuming the pilot and larger ISF are co-located on the same site;[hhh] therefore, the receipt facilities and other infrastructure items are planned to be shared. The pilot operations are assumed to stop when the larger ISF is available; therefore, the capacities of the two facilities are additive.

R 2.0.4 The SS shall be sited on a minimum of 640 acres (one square mile) **[TBV].**

Basis: As defined in *Preliminary Site Factors and Considerations for Interim Used Fuel Storage Facilities*, FCRD-NFST-2013-000370,[iii] the Site location must include space, terrain, utilities and infrastructure to support the following four (4) major operations functions:

1. Receiving/Servicing Operations for rail and truck casks, trailers, rail cars and ancillary equipment.
2. Fleet Operations infrastructure for rail and truck operations including spare parts and routine maintenance.
3. Storage operations to support a fleet of trailers, a fleet of specialized railcars and transport casks and their ancillary equipment.

[ddd] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*" p. 3 states "Currently more than 68,000 metric tons heavy metal (MTHM) of used nuclear fuel are stored at 72 commercial power plants around the country with approximately 2,000 MTHM added to that amount every year."

[eee] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*," p. 3, "outlines a strategy that is intended to limit, and then end, liability costs by making it possible for the government to begin performing on its contractual obligations."

[fff] "Office of Civilian Radioactive Waste Management (OCRWM), *Civilian Radioactive Waste Management System Requirements Document*, DOE/RW-0406, Rev. 8. September 12, 2007.

[ggg] *Used Fuel Management System Architecture Evaluation, Fiscal Year 2013*, FCRD-NFST-2013-000377, Rev. 1, October 31, 2013.

[hhh] *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*, p. 2, outlines a strategy in which "A consent-based siting process could result in more than one storage facility and/or repository, depending on the outcome of discussions with host communities; the Nuclear Waste Policy Act of 1982 (NWPA) envisaged the need for multiple repositories as a matter of equity between regions of the country. As a starting place, this Strategy is focused on just one of each facility."

[iii] .TBD

4. Administrative, physical plant, waste and emergency operation center activities.

The dose limits specified in 10 CFR 72.104 and 10 CFR 72.106 are used as criteria to establish site boundaries and determine that an ISF at the proposed site would be able to comply with the NRC regulations. The area required is based on the location of the storage units and the distance between the site boundary and facilities or structures where spent fuel is handled or stored. However, the total area required for the ISF facility varies with the storage systems deployed and incremental modular expansion.

R 2.0.5 Provide a platform for ongoing R&D to better understand how the storage system will perform over time.

Basis: Per the *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy,* January, 2012,[jjj] "A federal facility with spent fuel receipt, handling and storage capabilities can support other valuable activities that would benefit the waste management system. These include long-term monitoring and periodic inspection of dry storage systems and work on improved storage methods. Many current dry cask systems lack instrumentation to measure key parameters such as gas pressure, the release of volatile fission products, and moisture. Some of this work can be done in laboratories, but key aspects require the ability to handle and open loaded spent fuel storage containers and examine the fuel. A consolidated storage facility with laboratory and hot cell facilities and access to a substantial quantity and variety of spent fuel would provide an excellent platform for ongoing research and development to better understand how the storage systems currently in use at both commercial and DOE sites perform over time."

A. As a minimum, the R&D system shall be capable of performing research and development activities on the following components

- Existing commercial LWR SNF
- Existing DOE-owned SNF
- Commercial pool
- Commercial dry storage systems
- Commercial SNF transportation systems (primary focus on transportation of high-burn up SNF)

Basis: These components make the required SSCs that are important to safety and to support aging management considerations. [kkk]

[jjj] *Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy,* January, 2012, p 39, paragraph 4.
[kkk] *Used Nuclear Fuel Storage and Transportation Research, Development, and Demonstration Plan,* FCRD-FCT-2012-000053, Rev. 0, April 2012, p. 4, Section 1.2.2.

B. The R&D system shall have an annual processing rate of a maximum of six cask/year **[TBV]** including:

- Opening at most six **[TBV]** casks/year.
- Disassembling and examining 1 (one) assembly per cask **[TBV].**
- Retrieving and examining 1 (one) rod per assembly [**TBV].**
- Cutting and examining 3 segments per rod [**TBV].**

Basis: [**TBD -** The size of the R&D system, size of the major processing areas, and number of pieces of equipment shall be based on an annual processing rate. Trade studies will be conducted to verify the assumed annual processing rate of samples (casks, canisters, rods, segments, cladding, and fuel) needed to assure that the DOE-NE RD&D findings are sufficiently robust to support a comprehensive regulatory regime.

C. The R&D system shall be capable of examining SSCs in storage via *in-situ* inspection and monitoring: [**TBV]**

Basis: **[TBD –** The R&D Program includes requirements to examine storage pads, vaults as well as dry storage casks, canisters, and instrumented canisters while in dry storage to determine how these structures and components will perform over time. Trade studies are needed to determine the methodologies that will be used for conducting examinations.**]**

D. The R&D system shall receive the SNF materials from storage

Basis: **[TBD –**Trade studies will be conducted to specify the receipt requirements (e.g., facility through-put, facility size, number of cranes, size of cranes, shielding requirements, temporary storage requirements, etc.)]

E. The R&D System shall open storage canisters specified in R 0.2 by removing welded and/or bolted canisters' lids.

Basis: Canister lid (bolted or welded) removal is required to access fuel.

i. Canisters shall be maintained under inert gas atmosphere in order to limit oxidation of the cladding and fuel during R&D fuel handling steps.

Basis: R&D functions will require the removal of used fuel assemblies from dry storage systems. Per the *Gap Analysis to Support Extended Storage of Used Nuclear Fuel*[lll] this is the preferred process.

ii. The cavity gas shall be sampled and monitored.

[lll] *The Gap Analysis to Support Extended Storage of Used Nuclear Fuel, page TBD, paragraph TBD*

Basis: Gas sampling will detect the presence of a cladding breach or the accumulation of flammable gases during lid removal operations.

F. The R&D System shall retrieve fuel assemblies from storage canister specified in R 0.2
Basis: **[TBD-** Trade studies will be conducted to evaluate alternatives.]

G. The R&D system shall be designed to pull selected rods from assemblies.
Basis: Individual fuel rods are removed from the fuel assemblies in order to perform fuel rod examinations.

*i***.** The removal force shall be monitored

Basis: The removal force is monitored to provide information regarding sticking caused by structural changes of the fuel assembly during storage.

H. The R&D system shall be designed to section the fuel rods into segments.
Basis: Rods are segmented to provide materials for testing.

J. The R&D system shall remove fuel from rod segments.
Basis: Fuel is removed from selected rod segments to allow non-fueled clad mechanical properties testing. Both mechanical and chemical methods for fuel removal will be required depending on the specific test protocol required.

K. The R&D system shall prepare the cladding for testing.

Basis: Cladding is cut to size, cleaned, mounted, and polished to produce metallographic and tensile samples [**TBV].**

L. The R&D system shall seal the canisters (existing- if the original canister can be revised or new-if the contents have been transferred to an unused canister); containing assemblies not selected for R&D, prior to sending the canister store the SS.
Basis The canister will be stored

i. The lid shall be placed on the canister and closed, which can be either bolted or welded.
Basis: Requisite step

ii. Moisture shall be removed from the canister by vacuum drying to meet NRC regulation. Moisture and oxygen are removed prior to storage to limit oxidation of the cladding and fuel.
Basis: Requisite step

iii. The canisters shall be inerted and leak tested before moving the canister back to the ISF.
Basis: Requisite step

M. The WMS shall transfer canisters to storage.
Basis: The canister will be returned to storage

N. Provide required testing capabilities.
Basis: **[TBD –** Trade studies will be conducted to identify the testing requirements]

F 2.1 Receive SNF, HLW and GTCC-LLW

R 2.1.1 The SS shall receive the materials specified in R 0.1.
Basis: Same as R.0.1

R 2.1.2 The SS shall receive materials per the rates specified in R 0.4.
Basis: Same as R 0.4

R 2.1.3 The SS shall receive the casks defined in R 1.2.2
Basis: Same as R 1.2.2

R 2.1.4 The SS shall transfer casks to a cask handling area for unloading and loading operations
Basis: The cask handling area will provide radiation shielding and confinement during loading and unloading operations.

R 2.1.5 The SS shall perform post transportation inspections, radiological surveys, decontamination, and security receipt inspections.
Basis: Inspections are required to ensure integrity of equipment, and compliance with regulations for packaging and shipment. (10 CFR71 and 49 CFR173) [**TBV**].

Basis: Prior to opening the transportation cask, the interior of the cask shall be sampled to determine if the interior has been contaminated.

R 2.1.6 The SS shall open the casks defined in R 1.2.2.
Basis: The cask shall be opened to retrieve the canistered or bare fuel

R 2.1.7 The SS shall unload the canisters defined in R 0.2 from the cask.
Basis: The canisters shall be unloaded so that they can be prepared for or sent to long term storage.

R 2.1.8 The SS shall unload the bare fuel from the cask.

Basis: The bare fuel shall be unloaded so that it can be prepared for or sent long term storage.

F 2.2 Transfer SNF, HLW and GTCC-LLW to Storage

R 2.2.1 The SS shall transfer SNF, HLW, and GTCC - LLW to storage
Basis Not Applicable. Refer to lower level requirements.

A. The SS shall transfer dry storage canisters to storage **[TBV.]**
Basis Requisite step

B. The SS shall transfer damaged/failed fuel cans to storage **[TBV]**
Basis: **S**NF in damaged/failed fuel cans are included in some dry storage canisters received and expected in some bare fuel shipments.

C. The SS shall transfer bare fuel assemblies to storage**[TBV]**
Basis: Receipt of bare fuel could reduce the liability sooner versus continued acceptance of only canistered dry fuel.

D. The SS shall transfer GTCC-LLW canisters to storage **[TBV.]**
Basis: Requisite step

E. The SS shall transfer government-owned SNF canisters **t**o storage **[TBV.]**
Basis: Requisite step

F. The SS shall transfer government-owned HLW canisters to storage **[TBV.]**
Basis: Requisite step

F 2.3 Store SNF, HLW and GTCC - LLW at Storage

R 2.3.1 The SS shall store the materials specified in R 0.1.
Basis: Same as R.0.1.

R 2.3.2 The SS shall store the canisters specified in R 0.2.
Basis: Same as R.0.2.

R 2.3.3 Canistered SSNF will be stored dry at the pilot ISF in the same canister it was received.
Basis: The pilot will only accept SNF stored in dry storage canisters to facilitate the design, licensing, and initial operations of the pilot**.**

R 2.3.4 Canistered SNF and HLW will be stored dry at the larger ISF in the same canister it was received.
Basis: Packaging will not be implemented prior to storage at the ISF **[TBV].**

R 2.3.5 Damaged/failed fuel will be stored dry at the larger ISF in the same canister it was received
Basis: Same as R 2.3.4

R 2.3.6 Bare fuel including some received in damaged fuel cans will be stored at the larger ISF in **[TBD]** configuration.

Basis: **[TBD-** Trade studies will be conducted to evaluate alternatives.]

R 2.3.7 The SS shall provide inspection and monitoring capability for material in storage.
Basis: The material in storage will need to be inspected and monitoring to satisfy the **[TBD]** licensing requirements.

F 2.4 Discharge SNF, HLW and GTCC-LLW from Storage

R 2.4.1 Discharge HLW from storage to TS.
Basis: Requisite Steps

R 2.4.2 Discharge GTCC-LLW from storage to TS.
Basis: Requisite Steps

R 2.4.3 Discharge SNF from storage.
Basis: Requisite Steps

A. Discharge SNF in disposal canisters from storage to TS (Direct to DS).
Basis: Requisite Steps

B. Discharge SNF from storage to TS (packaging not co-located with DS)
Basis: Requisite Steps

C. Discharge SNF from storage to Packaging (packaging co-located with SS)
Basis: Requisite Steps

R 2.4.4 SNFSNF debris generated by WMS onsite operations shall be transferred to the TS and/or the packaging for disposal.
Basis: **[TBD-** Trade studies will be conducted to evaluate alternatives.]

R 2.4.5 GTCC-LLW generated by WMS onsite operations shall be transferred from WMS to TS.
Basis: Requisite Step

R 2.4.6 LLW generated by WMS onsite operations shall be transferred from WMS to TS.
Basis: Requisite Step

R 2.4.7 Hazardous, mixed, and sanitary waste generated by WMS onsite operations shall be managed per site specific requirements **[TBV].**
Basis: Requisite Step

F 2.5 Package SNF for Disposal

R 2.5.1 Receive SNF for packaging for disposal
Basis: Requisite step

R 2.5.2 Open storage canisters specified in R 0.2 by removing welded and/or bolted canisters' lids.
Basis: Requisite step

A. Canisters shall be maintained under a positive pressure using inert gas or under water in order to limit oxidation of the cladding and fuel during storage.
Basis: Canister opening requires the canister inert gas to be monitored for fission gases, vented, and the lid removed to allow access to the fuel.

B. The cavity gas shall be sampled and monitored.

Basis: Gas sampling will detect the presence of a cladding breach or the accumulation of flammable gases during lid removal operations.

R 2.5.3 Retrieve fuel assemblies from storage canisters specified in R 0.2
Basis: Requisite step

R 2.5.4 Package fuel assemblies for disposal
Basis: Requisite step

R 2.5.5 Seal the new storage system/canister.

A. Moisture shall be removed from the new storage system/canister using methods acceptable to NRC.
Basis: Moisture and oxygen are removed prior to storage to limit oxidation of the cladding and fuel.

B. The lid must be placed on the canister and sealed, which can be either bolted or welded.
Basis: Sealing of canister ensures confinement of radioactive materials during transport and disposal.

C. The canisters must be inerted and leak tested before transferring it to the TS.
Basis: Seal and leak test ensures confinement of radioactive materials during transport and disposal.

R 2.5.6 Release the new canister into **[TBD]** configurations

Basis: **[TBD-** Trade studies will be conducted to evaluate alternatives.]R 2.5.7 Packaging for disposal will be co-located with storage (larger ISF) and/or Disposal System.

Basis: **[TBD-** Trade studies will be conducted to evaluate alternatives.]

WMS INTERNAL INTERFACES

11. WMS Internal Interfaces

WMS internal interface systems shall include 1) process support 2) ES&H discharge, 3) secondary waste management, and 4) infrastructure systems. Examples of these systems and their interfaces are shown below in Figure 7.

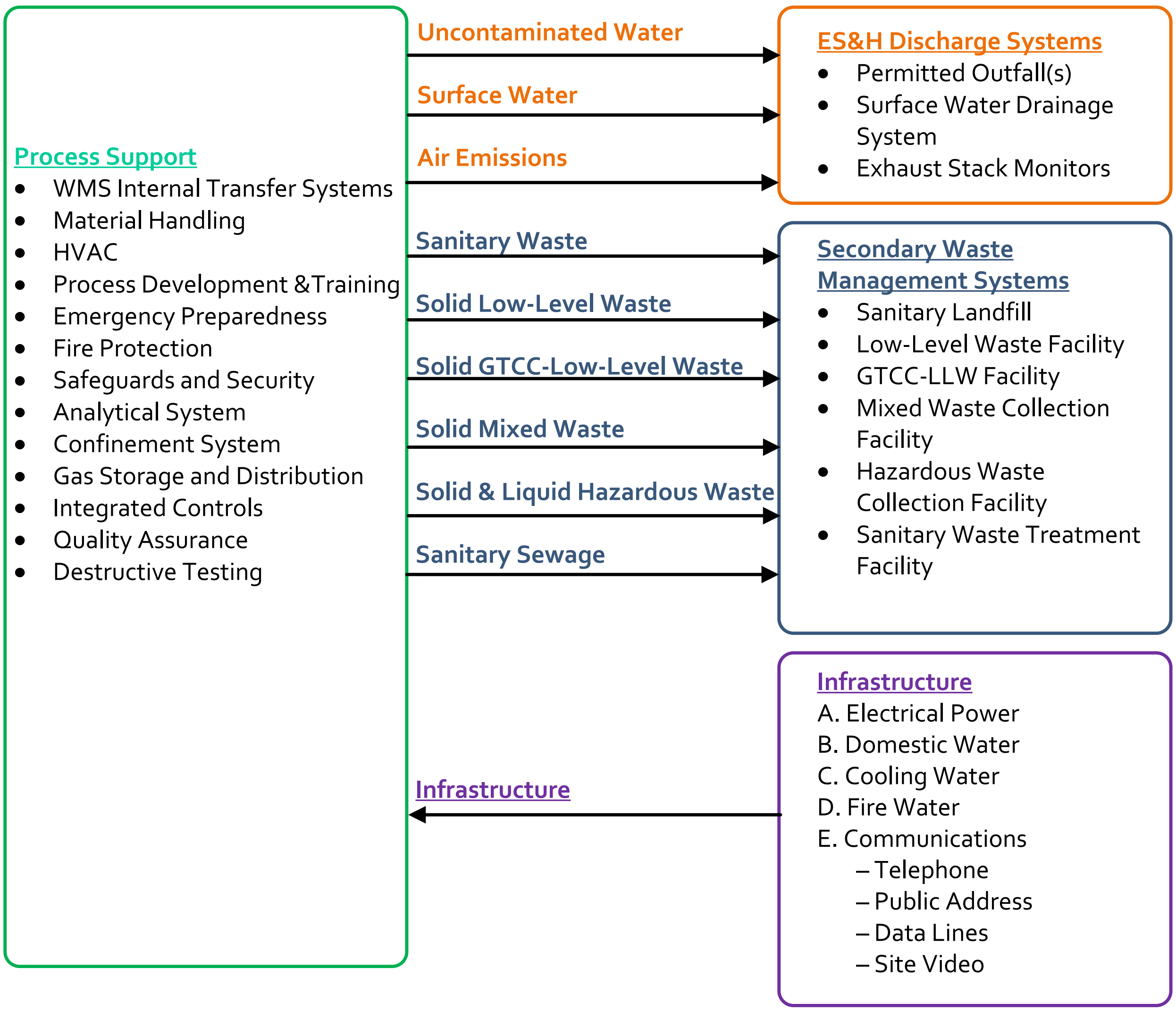

Figure 7 – WMS Internal Interfaces

12. REFERENCES

See Appendix C for a "Consolidated NFST Reference List," which in the future will replace the below references.

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A-1. DEFINITIONS

This section provides definitions of key terms used in this document. These definitions are not requirements but are provided to ensure consistency when describing the WMS and its requirements.

Assumptions	In order to bridge the gap between the Program Guidance provided in the *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*, January 2013, and the performance requirements provided in this documents, several assumptions were developed. If applicable, these assumptions were written as performance requirements. If the assumptions are validated, they will be deleted from the assumptions sections. If they are found to be invalid, they will be removed from the assumptions section and the performance requirements will be rewritten.
Bare Fuel	Bare fuel is uncanistered, individual nuclear fuel assemblies.
Bases	In support of the requirements, bases are provided to describe the reason and/or source for the performance requirement.
Canister	A canister is the structure surrounding the waste form (e.g., HLW immobilized in borosilicate glass) that facilitates handling, storage, transportation, and/or disposal. A canister is a metal receptacle with the following purpose: (1) for solidified HLW, its purpose is a pour mold and (2) for SNF, it may provide structural support for intact SNF, loose rods, nonfuel components, or containment of radionuclides.
Cask	A cask is a container for shipping and/or storing SNF (bare or canistered) and/or canistered HLW that is certified by the NRC.
Damaged and Failed Fuel	These terms are used interchangeably by the Utilities. In this document, "damaged and failed fuel" is referred to as "damaged/failed fuel" Below are failed fuel categories defined 10 CFR 961 Failed Fuel: Class F-1: Visual Failure or Damaged Class F-2: Radioactive "Leakage" Class F-3: Encapsulated
Function	A function is a primary statement of purpose; it defines what a system or subsystem must accomplish to meet the system mission.

Functional Flow Diagram	A functional flow diagram shows a logical interrelationship between the functions to be performed
Functional Hierarchy Diagram	A functional hierarchy diagram provides the hierarchical relationship between the functions to be performed. The top-level function was broken down into sub-functions. The sub-functions are the actions or capabilities necessary to perform the top-level function.
Government Owned High Level Waste (HLW)	Government owned HLW is HLW that is currently managed by the government.
Government Owned Spent Nuclear Fuel (SNF)	Government owned SNF is fuel that is currently managed by the government, and includes fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated.
Great than Class C (GTCC) - Low Level Waste (LLW)	In this document Low Level Radioactive Waste (LLRW) is referred to as Low Level Waste (LLW). GTCC - LLRW is defined by the NRC as LLRW that has radionuclide concentrations exceeding the limits for Class C LLRW in 10 CFR 61.55. The NRC identifies four classes of LLRW in 10 CFR 61.55 for disposal purposes based on the concentrations of specific long- and short-lived radionuclides: Classes A, B, C, and GTCC. Classes A, B, and C LLRW can be disposed of in near-surface disposal facilities licensed by the NRC or an Agreement State.
High Level Waste (HLW)	In this document High Level Radioactive Waste (HLRW) is referred to as High Level Waste HLW). High Level Radioactive Waste (HLRW) is (1) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that the Nuclear Regulatory Commission, consistent with existing law, determines by rule requires permanent isolation.
Module or modular	Module or modular means the minimum functional capabilities constructed at one time, Modular concepts: ▪ Allows for additional capacity ▪ Allows for additional functional capability”

Packaging

As defined in 10 CFR 71.4, packaging is the assembly of components necessary to ensure compliance with packaging requirements of 10 CFR Part 71. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling and absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging. (as defined in 10 CFR 71.4)

In this document, packaging also refers to the assembly of components necessary to ensure compliance with transportation and disposal requirements.

Program Guidance

Program Guidance is described in Section 3 of this document and is based on The functions and requirements provided in this document were based on previously approved reports as well as guidance provided by the *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*. The relevant requirements were extracted from the Strategy and captured under the section titled “Program Guidance”.

Programmatic Interfaces

Programmatic interfaces are the administrative activities that must be performed to transport SNF, HLW, and GTCC-LLW between the 1) Generator Sites and the TS, and 2) the TS and the Disposal System. In general, implementation of these requirements should only generate documentation such as contracts, memorandum of understandings, waste acceptance criteria, response plans, communication plans, routing plans, shipping papers, titles, etc. Unlike performance requirements, implementation of these requirements should not generate systems, structures, or components (SSCs). These requirements also help ensure consistency and promote clear communications between the purchaser and the custodian. In some cases, regulations are included with the requirements as bases.

Resource Documents

Several previously generated reports were used as resources for this document. They are captured as resource documents and reference documents. In the “Resource Section” the bases for why they were selected is provided.

Requirement

A requirement is a qualitative or quantitative statement that describes a characteristic or constraint that must be met for a system, product or process to be acceptable. Requirements define how well a function must perform. The following types of requirements were used in this document.

Regulatory Requirement

A requirement that is established by, or derived from, key Federal laws and regulations.

Performance Requirement	A requirement that defines the capability the WMS or one of its elements must have to accomplish its allocated function.
External and Internal Interface Requirement	A requirement that applies to the inputs to, or outputs from, the functions; or the physical connection or dependence between architectural items.
Shutdown Reactor	A commercial nuclear power reactor for which the NRC license has been terminated.
Shutdown Reactor Site	A commercial nuclear power generator location where all nuclear reactors have their NRC licenses terminated.
Spent Nuclear Fuel (SNF) and Used Nuclear Fuel (UNF)	The terms “used nuclear fuel” or UNF and “spent nuclear fuel” or SNF are used interchangeably because the distinction between terms is unimportant. SNF (or UNF) is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.
To Be Determined (TBDs)	A **TBD** indicates places where descriptive information or quantitative values are not yet available.
To Be Verified **(**TBVs**)**	A **TBV** is used when descriptive or quantitative information is provided but requires further development because it: ➢ Is preliminary and unapproved; ➢ Involves an uncertain design feature; ➢ Has insufficient technical justification; ➢ Needs verification; or ➢ Creates a discrepancy or inconsistency.
Transfer	Movement of materials and components within the WMS.
Transport	Movement of materials and components outside the WMS. Transport meets 10 CFR 71 requirements.

A-2. DETAILED REQUIREMENTS OF 10 CFR Part 72, NUREG-1567, Regulatory Guide 3.48, Regulatory Guide 3.60, Regulatory Guide 3.50 (Rev 2), Regulatory Guide 3.53, and Regulatory Guide 3.73

Please see file named “NF Storage Requirements.pdf”for details.

A-3. CONSOLIDATED NFST REFERENCE LIST

Please see file named "Consolidated List of References_NFST.pdf"for details.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
General	A	72.1	The regulations in this part establish requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel, power reactor-related Greater than Class C (GTCC) waste, and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI) and the terms and conditions under which the Commission will issue these licenses. The regulations in this part also establish requirements, procedures, and criteria for the issuance of licenses to the Department of Energy (DOE) to receive, transfer, package, and possess power reactor spent fuel, high-level radioactive waste, power reactor-related GTCC waste, and other radioactive materials associated with the storage of these materials in a monitored retrievable storage installation (MRS). The term Monitored Retrievable Storage Installation or MRS, as defined in § 72.3, is derived from the Nuclear Waste Policy Act (NWPA) and includes any installation that meets this definition. The regulations in this part also establish requirements, procedures, and criteria for the issuance of Certificates of Compliance approving spent fuel storage cask designs.	General Provisions - Propose
General	A	72.2 (a)	Except as provided in § 72.6(b), licenses issued under this part are limited to the receipt, transfer, packaging, and possession of: (1) Power reactor spent fuel to be stored in a complex that is designed and constructed specifically for storage of power reactor spent fuel aged for at least one year, other radioactive materials associated with spent fuel storage, and power reactor-related GTCC waste in a solid form in an independent spent fuel storage installation (ISFSI); or (2) Power reactor spent fuel to be stored in a monitored retrievable storage installation (MRS) owned by DOE that is designed and constructed specifically for the storage of spent fuel aged for at least one year, high-level radioactive waste that is in a solid form, other radioactive materials associated with storage of these materials, and power reactor-related GTCC waste that is in a solid form.	General Provisions - Scope.
General	A	72.2 (b)	The regulations in this part pertaining to an independent spent fuel storage installation (ISFSI) and a spent fuel storage cask apply to all persons in the United States, including persons in Agreement States. The regulations in this part pertaining to a monitored retrievable storage installation (MRS) apply only to DOE.	General Provisions - Scope.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
General	A	72.2 (c)	The requirements of this regulation are applicable, as appropriate, to both wet and dry modes of storage of-- (1) spent fuel in an independent spent fuel storage installation (ISFSI) and (2) spent fuel and solid high-level radioactive waste in a monitored retrievable storage installation (MRS).	General Provisions, Scope.
General	A	72.2 (d)	Licenses covering the storage of spent fuel in an existing spent fuel storage installation shall be issued in accordance with the requirements of this part as stated in § 72.40, as applicable.	General Provisions - Scope.
General	A	72.2 (e)	This part also gives notice to all persons who knowingly provide to any licensee, certificate holder, applicant for a license or certificate, contractor, or subcontractor, components, equipment, materials, or other goods or services, that relate to a licensee's, certificate holder's, or applicant's activities subject to this part, that they may be individually subject to NRC enforcement action for violation of § 72.12.	General Provisions - Scope.
General	A	72.2 (f)	Certificates of Compliance approving spent fuel storage cask designs shall be issued in accordance with the requirements of subpart L of this part.	General Provisions - Scope.
General	A	72.3	Provides the definitions as they are to be used in part 72.	General Provisions - Definitions.

Category	Sub part	Section	Description	Comments
General	A	72.4	Except where otherwise specified, all communications and reports concerning the regulations in this part and applications filed under them should be sent by mail addressed: ATTN: Document Control Desk, Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; by hand delivery to the NRC's offices at One White Flint North, 11555 Rockville Pike, Rockville, Maryland between 7:30 a.m. and 4:15 p.m. eastern time; or, where practicable, by electronic submission, for example, via Electronic Information Exchange, or CD-ROM. Electronic submissions must be made in a manner that enables the NRC to receive, read, authenticate, distribute, and archive the submission, and process and retrieve it a single page at a time. Detailed guidance on making electronic submissions can be obtained by visiting the NRC’s Web site at http://www.nrc.gov/site-help/e-submittals.html; by e-mail to MSHD.Resource@nrc.gov; or by writing the Office of Information Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. The guidance discusses, among other topics, the formats the NRC can accept, the use of electronic signatures, and the treatment of nonpublic information. If the submission deadline date falls on a Saturday, or Sunday, or a Federal holiday, the next Federal working day becomes the official due date.	General Provisions - Communications.
General	A	72.5	Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of the Commission, other than a written interpretation by the General Counsel, will be recognized to be binding upon the Commission.	General Provisions - Interpretations.
Licensing	A	72.6 (a)	Licenses for the receipt, handling, storage, and transfer of spent fuel or high-level radioactive waste are of two types: general and specific. Licenses for the receipt, handling, storage, and transfer of reactor-related GTCC are specific licenses. Any general license provided in this part is effective without the filing of an application with the Commission or the issuance of a licensing document to a particular person. A specific license is issued to a named person upon application filed pursuant to regulations in this part.	General Provisions - License required; types of licenses.

Category	Sub part	Section	Description	Comments
Licensing	A	72.6 (b)	A general license is hereby issued to receive title to and own spent fuel, high-level radioactive waste, or reactor-related GTCC waste without regard to quantity. Notwithstanding any other provision of this chapter, a general licensee under this paragraph is not authorized to acquire, deliver, receive, possess, use, or transfer spent fuel, high-level radioactive waste, or reactor-related GTCC waste except as authorized in a specific license.	General Provisions - License required; types of licenses.
Licensing	A	72.6 (c)	Except as authorized in a specific license and in a general license under subpart K of this part issued by the Commission in accordance with the regulations in this part, no person may acquire, receive, or possess-- (1) Spent fuel for the purpose of storage in an ISFSI; or (2) Spent fuel, high-level radioactive waste, or radioactive material associated with high-level radioactive waste for the purpose of storage in an MRS.	General Provisions - License required; types of licenses.
Licensing	A	72.7	The Commission may, upon application by any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.	General Provisions - Specific exemptions.
Licensing	A	72.8	Agreement States may not issue licenses covering the storage of spent fuel and reactor-related GTCC waste in an ISFSI or the storage of spent fuel, high-level radioactive waste, and reactor-related GTCC waste in an MRS.	General Provisions - Denial of licensing by Agreement States.
General	A	72.9 (a)	The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). OMB has approved the information collection requirements contained in this part under control number 3150–0132.	General Provisions - Information collection requirements: OMB approval.
General	A	72.9 (b)	The approved information collection requirements contained in this part appear in §§ 72.7, 72.11, 72.16, 72.22 through 72.34, 72.42, 72.44, 72.48 through 72.56, 72.62, 72.70, through 72.80, 72.90, 72.92, 72.94, 72.98, 72.100, 72.102, 72.103, 72.104, 72.108, 72.120, 72.126, 72.140 through 72.176, 72.180 through 72.186, 72.192, 72.206, 72.212, 72.218, 72.230, 72.232, 72.234, 72.236, 72.240, 72.242, 72.244, 72.248.	General Provisions - Information collection requirements: OMB approval.
General	A	72.9 (c)	In § 72.79, Form N–71 and associated forms are approved under control number 3150–0056, and DOC/NRC Forms AP –1, AP–A, and associated forms are approved under control number 0694–0135.	General Provisions - Information collection requirements: OMB approval.

Category	Sub part	Section	Description	Comments
General	A	72.10 (a)	Discrimination by a Commission licensee, certificate holder, an applicant for a Commission license or a CoC, or a contractor or subcontractor of any of these, against an employee for engaging in certain protected activities, is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, or privileges of employment. The protected activities are established in section 211 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act or the Energy Reorganization Act are outlined in § 72.10 (a)(1, 2, and 3)	General Provisions - Employee protection.
General	A	72.10 (a)(1)	(1) The protected activities include but are not limited to: (i) Providing the Commission or his or her employer information about alleged violations of either of the statutes named in paragraph (a) introductory text of this section or possible violations of requirements imposed under either of those statutes; (ii) Refusing to engage in any practice made unlawful under either of the statutes named in paragraph (a) introductory text or under these requirements if the employee has identified the alleged illegality to the employer; (iii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements; (iv) Testifying in any Commission proceeding, or before Congress, or at any Federal or State proceeding regarding any provision (or proposed provision) of either of the statutes named in paragraph (a) introductory text; and (v) Assisting or participating in, or is about to assist or participate in, these activities.	General Provisions - Employee protection.
General	A	72.10 (a)(2)	These activities are protected even if no formal proceeding is actually initiated as a result of the employee assistance or participation.	General Provisions - Employee protection.
General	A	72.10 (a)(3)	This section has no application to any employee alleging discrimination prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.	General Provisions - Employee protection.

Category	Sub part	Section	Description	Comments
General	A	72.10 (b)	Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 180 days after an alleged violation occurs. The employee may do this by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.	General Provisions - Employee protection.
General	A	72.10 (c)	A violation of paragraph (a), (e), or (f) of this section by a Commission licensee, certificate holder, applicant for a Commission license or a CoC, or a contractor or subcontractor of any of these may be grounds for: (1) Denial, revocation, or suspension of the license or the CoC. (2) Imposition of a civil penalty on the licensee, applicant, or a contractor or subcontractor of the licensee or applicant. (3) Other enforcement action.	General Provisions - Employee protection.
General	A	72.10 (d)	Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.	General Provisions - Employee protection.
General, Operation	A	72.10 (e)	(1) Each licensee, certificate holder, and applicant for a license or CoC must prominently post the revision of NRC Form 3, "Notice to Employees," referenced in 10 CFR 19.11(c). This form must be posted at locations sufficient to permit employees protected by this section to observe a copy on the way to or from their place of work. The premises must be posted not later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license or CoC, and for 30 days following license or CoC termination. (2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in appendix D to part 20 of this chapter, via email to Forms.Resource@nrc.gov, or by visiting the NRC's online library at http://www.nrc.gov/reading-rm/doc-collections/forms/.	General Provisions - Employee protection.

Category	Sub part	Section	Description	Comments
General	A	72.10 (f)	No agreement affecting the compensation, terms, conditions, or privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor pursuant to section 211 of the Energy Reorganization Act of 1974, as amended, may contain any provision which would prohibit, restrict, or otherwise discourage an employee from participating in protected activity as defined in paragraph (a)(1) of this section including, but not limited to, providing information to the NRC or to his or her employer on potential violations or other matters within NRC's regulatory responsibilities.	General Provisions - Employee protection.
General, Operation	A	72.11 (a)	Information provided to the Commission by a licensee, certificate holder, or an applicant for a license or CoC; or information required by statute or by the Commission's regulations, orders, license or CoC conditions, to be maintained by the licensee or certificate holder, must be complete and accurate in all material respects.	General Provisions - Completeness and accuracy of information. Requirment to maintain and retain documentation.
General, Operation	A	72.11 (b)	Each licensee, certificate holder, or applicant for a license or CoC must notify the Commission of information identified by the licensee, certificate holder, or applicant for a license or CoC as having, for the regulated activity, a significant implication for public health and safety or common defense and security. A licensee, certificate holder, or an applicant for a license or CoC violates this paragraph only if the licensee, certificate holder, or applicant for a license or CoC fails to notify the Commission of information that the licensee, certificate holder, or applicant for a license or CoC has identified as having a significant implication for public health and safety or common defense and security. Notification must be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.	General Provisions -Completeness and accuracy of information.

Category	Sub part	Section	Description	Comments
General, Operation	A	72.12 (a)	Any licensee, certificate holder, applicant for a license or certificate, employee of a licensee, certificate holder, or applicant for a license or certificate; or any contractor (including a supplier or consultant) or subcontractor, employee of a contractor or subcontractor of any licensee, certificate holder, or applicant for a license or certificate who knowingly provides to any licensee, certificate holder, applicant for a license or certificate, contractor, or subcontractor, any components, materials, or other goods or services that relate to a licensee's, certificate holder's, or applicant's activities subject to this part, may not: (1) Engage in deliberate misconduct that causes or would have caused, if not detected, a licensee, certificate holder or applicant to be in violation of any rule, regulation, or order; or any term, condition, or limitation of any license or certificate issued by the Commission; or (2) Deliberately submit to the NRC, a licensee, a certificate holder, an applicant for a license or certificate, or a licensee's, applicant's, or certificate holder's contractor or subcontractor, information that the person submitting the information knows to be incomplete or inaccurate in some respect material to the NRC.	General Provisions - Deliberate misconduct.
General	A	72.12 (b)	A person who violates paragraph (a)(1) or (a)(2) of this section may be subject to enforcement action in accordance with the procedures in 10 CFR part 2, subpart B.	General Provisions - Deliberate misconduct.
General	A	72.12 (c)	For the purposes of paragraph (a)(1) of this section, deliberate misconduct by a person means an intentional act or omission that the person knows: 1) Would cause a licensee, certificate holder or applicant for a license or certificate to be in violation of any rule, regulation, or order; or any term, condition, or limitation, of any license or certificate issued by the Commission; or (2) Constitutes a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee, certificate holder, applicant, contractor, or subcontractor.	General Provisions - Deliberate misconduct.
Licensing, Operation	A	72.13 (a)	This section identifies those sections, under this part, that apply to the activities associated with a specific license, a general license, or a certificate of compliance.	General Provisions - Applicability.

Category	Sub part	Section	Description	Comments
Licensing, Operation	A	72.13 (b)	The following sections apply to activities associated with a specific license: Secs. 72.1; 72.2(a) through (e); 72.3 through 72.13(b); 72.16 through 72.34; 72.40 through 72.62; 72.70 through 72.86; 72.90 through 72.108; 72.120 through 72.130; 72.140 through 72.176; 72.180 through 72.186; 72.190 through 72.194; and 72.200 through 72.206.	General Provisions - Applicability.
Licensing, Operation	A	72.13 (c)	The following sections apply to activities associated with a general license: 72.1; 72.2(a)(1), (b), (c), and (e); 72.3 through 72.6(c)(1); 72.7 through 72.13(a) and (c); 72.30(b), (c), (d), (e) and (f); 72.32(c) and (d); 72.44(b) and (f); 72.48; 72.50(a); 72.52(a), (b), (d), and (e); 72.60; 72.62; 72.72 through 72.80(f); 72.82 through 72.86; 72.104; 72.106; 72.122; 72.124; 72.126; 72.140 through 72.176; 72.190; 72.194; 72.210 through 72.220, and 72.240(a).	General Provisions - Applicability.
Licensing, Operation	A	72.13 (d)	The following sections apply to activities associated with a certificate of compliance: Secs. 72.1; 72.2(e) and (f); 72.3; 72.4; 72.5; 72.7; 72.9 through 72.13(a) and (d); 72.48; 72.84(a); 72.86; 72.124; 72.140 through 72.176; 72.214; and 72.230 through 72.248.	General Provisions - Applicability.
Licensing	B	72.16 (a)	*Place of filing* . Each application for a license, or amendment thereof, under this part should be filed with the Director of the NRC's Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards in accordance with § 72.4, *Communications* .	License Application, Form, and Contents - Filing of application for specific license.
Licensing	B	72.16 (b)	*Oath or affirmation* . Each application for a license or license amendment (including amendments to such applications), except for those filed by DOE, must be executed in an original signed by the applicant or duly authorized officer thereof under oath or affirmation. Each application for a license or license amendment (including amendments to such applications) filed by DOE must be signed by the Secretary of Energy or the Secretary's authorized representative.	License Application, Form, and Contents - Filing of application for specific license.
Licensing	B	72.16 (c)	*Copies of application on paper or CD-ROM.* If the application is on paper, it must be the signed original. The applicant shall maintain the capability to generate additional copies for distribution in accordance with instruction from the Director or the Director's designee.	License Application, Form, and Contents - Filing of application for specific license.
Licensing	B	72.16 (d)	*Fees. The application, amendment, and renewal fees applicable to a license covering an ISFSI are those shown in § 170.31 of this chapter.*	License Application, Form, and Contents - Filing of application for specific license.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	B	72.16 (e)	*Notice of docketing.* Upon receipt of an application for a license or license amendment under this part, the Director, Office of Nuclear Material Safety and Safeguards or the Director's designee will assign a docket number to the application, notify the applicant of the docket number, instruct the applicant to distribute copies retained by the applicant in accordance with paragraph (c) of this section, and cause a notice of docketing to be published in the Federal Register. The notice of docketing shall identify the site of the ISFSI or the MRS by locality and State and may include a notice of hearing or a notice of proposed action and opportunity for hearing as provided by § 72.46 of this part. In the case of an application for a license or an amendment to a license for an MRS, the Director, Office of Nuclear Material Safety and Safeguards, or the Director's designee, in accordance with § 72.200 of this part, shall send a copy of the notice of docketing to the Governor and legislature of any State in which an MRS is or may be located, to the Chief Executive of the local municipality, to the Governors of any contiguous States and to the governing body of any affected Indian tribe.	License Application, Form, and Contents - Filing of application for specific license.
Licensing	B	72.18	In any application under this part, the applicant may incorporate by reference information contained in previous applications, statements, or reports filed with the Commission: Provided, That such references are clear and specific.	License Application, Form, and Contents - Elimination of repetition.
Licensing	B	72.20	Applications and documents submitted to the Commission in connection with applications may be made available for public inspection in accordance with provisions of the regulations contained in parts 2 and 9 of this chapter.	License Application, Form, and Contents - Public inspection of application.
Licensing	B	72.22 (a), (b), (c)	Each application must state: (a) Full name of applicant; (b) Address of applicant; (c) Description of business or occupation of applicant;	License Application, Form, and Contents - Contents of application: General and financial information.

Category	Sub part	Section	Description	Comments
Licensing	B	72.22 (d)	Dependent on type of applicant, application must state: (1) An individual: Citizenship and age; (2) A partnership: Name, citizenship, and address of each partner and the principal location at which the partnership does business; (3) A corporation or an unincorporated association: (i) The State in which it is incorporated or organized and the principal location at which it does business; and (ii) The names, addresses, and citizenship of its directors and principal officers; (4) Acting as an agent or representative of another person in filing the application: The identification of the principal and the information required under this paragraph with respect to such principal. (5) The Department of Energy: (i) The identification of the DOE organization responsible for the construction and operation of the ISFSI or MRS, including a description of any delegations of authority and assignments of responsibilities. (ii) For each application for a license for an MRS, the provisions of the public law authorizing the construction and operation of the MRS.	License Application, Form, and Contents - Contents of application: General and financial information.
Licensing	B	72.22, (e)	Except for DOE, information sufficient to demonstrate to the Commission the financial qualifications of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the license is sought. The information must state the place at which the activity is to be performed, the general plan for carrying out the activity, and the period of time for which the license is requested. The information must show that the applicant either possesses the necessary funds, or that the applicant has reasonable assurance of obtaining the necessary; funds or that by a combination of the two, the applicant will have the necessary funds available to cover the following: (1) Estimated construction costs; (2) Estimated operating costs over the planned life of the ISFSI; and (3) Estimated decommissioning costs, and the necessary financial arrangements to provide reasonable assurance before licensing, that decommissioning will be carried out after the removal of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste from storage.	License Application, Form, and Contents - Contents of application: General and financial information.

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Category	Sub part	Section	Description	Comments
Licensing	B	72.22, (f)	Each applicant for a license under this part to receive, transfer, and possess power reactor spent fuel, power reactor-related Greater than Class C (GTCC) waste, and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI) shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements in § 73.21 and the requirements of § 73.22 or § 73.23, as applicable.	License Application, Form, and Contents - Contents of application: General and financial information.
Licensing	B	72.24	Each application for a license under this part must include a Safety Analysis Report describing the proposed ISFSI or MRS for the receipt, handling, packaging, and storage of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste as appropriate, including how the ISFSI or MRS will be operated.	License Application, Form, and Contents - Contents of application: General and financial information.
Licensing	B	72.24 (a)	The Safety Analysis Report must contain a description and safety assessment of the site on which the ISFSI or MRS is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI or MRS that bear on the suitability of the site when the ISFSI or MRS is operated at its design capacity. If the proposed ISFSI or MRS is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI or MRS and such other facility--including shared common utilities and services--must be evaluated.	License Application, Form, and Contents - Contents of application: General and financial information.
Licensing	B	72.24 (b)	The Safety Analysis Report must contain a description and discussion of the ISFSI or MRS structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.	License Application, Form, and Contents - Contents of application: General and financial information.

Category	Sub part	Section	Description	Comments
Licensing	B	72.24 (c)	The Safety Analysis Report must contain the design of the ISFSI or MRS in sufficient detail to support the findings in § 72.40 for the term requested in the application, including: (1) The design criteria for the ISFSI or MRS pursuant to subpart F of this part, with identification and justification for any additions to or departures from the general design criteria; (2) the design bases and the relation of the design bases to the design criteria; (3) Information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI or MRS will satisfy the design bases with an adequate margin for safety; and (4) Applicable codes and standards.	License Application, Form, and Contents - Contents of application: General and financial information.
Licensing	B	72.24 (d)	The Safety Analysis Report must contain An analysis and evaluation of the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI or MRS and including determination of: (1) The margins of safety during normal operations and expected operational occurrences during the life of the ISFSI or MRS; and (2) The adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents, including natural and manmade phenomena and events.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (e)	The Safety Analysis Report must contain the means for controlling and limiting occupational radiation exposures within the limits given in 10CFR20, and for meeting the objective of maintaining exposures as low as is reasonably achievable.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (f)	The Safety Analysis Report must contain the features of ISFSI or MRS design and operating modes to reduce to the extent practicable radioactive waste volumes generated at the installation.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (g)	The Safety Analysis Report must contain an identification and justification for the selection of those subjects that will be probable license conditions and technical specifications. These subjects must cover the design, construction, preoperational testing, operation, and decommissioning of the ISFSI or MRS.	License Application, Form, and Contents - Contents of application: Technical information.

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Category	Sub part	Section	Description	Comments
Licensing	B	72.24 (h)	The Safety Analysis Report must contain a plan for the conduct of operations, including the planned managerial and administrative controls system, and the applicant's organization, and program for training of personnel pursuant to subpart I.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (i)	If the proposed ISFSI or MRS incorporates structures, systems, or components important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, then the Safety Analysis Report must provide an indentification of these structures, systems, or components along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste as appropriate for storage at the ISFSI or MRS.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (j)	The Safety Analysis Report must contain the technical qualifications of the applicant to engage in the proposed activities, as required by § 72.28.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (k)	The Safety Analysis Report must contain a description of the applicant's plans for coping with emergencies, as required by § 72.32.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (l)	The Safety Analysis Report must contain a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences. The description must identify the design objectives and the means to be used for keeping levels of radioactive material in effluents to the environment as low as is reasonably achievable and within the exposure limits stated in § 72.104. The description must include: (1) An estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal ISFSI or MRS operations; (2) A description of the equipment and processes used in radioactive waste systems; and (3) A general description of the provisions for packaging, storage, and disposal of solid wastes containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.	License Application, Form, and Contents - Contents of application: Technical information.

Category	Sub part	Section	Description	Comments
Licensing	B	72.24 (m)	The Safety Analysis Report must contain an analysis of the potential dose equivalent or committed dose equivalent to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environment or direct radiation from the ISFSI or MRS. The calculations of individual dose equivalent or committed dose equivalent must be performed for direct exposure, inhalation, and ingestion occurring as a result of the postulated design basis event.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (n)	The Safety Analysis Report must contain a description of the quality assurance program that satisfies the requirements of subpart G to be applied to the design, fabrication, construction, testing, operation, modification, and decommissioning of the structures, systems, and components of the ISFSI or MRS important to safety. The description must identify the structures, systems, and components important to safety. The program must also apply to managerial and administrative controls used to ensure safe operation of the ISFSI or MRS.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.24 (o)	The Safety Analysis Report must contain a description of the detailed security measures for physical protection, including design features and the plans required by subpart H. For an application from DOE for an ISFSI or MRS, DOE will provide a description of the physical protection plan for protection against radiological sabotage as required by subpart H.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing, Operation	B	72.24 (p)	The Safety Analysis Report must contain a description of the program covering preoperational testing and initial operations.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing, Decommis sioning	B	72.24 (q)	The Safety Analysis Report must contain a description of the decommissioning plan required under § 72.30.	License Application, Form, and Contents - Contents of application: Technical information.
Licensing	B	72.26	Each application under this part shall include proposed technical specifications in accordance with the requirements of § 72.44 and a summary statement of the bases and justifications for these technical specifications.	License Application, Form, and Contents - Contents of application: Technical specifications.

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Category	Sub part	Section	Description	Comments
Licensing	B	72.28 (a), (b), (c), (d)	Each application under this part must include: (a) The technical qualifications, including training and experience, of the applicant to engage in the proposed activities; (b) A description of the personnel training program required under subpart I; (c) A description of the applicant's operating organization, delegations of responsibility and authority and the minimum skills and experience qualifications relevant to the various levels of responsibility and authority; and (d) A commitment by the applicant to have and maintain an adequate complement of trained and certified installation personnel prior to the receipt of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste as appropriate for storage.	License Application, Form, and Contents - Contents of application: Applicant's technical qualifications.
Licensing, Decommissioning	B	72.30 (a)	Each application under this part must include a proposed decommissioning plan that contains sufficient information on proposed practices and procedures for the decontamination of the site and facilities and for disposal of residual radioactive materials after all spent fuel, high-level radioactive waste, and reactor-related GTCC waste have been removed, in order to provide reasonable assurance that the decontamination and decommissioning of the ISFSI or MRS at the end of its useful life will provide adequate protection to the health and safety of the public. This plan must identify and discuss those design features of the ISFSI or MRS that facilitate its decontamination and decommissioning at the end of its useful life.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.

Category	Sub part	Section	Description	Comments
Licensing, Decommissioning	B	72.30 (b)	Each holder of, or applicant for, a license under this part must submit for NRC review and approval a decommissioning funding plan that must contain: (1) Information on how reasonable assurance will be provided that funds will be available to decommission the ISFSI or MRS. (2) A detailed cost estimate for decommissioning, in an amount reflecting: (i) The cost of an independent contractor to perform all decommissioning activities; (ii) An adequate contingency factor; and (iii) The cost of meeting the § 20.1402 of this chapter criteria for unrestricted use, provided that, if the applicant or licensee can demonstrate its ability to meet the provisions of § 20.1403 of this chapter, the cost estimate may be based on meeting the § 20.1403 criteria. (3) Identification of and justification for using the key assumptions contained in the DCE. (4) A description of the method of assuring funds for decommissioning from paragraph (e) of this section, including means for adjusting cost estimates and associated funding levels periodically over the life of the facility. (5) The volume of onsite subsurface material containing residual radioactivity that will require remediation to meet the criteria for license termination. (6) A certification that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (c)	At the time of license renewal and at intervals not to exceed 3 years, the decommissioning funding plan must be resubmitted with adjustments as necessary to account for changes in costs and the extent of contamination. If the amount of financial assurance will be adjusted downward, this can not be done until the updated decommissioning funding plan is approved. The decommissioning funding plan must update the information submitted with the original or prior approved plan and must specifically consider the effect of the following events on decommissioning costs: (1) Spills of radioactive material producing additional residual radioactivity in onsite subsurface material. (2) Facility modifications. (3) Changes in authorized possession limits. (4) Actual remediation costs that exceed the previous cost estimate.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.

Category	Sub part	Section	Description	Comments
Licensing, Decommissioning	B	72.30 (d)	If, in surveys made under 10 CFR 20.1501(a), residual radioactivity in soils or groundwater is detected at levels that would require such radioactivity to be reduced to a level permitting release of the property for unrestricted use under the decommissioning requirements in part 20 of this chapter, the licensee must submit a new or revised decommissioning funding plan within one year of when the survey is completed.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)	The financial instrument must include the licensee's name, license number, and docket number; and the name, address, and other contact information of the issuer, and, if a trust is used, the trustee. When any of the foregoing information changes, the licensee must, within 30 days, submit financial instruments reflecting such changes. Financial assurance for decommissioning must be provided by one or more of the following methods: (1) Prepayment, (2) A surety method, insurance, or other guarantee method, (3) An external sinking fund, (4) A statement of intent (only applicable to Federal, State, or local government licensees), (5) Applicable methods outlined in 10 CFR 50.75(b), (e), and (h) (only applicable to licensees who are issued a power reactor license under part 50 of this chapter or ISFSI licensees who are an electric utility, as defined in part 50 of this chapter, with a specific license issued under this part), and/or (6) An acceptable arrangement (only applicable to a governmental entity who is assuming custody and ownership of a site)	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(1)	Prepayment. Prepayment is the deposit before the start of operation into an account segregated from licensee assets and outside the licensee's administrative control of cash or liquid assets such that the amount of funds would be sufficient to pay decommissioning costs. Prepayment must be made into a trust account, and the trustee and the trust must be acceptable to the Commission.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.

Category	Sub part	Section	Description	Comments
Licensing, Decommissioning	B	72.30 (e)(2)	A surety method, insurance, or other guarantee method. These methods guarantee that decommissioning costs will be paid. A surety method may be in the form of a surety bond, or letter of credit. A parent company guarantee of funds for decommissioning costs based on a financial test may be used if the guarantee and test are as contained in Appendix A to part 30 of this chapter. For commercial corporations that issue bonds, a guarantee of funds by the applicant or licensee for decommissioning costs based on a financial test may be used if the guarantee and test are as contained in Appendix C to part 30 of this chapter. For commercial companies that do not issue bonds, a guarantee of funds by the applicant or licensee for decommissioning costs may be used if the guarantee and test are as contained in Appendix D to part 30 of this chapter. Except for an external sinking fund, a parent company guarantee or a guarantee by the applicant or licensee may not be used in combination with other financial methods to satisfy the requirements of this section. A guarantee by the applicant or licensee may not be used in any situation where the applicant or licensee has a parent company holding majority control of the voting stock of the company. Any surety method or insurance used to provide financial assurance for decommissioning must contain the following conditions provided in § 72.30 (e)(2)(i, ii, and iii).	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(2) (i)	The surety method or insurance must be open-ended or, if written for a specified term, such as five years, must be renewed automatically unless 90 days or more prior to the renewal date, the issuer notifies the Commission, the beneficiary, and the licensee of its intention not to renew. The surety method or insurance must also provide that the full face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if the licensee fails to provide a replacement acceptable to the Commission withing 30 days after receipt of notification or cancellation.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(2) (ii)	The surety method or insurance must be payable to a trust established for decommissioning costs. The trustee and trust must be acceptable to the Commission. An acceptable trustee includes an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(2) (iii)	The surety or insurance must remain in effect until the Commission has terminated the license.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.

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Category	Sub part	Section	Description	Comments
Licensing, Decommissioning	B	72.30 (e)(3)	An external sinking fund in which deposits are made at least annually, coupled with a surety method, insurance, or other guarantee method, the value of which may decrease by the amount being accumulated in the sinking fund. An external sinking fund is a fund established and maintained by setting aside funds periodically in an account segregated from licensee assets and outside the licensee's administrative control in which the total amount of funds would be sufficient to pay decommissioning costs at the time termination of operation is expected. An external sinking fund must be in the form of a trust. If the other guarantee method is used, no surety or insurance may be combined with the external sinking fund. The surety, insurance, or other guarantee provisions must be as stated in paragraph (e)(2) of this section.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(4)	In the case of Federal, State, or local government licensees, a statement of intent containing a cost estimate for decommissioning, and indicating that funds for decommissioning will be obtained when necessary.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(5)	In the case of licensees who are issued a power reactor license under part 50 of this chapter or ISFSI licensees who are an electric utility, as defined in part 50 of this chapter, with a specific license issued under this part, the methods of 10 CFR 50.75(b), (e), and (h), as applicable. In the event that funds remaining to be placed into the licensee's ISFSI decommissioning external sinking fund are no longer approved for recovery in rates by a competent rate making authority, the licensee must make changes to provide financial assurance using one or more of the methods stated in paragraphs (1) through (4) of this section.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Decommissioning	B	72.30 (e)(6)	When a governmental entity is assuming custody and ownership of a site, an arrangement that is deemed acceptable by such governmental entity.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Operation, Decommissioning	B	72.30 (f)	Each person licensed under this part shall keep records of information important to the decommissioning of a facility in an identified location until the site is released for unrestricted use. If records important to the decommissioning of a facility are kept for other purposes, reference to these records and their locations may be used. Information the Commission considers important to decommissioning consists of items outlined in provided in § 72.30 (f)(1, 2, 3, and 4)	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.

Category	Sub part	Section	Description	Comments
Licensing, Operation, Decommissioning	B	72.30 (f)(1)	Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records must include any known information on identification of involved nuclides, quantities, forms, and concentrations.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Operation, Decommissioning	B	72.30 (f)(2)	As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored, and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. If required drawings are referenced, each relevant document need not be indexed individually. If drawings are not available, the licensee shall substitute appropriate records of available information concerning these areas and locations.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Operation, Decommissioning	B	72.30 (f)(3)	A list contained in a single document and updated no less than every 2 years of the following: (i) All areas designated and formerly designated as restricted areas as defined under 10 CFR 20.1003; and (ii) All areas outside of restricted areas that require documentation under § 72.30(f)(1).	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing, Operation, Decommissioning	B	72.30 (f)(4)	Records of the cost estimate performed for the decommissioning funding plan and records of the funding method used for assuring funds are available for decommissioning.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.

Category	Sub part	Section	Description	Comments
Licensing, Operation, Decommissioning	B	72.30 (g)	In providing financial assurance under this section, each licensee must use the financial assurance funds only for decommissioning activities and each licensee must monitor the balance of funds held to account for market variations. The licensee must replenish the funds, and report such actions to the NRC, as follows: (1) If, at the end of a calendar year, the fund balance is below the amount necessary to cover the cost of decommissioning, but is not below 75 percent of the cost, the licensee must increase the balance to cover the cost, and must do so within 30 days after the end of the calendar year. (2) If, at any time, the fund balance falls below 75 percent of the amount necessary to cover the cost of decommissioning, the licensee must increase the balance to cover the cost, and must do so within 30 days of the occurrence. (3) Within 30 days of taking the actions required by paragraph (g)(1) or (g)(2) of this section, the licensee must provide a written report of such actions to the Director, Office of Nuclear Material Safety and Safeguards, and state the new balance of the fund.	License Application, Form, and Contents - Financial assurance and recordkeeping for decommissioning.
Licensing	B	72.32 (a)	Each application for an ISFSI that is licensed under this part which is: Not located on the site of a nuclear power reactor, or not located within the exclusion area as defined in 10 CFR part 100 of a nuclear power reactor, or located on the site of a nuclear power reactor which does not have an operating license, or located on the site of a nuclear power reactor that is not authorized to operate must be accompanied by an Emergency Plan that includes the items describe in § 72.32 (a)(1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, and 16)	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(1)	*Facility description* . A brief description of the licensee's facility and area near the site.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(2)	*Types of accidents* . An identification of each type of radioactive materials accident.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(3)	*Classification of accidents* . A classification system for classifying accidents as "alerts."	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(4)	*Detection of accidents* . Identification of the means of detecting an accident condition.	License Application, Form, and Contents - Emergency Plan

Category	Sub part	Section	Description	Comments
Licensing	B	72.32 (a)(5)	*Mitigation of consequences* . A brief description of the means of mitigating the consequences of each type of accident, including those provided to protect workers onsite, and a description of the program for maintaining the equipment.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(6)	*Assessment of releases* . A brief description of the methods and equipment to assess releases of radioactive materials.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(7)	*Responsibilities* . A brief description of the responsibilities of licensee personnel should an accident occur, including identification of personnel responsible for promptly notifying offsite response organizations and the NRC; also responsibilities for developing, maintaining, and updating the plan.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(8)	*Notification and coordination* . A commitment to and a brief description of the means to promptly notify offsite response organizations and request offsite assistance, including medical assistance for the treatment of contaminated injured onsite workers when appropriate. A control point must be established. The notification and coordination must be planned so that unavailability of some personnel, parts of the facility, and some equipment will not prevent the notification and coordination. The licensee shall also commit to notify the NRC operations center immediately after notifications of the appropriate offsite response organizations and not later than one hour after the licensee declares an emergency. [Note: These reporting requirements do not supersede or release licensees of complying with the requirements under the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Pub. L. 99-499 or other State or Federal reporting requirements.]	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(9)	*Information to be communicated* . A brief description of the types of information on facility status; radioactive releases; and recommended protective actions, if necessary, to be given to offsite response organizations and to the NRC.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(10)	*Training* . A brief description of the training the licensee will provide workers on how to respond to an emergency and any special instructions and orientation tours the licensee would offer to fire, police, medical and other emergency personnel.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(11)	*Safe condition* . A brief description of the means of restoring the facility to a safe condition after an accident.	License Application, Form, and Contents - Emergency Plan

Category	Sub part	Section	Description	Comments
Licensing	B	72.32 (a)(12)	*Exercises* . (i) Provisions for conducting semiannual communications checks with offsite response organizations and biennial onsite exercises to test response to simulated emergencies. Radiological/Health Physics, Medical, and Fire drills shall be conducted annually. Semiannual communications checks with offsite response organizations must include the check and update of all necessary telephone numbers. The licensee shall invite offsite response organizations to participate in the biennial exercise. (ii) Participation of offsite response organizations in biennial exercises, although recommended, is not required. Exercises must use scenarios not known to most exercise participants. The licensee shall critique each exercise using individuals not having direct implementation responsibility for conducting the exercise. Critiques of exercises must evaluate the appropriateness of the plan, emergency procedures, facilities, equipment, training of personnel, and overall effectiveness of the response. Deficiencies found by the critiques must be corrected.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(13)	*Hazardous chemicals* . A certification that the applicant has met its responsibilities under the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Pub. L. 99-499, with respect to hazardous materials at the facility.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(14)	*Comments on Plan* . The licensee shall allow the offsite response organizations expected to respond in case of an accident 60 days to comment on the initial submittal of the licensee's emergency plan before submitting it to NRC. Subsequent plan changes need not have the offsite comment period unless the plan changes affect the offsite response organizations. The licensee shall provide any comments received within the 60 days to the NRC with the emergency plan.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(15)	*Offsite assistance* . The applicant's emergency plans shall include a brief description of the arrangements made for requesting and effectively using offsite assistance on site and provisions that exist for using other organizations capable of augmenting the planned onsite response.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (a)(16)	Arrangements made for providing information to the public.	License Application, Form, and Contents - Emergency Plan

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Category	Sub part	Section	Description	Comments
Licensing	B	72.32 (b)	Each application for an MRS that is licensed under this part and each application for an ISFSI that is licensed under this part and that may process and/or repackage spent fuel, must be accompanied by an Emergency Plan that includes the items describe in § 72.32 (b)(1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, and 16)	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(1)	*Facility description* . A brief description of the licensee facility and area near the site.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(2)	*Types of accidents* . An identification of each type of radioactive materials accident.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(3)	*Classification of accidents* . A classification system for classifying accidents as "alerts" or "site area emergencies."	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(4)	*Detection of accidents* . Identification of the means of detecting an accident condition.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(5)	*Mitigation of consequences* . A brief description of the means of mitigating the consequences of each type of accident, including those provided to protect workers on site, and a description of the program for maintaining the equipment.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(6)	*Assessment of releases* . A brief description of the methods and equipment to assess releases of radioactive materials.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(7)	*Responsibilities* . A brief description of the responsibilities of licensee personnel should an accident occur, including identification of personnel responsible for promptly notifying offsite response organizations and the NRC; also responsibilities for developing, maintaining, and updating the plan.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(8)	*Notification and coordination* . A commitment to and a brief description of the means to promptly notify offsite response organizations and request offsite assistance, including medical assistance for the treatment of contaminated injured onsite workers when appropriate. A control point must be established. The notification and coordination must be planned so that unavailability of some personnel, parts of the facility, and some equipment will not prevent the notification and coordination. The licensee shall also commit to notify the NRC operations center immediately after notifications of the appropriate offsite response organizations and not later than one hour after the licensee declares an emergency. [Note: These reporting requirements do not supersede or release licensees of complying with the requirements under the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Pub. L. 99-499 or other State or Federal reporting requirements.]	License Application, Form, and Contents - Emergency Plan

Category	Sub part	Section	Description	Comments
Licensing	B	72.32 (b)(9)	*Information to be communicated* . A brief description of the types of information on facility status; radioactive releases; and recommended protective actions, if necessary, to be given to offsite response organizations and to the NRC.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(10)	*Training* . A brief description of the training the licensee will provide workers on how to respond to an emergency and any special instructions and orientation tours the licensee would offer to fire, police, medical and other emergency personnel.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(11)	*Safe condition* . A brief description of the means of restoring the facility to a safe condition after an accident.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(12)	*Exercises* . (i) Provisions for conducting quarterly communications checks with offsite response organizations and biennial onsite exercises to test response to simulated emergencies. Radiological/Health Physics, Medical, and Fire Drills shall be held semiannually. Quarterly communications checks with offsite response organizations must include the check and update of all necessary telephone numbers. The licensee shall invite offsite response organizations to participate in the biennial exercises. (ii) Participation of offsite response organizations in the biennial exercises, although recommended, is not required. Exercises must use scenarios not known to most exercise participants. The licensee shall critique each exercise using individuals not having direct implementation responsibility for conducting the exercise. Critiques of exercises must evaluate the appropriateness of the plan, emergency procedures, facilities, equipment, training of personnel, and overall effectiveness of the response. Deficiencies found by the critiques must be corrected.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(13)	*Hazardous chemicals* . A certification that the applicant has met its responsibilities under the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Pub. L. 99-499, with respect to hazardous materials at the facility.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(14)	*Comments on Plan* . The licensee shall allow the offsite response organizations expected to respond in case of an accident 60 days to comment on the initial submittal of the licensee's emergency plan before submitting it to NRC. Subsequent plan changes need not have the offsite comment period unless the plan changes affect the offsite response organizations. The licensee shall provide any comments received within the 60 days to the NRC with the emergency plan.	License Application, Form, and Contents - Emergency Plan

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	B	72.32 (b)(15)	Offsite assistance. The applicant's emergency plans shall include the following: (i) A brief description of the arrangements made for requesting and effectively using offsite assistance on site and provisions that exist for using other organizations capable of augmenting the planned onsite response. (ii) Provisions that exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite. (iii) Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained. (iv) Adequate methods, systems, and equipment for assessing and monitoring actual or potential consequences of a radiological emergency condition are available. (v) Arrangements are made for medical services for contaminated and injured onsite individuals. (vi) Radiological Emergency Response Training has been made available to those offsite who may be called to assist in an emergency onsite.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (b)(16)	Arrangements made for providing information to the public.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (c)	For an ISFSI that is: (1) located on the site, or (2) located within the exclusion area as defined in 10 CFR part 100, of a nuclear power reactor licensed for operation by the Commission, the emergency plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.32 (d)	A licensee with a license issued under this part may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.	License Application, Form, and Contents - Emergency Plan
Licensing	B	72.34 (a)	Each application for an ISFSI or MRS license under this part must be accompanied by an Environmental Report which meets the requirements of subpart A of part 51 of this chapter.	License Application, Form, and Contents - Environmental Report

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	C	72.40 (a)	Except as provided in paragraph (c) of this section, the Commission will issue a license under this part upon a determination that the application for a license meets the standards and requirements of the Act and the regulations of the Commission, and upon finding that items describe in § 72.40 (a)(1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, and 14) are met.	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(1)	The applicant's proposed ISFSI or MRS design complies with subpart F;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(2)	The proposed site complies with the criteria in subpart E;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(3)	If on the site of a nuclear power plant or other licensed activity or facility, the proposed ISFSI would not pose an undue risk to the safe operation of such nuclear power plant or other licensed activity or facility;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(4)	The applicant is qualified by reason of training and experience to conduct the operation covered by the regulations in this part;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(5)	The applicant's proposed operating procedures to protect health and to minimize danger to life or property are adequate;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(6)	Except for DOE, the applicant for an ISFSI or MRS is financially qualified to engage in the proposed activities in accordance with the regulations in this part;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(7)	The applicant's quality assurance plan complies with subpart G;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(8)	The applicant's physical protection provisions comply with subpart H. DOE has complied with the safeguards and physical security provisions identified in § 72.24(o);	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(9)	The applicant's personnel training program complies with subpart I;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(10)	Except for DOE, the applicant's decommissioning plan and its financing pursuant to § 72.30 provide reasonable assurance that the decontamination and decommissioning of the ISFSI or MRS at the end of its useful life will provide adequate protection to the health and safety of the public;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(11)	The applicant's emergency plan complies with § 72.32;	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(12)	The applicable provisions of part 170 of this chapter have been satisfied;	Issuance and Conditions of License - Issuance of license.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	C	72.40 (a)(13)	There is reasonable assurance that: (i) The activities authorized by the license can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of this chapter; and	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (a)(14)	The issuance of the license will not be inimical to the common defense and security.	Issuance and Conditions of License - Issuance of license.
Licensing	C	72.40 (b)	A license to store spent fuel and reactor-related GTCC waste in the proposed ISFSI or to store spent fuel, high-level radioactive waste, and reactor-related GTCC waste in the proposed MRS may be denied if construction on the proposed facility begins before a finding approving issuance of the proposed license with any appropriate conditions to protect environmental values. Grounds for denial may be the commencement of construction prior to a finding by the Director, Office of Nuclear Materials Safety and Safeguards or designee or a finding after a public hearing by the presiding officer, Atomic Safety and Licensing Board, or the Commission acting as a collegial body, as appropriate, that the action called for is the issuance of the proposed license with any appropriate conditions to protect environmental values. This finding is to be made on the basis of information filed and evaluations made pursuant to subpart A of part 51 of this chapter or in the case of an MRS on the basis of evaluations made pursuant to sections 141(c) and (d) or 148(a) and (c) of NWPA (96 Stat. 2242, 2243, 42 U.S.C. 10161(c), (d); 101 Stat. 1330-235, 1330-236, 42 U.S.C. 10168(a), (c)), as appropriate, and after weighing the environmental, economic, technical and other benefits against environmental costs and considering available alternatives.	Issuance and Conditions of License -Issuance of license. An approved license is required prior to start of construction.
Licensing	C	72.40 (c)	For facilities that have been covered under previous licensing actions including the issuance of a construction permit under part 50 of this chapter, a reevaluation of the site is not required except where new information is discovered which could alter the original site evaluation findings. In this case, the site evaluation factors involved will be reevaluated.	Issuance and Conditions of License -Issuance of license.

Category	Sub part	Section	Description	Comments
Licensing	C	72.42 (a)	Each license issued under this part must be for a fixed period of time to be specified in the license. The license term for an ISFSI must not exceed 40 years from the date of issuance. The license term for an MRS must not exceed 40 years from the date of issuance. Licenses for either type of installation may be renewed by the Commission at the expiration of the license term upon application by the licensee for a period not to exceed 40 years and under the requirements of this rule. Application for ISFSI license renewals must include the following: (1) TLAAs that demonstrate that structures, systems, and components important to safety will continue to perform their intended function for the requested period of extended operation; and (2) A description of the AMP for management of issues associated with aging that could adversely affect structures, systems, and components important to safety.	Issuance and Conditions of License - Duration of license; renewal
Licensing	C	72.42 (b)	Applications for renewal of a license should be filed in accordance with the applicable provisions of subpart B of this part at least 2 years before the expiration of the existing license. The application must also include design bases information as documented in the most recently updated FSAR as required by § 72.70. Information contained in previous applications, statements, or reports filed with the Commission under the license may be incorporated by reference provided that these references are clear and specific.	Issuance and Conditions of License - Duration of license; renewal
Licensing	C	72.42 (c)	In any case in which a licensee, not less than two years prior to expiration of its existing license, has filed an application in proper form for renewal of a license, the existing license shall not expire until a final decision concerning the application for renewal has been made by the Commission.	Issuance and Conditions of License - Duration of license; renewal
Licensing	C	72.44 (a)	Each license issued under this part shall include license conditions. The license conditions may be derived from the analyses and evaluations included in the Safety Analysis Report and amendments thereto submitted pursuant to § 72.24. License conditions pertain to design, construction and operation. The Commission may also include additional license conditions as it finds appropriate.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (b)	Each license issued under this part shall be subject to the following conditions outlined in § 72.44 (b) (1, 2, 3, 4, 5, and 6), even if they are not explicitly stated therein;	Issuance and Conditions of License -Licensing conditions.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	C	72.44 (b)(1)	Neither the license nor any right thereunder shall be transferred, assigned, or disposed of in any manner, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall, after securing full information, find that the transfer is in accordance with the provisions of the Atomic Energy Act of 1954, as amended, and give its consent in writing.	Issuance and Conditions of License -Licensing conditions. See NUREG 1567, § 1.4 for acceptance criteria.
Licensing	C	72.44 (b)(2)	The license shall be subject to revocation, suspension, modification, or amendment in accordance with the procedures provided by the Atomic Energy Act of 1954, as amended, and Commission regulations.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (b)(3)	Upon request of the Commission, the licensee shall, at any time before expiration of the license, submit written statements, signed under oath or affirmation if appropriate, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (b)(4)	The licensee shall have an NRC-approved program in effect that covers the training and certification of personnel that meets the requirements of subpart I before the licensee may receive spent fuel and/or reactor-related GTCC waste for storage at an ISFSI or the receipt of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste for storage at an MRS.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (b)(5)	The license shall permit the operation of the equipment and controls that are important to safety of the ISFSI or the MRS only by personnel whom the licensee has certified as being adequately trained to perform such operations, or by uncertified personnel who are under the direct visual supervision of a certified individual.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (b)(6)	(i) Each licensee shall notify the appropriate NRC Regional Administrator, in writing, immediately following the filing of a voluntary or involuntary petition for bankruptcy under any Chapter of Title II (Bankruptcy) of the United States Code by or against: (A) The licensee; (B) An entity (as that term is defined in 11 U.S.C. 101(14)) controlling the licensee or listing the license or licensee as property of the estate; or C) An affiliate (as that term is defined in 11 U.S.C. 101(2)) of the licensee. (ii) This notification must indicate: (A) The bankruptcy court in which the petition for bankruptcy was filed; and (B) The date of the filing of the petition.	Issuance and Conditions of License -Licensing conditions.

Category	Sub part	Section	Description	Comments
Licensing	C	72.44 (c)	Each license issued under this part must include technical specifications. Technical specifications must include requirements in the categories described in § 72.42 (c) (1, 2, 3, 4, and 5).	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (c)(1)	*Functional and operating limits and monitoring instruments and limiting control settings* . (i) Functional and operating limits for an ISFSI or MRS are limits on fuel or waste handling and storage conditions that are found to be necessary to protect the integrity of the stored fuel or waste container, to protect employees against occupational exposures and to guard against the uncontrolled release of radioactive materials; and (ii) Monitoring instruments and limiting control settings for an ISFSI or MRS are those related to fuel or waste handling and storage conditions having significant safety functions.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (c)(2)	*Limiting conditions* . Limiting conditions are the lowest functional capability or performance levels of equipment required for safe operation.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (c)(3)	*Surveillance requirements* . Surveillance requirements include: (i) Inspection and monitoring of spent fuel, high-level radioactive waste, or reactor-related GTCC waste in storage; (ii) inspection, test and calibration activities to ensure that the necessary integrity of required systems and components is maintained; (iii) confirmation that operation of the ISFSI or MRS is within the required functional and operating limits; and (iv) confirmation that the limiting conditions required for safe storage are met.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (c)(4)	*Design features* . Design features include items that would have a significant effect on safety if altered or modified, such as materials of construction and geometric arrangements.	Issuance and Conditions of License -Licensing conditions.
Licensing	C	72.44 (c)(5)	*Administrative controls* . Administrative controls include the organization and management procedures, recordkeeping, review and audit, and reporting requirements necessary to assure that the operations involved in the storage of spent fuel and reactor-related GTCC waste in an ISFSI and the storage of spent fuel, high-level radioactive waste, and reactor-related GTCC waste in an MRS are performed in a safe manner.	Issuance and Conditions of License -Licensing conditions.

Category	Sub part	Section	Description	Comments
Licensing	C	72.44 (d)	Each license authorizing the receipt, handling, and storage of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste under this part must include technical specifications that, in addition to stating the limits on the release of radioactive materials for compliance with limits of part 20 of this chapter and the "as low as is reasonably achievable" objectives for effluents, require that: (1) Operating procedures for control of effluents be established and followed, and equipment in the radioactive waste treatment systems be maintained and used, to meet the requirements of § 72.104; (2) An environmental monitoring program be established to ensure compliance with the technical specifications for effluents; and (3) An annual report be submitted to the Commission in accordance with Sec. 72.4, specifying the quantity of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the previous 12 months of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent releases. On the basis of this report and any additional information that the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate. The report must be submitted within 60 days after the end of the 12-month monitoring period.	Issuance and Conditions of License -Licensing conditions.
Licensing, Operation	C	72.44 (e)	The licensee shall make no change that would decrease the effectiveness of the physical security plan prepared pursuant to § 72.180 without the prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to the license pursuant to § 72.56. A licensee may make changes to the physical security plan without prior Commission approval, provided that such changes do not decrease the effectiveness of the plan. The licensee shall furnish to the Commission a report containing a description of each change within two months after the change is made, and shall maintain records of changes to the plan made without prior Commission approval for a period of 3 years from the date of the change.	Issuance and Conditions of License -Licensing conditions.

Category	Sub part	Section	Description	Comments
Licensing, Operation	C	72.44 (f)	A licensee shall follow and maintain in effect an emergency plan that is approved by the Commission. The licensee may make changes to the approved plan without Commission approval only if such changes do not decrease the effectiveness of the plan. Within six months after any change is made, the licensee shall submit, in accordance with § 72.4, a report containing a description of any changes made in the plan addressed to Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, with a copy to the appropriate NRC Regional Office shown in appendix D to part 20 of this chapter. Proposed changes that decrease the effectiveness of the approved emergency plan must not be implemented unless the licensee has received prior approval of such changes from the Commission.	Issuance and Conditions of License -Licensing conditions.
Licensing, Construc-tion, Operation	C	72.44 (g)	A license issued to DOE under this part for an MRS authorized by section 142(b) of NWPA (101 Stat. 1330-232, 42 U.S.C. 10162(b)) must include the following conditions: (1) Construction of the MRS may not begin until the Commission has authorized the construction of a repository under section 114(d) of NWPA (96 Stat. 2215, as amended by 101 Stat. 1330-230, 42 U.S.C. 10134 (d)) and part 60 or 63 of this chapter; (2) Construction of the MRS or acceptance of spent nuclear fuel, high-level radioactive waste, and/or reactor-related GTCC waste at the MRS is prohibited during such time as the repository license is revoked by the Commission or construction of the repository ceases. (3) The quantity of spent nuclear fuel or high-level radioactive waste at the site of the MRS at any one time may not exceed 10,000 metric tons of heavy metal until a repository authorized under NWPA and part 60 or 63 of this chapter first accepts spent nuclear fuel or solidified high-level radioactive waste; and (4) The quantity of spent nuclear fuel or high-level radioactive waste at the site of the MRS at any one time may not exceed 15,000 metric tons of heavy metal.	Issuance and Conditions of License -Licensing conditions.
Licensing, Operation	C	72.44 (h)	Each licensee shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 and the requirements of § 73.22 or § 73.23, as applicable.	Issuance and Conditions of License -Licensing conditions.

Category	Sub part	Section	Description	Comments
Licensing	C	72.46 (a)	In connection with each application for a license under this part, the Commission shall issue or cause to be issued a notice of proposed action and opportunity for hearing in accordance with § 2.105 or § 2.1107 of this chapter, as appropriate, or, if the Commission finds that a hearing is required in the public interest, a notice of hearing in accordance with § 2.104 of this chapter.	Issuance and Conditions of License -Public hearings.
Licensing	C	72.46 (b)	(1) In connection with each application for an amendment to a license under this part, the Commission shall, except as provided in paragraph (b)(2) of this section, issue or cause to be issued a notice of proposed action and opportunity for hearing in accordance with § 2.105 or § 2.1107 of this chapter, as appropriate, or, if the Commission finds that a hearing is required in the public interest, a notice of hearing in accordance with § 2.104 of this chapter. (2) The Director, Office of Nuclear Material Safety and Safeguards, or the Director's designee may dispense with a notice of proposed action and opportunity for hearing or a notice of hearing and take immediate action on an amendment to a license issued under this part upon a determination that the amendment does not present a genuine issue as to whether the health and safety of the public will be significantly affected. After taking the action, the Director or the Director's designee shall promptly publish a notice in the Federal Register of the action taken and of the right of interested persons to request a hearing on whether the action should be rescinded or modified. If the action taken amends an MRS license, the Director or the Director's designee shall also inform the appropriate State and local officials.	Issuance and Conditions of License -Public hearings.
Licensing	C	72.46 (c)	The notice of proposed action and opportunity for hearing or the notice of hearing may be included in the notice of docketing required to be published by § 72.16 of this part.	Issuance and Conditions of License -Public hearings.
Licensing	C	72.46 (d)	If no request for a hearing or petition for leave to intervene is filed within the time prescribed in the notice of proposed action and opportunity for hearing, the Director, Office of Nuclear Material Safety and Safeguards or the Director's designee may take the proposed action, and thereafter shall promptly inform the appropriate State and local officials and publish a notice in the Federal Register of the action taken. In accordance with § 2.764(c) of this chapter, the Director, Office of Nuclear Material Safety and Safeguards shall not issue an initial license for the construction and operation of an ISFSI located at a site other than a reactor site or an MRS until expressly authorized to do so by the Commission.	Issuance and Conditions of License -Public hearings.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	C	72.46 (d)	If an application for (or an amendment to) a specific license issued under this part incorporates by reference information on the design of a spent fuel storage cask for which NRC approval pursuant to subpart L of this part has been issued or is being sought, the scope of any public hearing held to consider the application will not include any cask design issues.	Issuance and Conditions of License -Public hearings.
Operation	C	72.48 (b)	This section applies to: (1) Each holder of a general or specific license issued under this part, and (2) Each holder of a Certificate of Compliance (CoC) issued under this part.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (c)(1)	A licensee or certificate holder may make changes in the facility or spent fuel storage cask design as described in the FSAR (as updated), make changes in the procedures as described in the FSAR (as updated), and conduct tests or experiments not described in the FSAR (as updated), without obtaining either: (i) A license amendment pursuant to § 72.56 (for specific licensees) or (ii) A CoC amendment submitted by the certificate holder pursuant to § 72.244 (for general licensees and certificate holders) if: (A) A change to the technical specifications incorporated in the specific license is not required; or (B) A change in the terms, conditions, or specifications incorporated in the CoC is not required; and (C) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.	Issuance and Conditions of License -Changes, tests, and experiments.

Category	Sub part	Section	Description	Comments
Operation	C	72.48 (c)(2)	A specific licensee shall obtain a license amendment pursuant to § 72.56, a certificate holder shall obtain a CoC amendment pursuant to § 72.244, and a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to § 72.244, prior to implementing a proposed change, test, or experiment if the change, test, or experiment would: (i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR (as updated); (ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a system, structure, or component (SSC) important to safety previously evaluated in the FSAR (as updated); (iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR (as updated); (iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR (as updated); (v) Create a possibility for an accident of a different type than any previously evaluated in the FSAR (as updated); (vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR (as updated); (vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or (viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (c)(3)	(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed pursuant to this section and analyses performed pursuant to § 72.56 or § 72.244 since the last update of the FSAR pursuant to § 72.70, or § 72.248 of this part.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (c)(4)	The provisions in this section do not apply to changes to the facility or procedures when the applicable regulations establish more specific criteria for accomplishing such changes.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (d)(1)	The licensee and certificate holder shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license or CoC amendment pursuant to paragraph (c)(2) of this section.	Issuance and Conditions of License -Changes, tests, and experiments.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Operation	C	72.48 (d)(2)	The licensee and certificate holder shall submit, as specified in § 72.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each. A report shall be submitted at intervals not to exceed 24 months.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (d)(3)	The records of changes in the facility or spent fuel storage cask design shall be maintained until: (i) Spent fuel is no longer stored in the facility or the spent fuel storage cask design is no longer being used, or (ii) The Commission terminates the license or CoC issued pursuant to this part.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (d)(4)	The records of changes in procedures and of tests and experiments shall be maintained for a period of 5 years.	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (d)(5)	The holder of a spent fuel storage cask design CoC, who permanently ceases operation, shall provide the records of changes to the new certificate holder or to the Commission, as appropriate, in accordance § 72.234(d)(3).	Issuance and Conditions of License -Changes, tests, and experiments.
Operation	C	72.48 (d)(6)	(i) A general licensee shall provide a copy of the record for any changes to a spent fuel storage cask design to the applicable certificate holder within 60 days of implementing the change. (ii) A specific licensee using a spent fuel storage cask design, approved pursuant to subpart L of this part, shall provide a copy of the record for any changes to a spent fuel storage cask design to the applicable certificate holder within 60 days of implementing the change. (iii) A certificate holder shall provide a copy of the record for any changes to a spent fuel storage cask design to any general or specific licensee using the cask design within 60 days of implementing the change.	Issuance and Conditions of License -Changes, tests, and experiments.
Licensing	C	72.50 (a)	No license or any part included in a license issued under this part for an ISFSI or MRS shall be transferred, assigned, or in any manner disposed of, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission gives its consent in writing.	Issuance and Conditions of License -Transfer of license.

Category	Sub part	Section	Description	Comments
Licensing	C	72.50 (b)	(1) An application for transfer of a license must include as much of the information described in §§ 72.22 and 72.28 with respect to the identity and the technical and financial qualifications of the proposed transferee as would be required by those sections if the application were for an initial license. The application must also include a statement of the purposes for which the transfer of the license is requested and the nature of the transaction necessitating or making desirable the transfer of the license. (2) The Commission may require any person who submits an application for the transfer of a license pursuant to the provisions of this section to file a written consent from the existing licensee, or a certified copy of an order or judgment of a court of competent jurisdiction, attesting to the person's right—subject to the licensing requirements of the Act and these regulations—to possession of the radioactive materials and the storage installation involved. (3) The application shall describe the financial assurance that will be provided for the decommissioning of the facility under § 72.30.	Issuance and Conditions of License -Transfer of license.
Licensing	C	72.50 (c)	After appropriate notice to interested persons, including the existing licensee, and observance of such procedures as may be required by the Act or regulations or orders of the Commission, the Commission will approve an application for the transfer of a license, if the Commission determines that: (1) The proposed transferee is qualified to be the holder of the license; and (2) Transfer of the license is consistent with applicable provisions of the law, and the regulations and orders issued by the Commission.	Issuance and Conditions of License - Transfer of license.
Licensing	C	72.52 (a)	This section does not apply to an ISFSI or MRS constructed and operated by DOE.	Issuance and Conditions of License - Creditor regulations.

Category	Sub part	Section	Description	Comments
Licensing, Operation	C	72.52 (b)	Pursuant to section 184 of the Act, the Commission consents, without individual application, to the creation of any mortgage, pledge, or other lien on special nuclear material contained in spent fuel not owned by the United States that is the subject of a license or on any interest in special nuclear material in spent fuel; Provided: (1) That the rights of any creditor so secured may be exercised only in compliance with and subject to the same requirements and restrictions as would apply to the licensee pursuant to the provisions of the license, the Atomic Energy Act of 1954, as amended, and regulations issued by the Commission pursuant to said Act; and (2) That no creditor so secured may take possession of the spent fuel and/or reactor-related GTCC waste under the provisions of this section before-- (i) The Commission issues a license authorizing possession; or (ii) The license is transferred.	Issuance and Conditions of License - Creditor regulations.
Licensing	C	72.52 (c)	Any creditor so secured may apply for transfer of the license covering spent fuel and/or reactor-related GTCC waste by filing an application for transfer of the license under § 72.50(b). The Commission will act upon the application under § 72.50(c).	Issuance and Conditions of License - Creditor regulations.
Licensing	C	72.52 (d)	Nothing contained in this regulation shall be deemed to affect the means of acquiring, or the priority of, any tax lien or other lien provided by law.	Issuance and Conditions of License - Creditor regulations.
Licensing	C	72.52 (e)	As used in this section, "creditor" includes, without implied limitation-- 1) The trustee under any mortgage, pledge, or lien on spent fuel and/or reactor-related GTCC waste in storage made to secure any creditor; (2) Any trustee or receiver of spent fuel and/or reactor-related GTCC waste appointed by a court of competent jurisdiction in any action brought for the benefit of any creditor secured by a mortgage, pledge, or lien; (3) Any purchaser of the spent fuel and/or reactor-related GTCC waste at the sale thereof upon foreclosure of the mortgage, pledge, or lien or upon exercise of any power of sale contained therein; or 4) Any assignee of any such purchaser.	Issuance and Conditions of License - Creditor regulations.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	C	72.54 (a)	Each specific license expires at the end of the day on the expiration date stated in the license except when a licensee has filed an application for renewal pursuant to § 72.42 not less than 24 months before the expiration of the existing license. If an application for renewal has been filed at least 24 months prior to the expiration date stated in the existing license, the existing license expires at the end of the day on which the Commission makes a final determination to deny the renewal application or, if the determination states an expiration date, the expiration date stated in the determination.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Licensing	C	72.54 (b)	Each specific license revoked by the Commission expires at the end of the day on the date of the Commission's final determination to revoke the license or on the expiration date stated in the determination or as otherwise provided by Commission Order.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Licensing, Operation	C	72.54 (c)	Each specific license continues in effect, beyond the expiration date if necessary, with respect to possession of licensed material until the Commission notifies the licensee in writing that the license is terminated. During this time, the licensee shall-- (1) Limit actions involving spent fuel, reactor-related GTCC waste, or other licensed material to those related to decommissioning; and (2) Continue to control entry to restricted areas until they are suitable for release in accordance with NRC requirements.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Decommissioning	C	72.54 (d)	As required by § 72.42(b), or within 60 days of the occurrence of any of the following, consistent with the administrative directions in § 72.4, each licensee shall notify the NRC in writing, and submit within 12 months of this notification, a final decommissioning plan and begin decommissioning upon approval of the plan if-- (1) The licensee has decided to permanently cease principal activities, as defined in this part, at the entire site or any separate building or outdoor area that contains residual radioactivity such that the building or outdoor area is unsuitable for release in accordance with NRC requirements; or (2) No principal activities under the license have been conducted for a period of 24 months; or (3) No principal activities have been conducted for a period of 24 months in any separate building or outdoor area that contains residual radioactivity such that the building or outdoor area is unsuitable for release in accordance with NRC requirements.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.

Category	Sub part	Section	Description	Comments
Decommissioning	C	72.54 (e)	Coincident with the notification required by paragraph (d) of this section, the licensee shall maintain in effect all decommissioning financial assurances established by the licensee pursuant to § 72.30 in conjunction with a license issuance or renewal or as required by this section. The amount of the financial assurance must be increased, or may be decreased, as appropriate, to cover the detailed cost estimate for decommissioning established pursuant to paragraph (g)(5) of this section. (1) Any licensee who has not provided financial assurance to cover the detailed cost estimate submitted with the decommissioning plan shall do so when this rule becomes effective November 24, 1995. (2) Following approval of the decommissioning plan, a licensee may reduce the amount of the financial assurance as decommissioning proceeds and radiological contamination is reduced at the site with the approval of the Commission.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Decommissioning	C	72.54 (f)	(1) The Commission may grant a request to delay or postpone initiation of the decommissioning process if the Commission determines that this relief is not detrimental to the public health and safety and is otherwise in the public interest. The request must be submitted no later than 30 days before notification pursuant to paragraph (d) of this section. The schedule for decommissioning set forth in paragraph (d) of this section may not commence until the Commission has made a determination on the request. (2) The Commission may approve an alternate schedule for submittal of the final decommissioning plan required pursuant to paragraph (d) of this section if the Commission determines that the alternate schedule is necessary to the effective conduct of decommissioning operations and presents no undue risk from radiation to the public health and safety, and is otherwise to the public interest.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Decommissioning	C	72.54 (g)	The proposed final decommissioning plan must include-- (1) A description of the current conditions of the site or separate building or outdoor area sufficient to evaluate the acceptability of the plan; (2) The choice of the alternative for decommissioning with a description of the activities involved; (3) A description of controls and limits on procedures and equipment to protect occupational and public health and safety; (4) A description of the planned final radiation survey; and (5) An updated detailed cost estimate for the chosen alternative for decommissioning, comparison of that estimate with present funds set aside for decommissioning, and plan for assuring the availability of adequate funds for completion of decommissioning including means for adjusting cost estimates and associated funding levels over any storage or surveillance period; and (6) A description of technical specifications and quality assurance provisions in place during decommissioning.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Decommissioning	C	72.54 (h)	For final decommissioning plans in which the major dismantlement activities are delayed by first placing the ISFSI or MRS in storage, planning for these delayed activities may be less detailed. Updated detailed plans must be submitted and approved prior to the start of these activities.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Decommissioning	C	72.54 (i)	If the final decommissioning plan demonstrates that the decommissioning will be completed as soon as practicable, performed in accordance with the regulations in this chapter, and will not be inimical to the common defense and security or to the health and safety of the public, and after notice to interested persons, the Commission will approve the plan subject to any appropriate conditions and limitations and issue an order authorizing decommissioning.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Decommissioning	C	72.54 (j)	(1) Except as provided in paragraph (k) of this section, each licensee shall complete decommissioning of the site or separate building or outdoor area as soon as practicable but no later than 24 months following approval of the final decommissioning plan by the Commission. (2) Except as provided in paragraph (k) of this section, when decommissioning involves the entire site, each licensee shall request license termination as soon as practicable but no later than 24 months following approval of the final decommissioning plan by the Commission.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.

Category	Sub part	Section	Description	Comments
Decommissioning	C	72.54 (k)	The Commission may approve a request for an alternate schedule for completion of decommissioning of the site or separate building or outdoor area, and license termination if appropriate, if the Commission determines that the alternate schedule is warranted by consideration of the following: (1) Whether it is technically feasible to complete decommissioning within the allotted 24-month period; (2) Whether sufficient waste disposal capacity is available to allow completion of decommissioning within the allotted 24-month period; (3) Whether a significant volume reduction in wastes requiring disposal will be achieved by allowing short-lived radionuclides to decay; (4) Whether a significant reduction in radiation exposure to workers can be achieved by allowing short-lived radionuclides to decay; and (5) Other site-specific factors that the Commission may consider appropriate on a case-by-case basis, such as regulatory requirements of other government agencies, lawsuits, ground-water treatment activities, monitored natural ground-water restoration, actions that could result in more environmental harm than deferred cleanup, and other factors beyond the control of the licensee.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Decommissioning	C	72.54 (l)	As the final step in decommissioning, the licensee shall-- (1) Certify the disposition of all licensed material, including accumulated wastes, by submitting a completed NRC Form 314 or equivalent information; and (2) Conduct a radiation survey of the premises where the licensed activities were conducted and submit a report of the results of this survey, unless the licensee demonstrates in some other manner that the premises are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E. The licensee shall, as appropriate-- (i) Report levels of gamma radiation in units of millisieverts (microroentgen) per hour at one meter from surfaces, and report levels of radioactivity, including alpha and beta, in units of megabecquerels (disintegrations per minute or microcuries) per 100 square centimeters removable and fixed for surfaces, megabecquerels (microcuries) per milliliter for water, and becquerels (picocuries) per gram for solids such as soils or concrete; and (ii) Specify the survey instrument(s) used and certify that each instrument is properly calibrated and tested.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.

Category	Sub part	Section	Description	Comments
Licensing, Decommissioning	C	72.54 (m)	Specific licenses, including expired licenses, will be terminated by written notice to the licensee when the Commission determines that-- (1) The decommissioning has been performed in accordance with the approved final decommissioning plan and the order authorizing decommissioning; and (2)(i) A radiation survey has been performed which demonstrates that the premises are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E; or (ii) Other information submitted by the licensee is sufficient to demonstrate that the premises are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E. (3) Records required by § 72.80(e) have been received.	Issuance and Conditions of License - Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.
Licensing	C	72.56	Whenever a holder of a specific license desires to amend the license (including a change to the license conditions), an application for an amendment shall be filed with the Commission fully describing the changes desired and the reasons for such changes, and following as far as applicable the form prescribed for original applications.	Issuance and Conditions of License - Application for amendment of license.
Licensing	C	72.58	In determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of initial licenses.	Issuance and Conditions of License - Issuance of amendment.
Licensing	C	72.60 (a)	The terms and conditions of all licenses are subject to amendment, revision, or modification by reason of amendments to the Atomic Energy Act of 1954, as amended, or by reason or rules, regulations, or orders issued in accordance with the Act or any amendments thereto.	Issuance and Conditions of License - Modification, revocation, and suspension of license.
Licensing	C	72.60 (b)	Any license may be modified, revoked, or suspended in whole or in part for any of the following: (1) Any material false statement in the application or in any statement of fact required under section 182 of the Act; (2) Conditions revealed by the application or statement of fact or any report, record, inspection or other means which would warrant the Commission to refuse to grant a license on an original application; (3) Failure to operate an ISFSI or MRS in accordance with the terms of the license; (4) Violation of, or failure to observe, any of the terms and conditions of the Act, or of any applicable regulation, license, or order of the Commission.	Issuance and Conditions of License - Modification, revocation, and suspension of license.

Category	Sub part	Section	Description	Comments
Licensing	C	72.60 (c)	Upon revocation of a license, the Commission may immediately cause the retaking of possession of all special nuclear material contained in spent fuel and/or reactor-related GTCC waste held by the licensee. In cases found by the Commission to be of extreme importance to the national defense and security or to the health and safety of the public, the Commission may cause the taking of possession of any special nuclear material contained in spent fuel and/or reactor-related GTCC waste held by the licensee before following any of the procedures provided under sections 551-558 of title 5 of the United States Code.	Issuance and Conditions of License - Modification, revocation, and suspension of license.
Licensing, Operation	C	72.62 (a)	As used in this section, backfitting means the addition, elimination, or modification, after the license has been issued, of: (1) Structures, systems, or components of an ISFSI or MRS, or (2) Procedures or organization required to operate an ISFSI or MRS.	Issuance and Conditions of License - Backfitting.
Licensing, Operation	C	72.62 (b)	The Commission will require backfitting of an ISFSI or MRS if it finds that such action is necessary to assure adequate protection to occupational or public health and safety, or to bring the ISFSI or MRS into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by a licensee.	Issuance and Conditions of License - Backfitting.
Licensing, Operation	C	72.62 (c)	The Commission may require the backfitting of an ISFSI or MRS if it finds: 1) That there is a substantial increase in the overall protection of the occupational or public health and safety to be derived from the backfit, and (2) That the direct and indirect costs of implementation for that ISFSI or MRS are justified in view of this increased protection.	Issuance and Conditions of License - Backfitting.
Licensing, Operation	C	72.62 (d)	The Commission may at any time require a holder of a license to submit such information concerning the backfitting or the proposed backfitting of an ISFSI or MRS as it deems appropriate.	Issuance and Conditions of License - Backfitting.
Licensing, Operation	D	72.70 (a)	Each specific licensee for an ISFSI or MRS shall update periodically, as provided in paragraphs (b) and (c) of this section, the final safety analysis report (FSAR) to assure that the information included in the report contains the latest information developed. (1) Each licensee shall submit an original FSAR to the Commission, in accordance with § 72.4, within 90 days after issuance of the license. (2) The original FSAR shall be based on the safety analysis report submitted with the application and reflect any changes and applicant commitments developed during the license approval and/or hearing process.	Records, Reports, Inspections, and Enforcement - Safety analysis report updating.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing, Operation	D	72.70 (b)	Each update shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last update to the FSAR under this section. The update shall include the effects of: [Note: Effects of changes includes appropriate revisions of descriptions in the FSAR such that the FSAR (as updated) is complete and accurate.] (1) All changes made in the ISFSI or MRS or procedures as described in the FSAR; (2) All safety analyses and evaluations performed by the licensee either in support of approved license amendments, or in support of conclusions that changes did not require a license amendment in accordance with § 72.48; (3) All final analyses and evaluations of the design and performance of structures, systems, and components that are important to safety taking into account any pertinent information developed during final design, construction, and preoperational testing; and (4) All analyses of new safety issues performed by or on behalf of the licensee at Commission request. The information shall be appropriately located within the updated FSAR.	Records, Reports, Inspections, and Enforcement - Safety analysis report updating.

Category	Sub part	Section	Description	Comments
Licensing, Operation	D	72.70 (c)	(1) The update of the FSAR must be filed in accordance with § 72.4. If the update is filed on paper, it should be filed on a page-replacement basis; if filed electronically, it should be filed on a full replacement basis. See Guidance for Electronic Submissions to the Commission at http://www.nrc.gov/site-help/e-submittals.html. (2) A paper update filed on a page-replacement basis must include a list that identifies the current pages of the FSAR following page replacement. If the update is filed electronically on a full replacement basis, it must include a list of changed pages. (3) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both); (4) The update shall include: (i) A certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, or that no such changes were made; and (ii) An identification of changes made under the provisions of § 72.48, but not previously submitted to the Commission; (5) The update shall reflect all changes implemented up to a maximum of 6 months prior to the date of filing; and (6) Updates shall be filed every 24 months from the date of issuance of the license.	Records, Reports, Inspections, and Enforcement - Safety analysis report updating.
Licensing, Operation	D	72.70 (d)	The updated FSAR shall be retained by the licensee until the Commission terminates the license.	Records, Reports, Inspections, and Enforcement - Safety analysis report updating.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Operation	D	72.72 (a)	Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all special nuclear material with quantities as specified in § 74.13(a) of this chapter and for source material as specified in § 40.64 of this chapter. The records must include as a minimum the name of shipper of the material to the ISFSI or MRS, the estimated quantity of radioactive material per item (including special nuclear material in spent fuel and reactor-related GTCC waste), item identification and seal number, storage location, onsite movements of each fuel assembly or storage canister, and ultimate disposal. These records for spent fuel and reactor-related GTCC waste at an ISFSI or for spent fuel, high-level radioactive waste, and reactorrelated GTCC waste at an MRS must be retained for as long as the material is stored and for a period of 5 years after the material is disposed of or transferred out of the ISFSI or MRS.	Records, Reports, Inspections, and Enforcement - Material balance, inventory, and records requirements for stored materials.
Operation	D	72.72 (b)	Each licensee shall conduct a physical inventory of all spent fuel, high-level radioactive waste, and reactor-related GTCC waste containing special nuclear material meeting the requirements in paragraph (a) of this section at intervals not to exceed 12 months unless otherwise directed by the Commission. The licensee shall retain a copy of the current inventory as a record until the Commission terminates the license.	Records, Reports, Inspections, and Enforcement - Material balance, inventory, and records requirements for stored materials.
Operation	D	72.72 (c)	Each licensee shall establish, maintain, and follow written material control and accounting procedures that are sufficient to enable the licensee to account for material in storage. The licensee shall retain a copy of the current material control and accounting procedures until the Commission terminates the license.	Records, Reports, Inspections, and Enforcement - Material balance, inventory, and records requirements for stored materials.
Operation	D	72.72 (d)	Records of spent fuel, high-level radioactive waste, and reactor-related GTCC waste containing special nuclear material meeting the requirements in paragraph (a) of this section must be kept in duplicate. The duplicate set of records must be kept at a separate location sufficiently remote from the original records that a single event would not destroy both sets of records. Records of spent fuel or reactor-related GTCC waste containing special nuclear material transferred out of an ISFSI or of spent fuel, high-level radioactive waste, or reactor-related GTCC waste containing special nuclear material transferred out of an MRS must be preserved for a period of five years after the date of transfer.	Records, Reports, Inspections, and Enforcement - Material balance, inventory, and records requirements for stored materials.

Category	Sub part	Section	Description	Comments
Operation	D	72.74 (a)	Each licensee shall notify the NRC Operations Center within one hour of discovery of accidental criticality or any loss of special nuclear material. [Note: Commercial telephone number of the NRC Operations Center is (301) 816-5100.]	Records, Reports, Inspections, and Enforcement - Reports of accidental criticality or loss of special nuclear material.
Operation	D	72.74 (b)	This notification must be made to the NRC Operations Center via the Emergency Notification System if the licensee is party to that system. If the Emergency Notification System is inoperative or unavailable, the licensee shall make the required notification via commercial telephonic service or any other dedicated telephonic system or any other method that will ensure that a report is received by the NRC Operations Center within one hour. The exemption of § 73.21(g)(3) of this chapter applies to all telephonic reports required by this section.	Records, Reports, Inspections, and Enforcement - Reports of accidental criticality or loss of special nuclear material.
Operation	D	72.74 (c)	Reports required under § 73.71 of this chapter need not be duplicated under the requirements of this section.	Records, Reports, Inspections, and Enforcement - Reports of accidental criticality or loss of special nuclear material.
Operation	D	72.75 (a)	*Emergency notifications* : Each licensee shall notify the NRC Headquarters Operations Center upon the declaration of an emergency as specified in the licensee's approved emergency plan addressed in § 72.32. The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies, but not later than one hour after the time the licensee declares an emergency.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (b)	*Non-emergency notifications:* Four-hour reports. Each licensee shall notify the NRC as soon as possible but not later than four hours after the discovery of any of the following events or conditions involving spent fuel, HLW, or reactor-related GTCC waste: (1) An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under this part when the action is immediately needed to protect the public health and safety, and no action consistent with license or certificate of compliance conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent. (2) Any event or situation related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other Government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.

Category	Sub part	Section	Description	Comments
Operation	D	72.75 (c)	*Non-emergency notifications:* Eight-hour reports. Each licensee shall notify the NRC as soon as possible but not later than eight hours after the discovery of any of the following events or conditions involving spent fuel, HLW, or reactor-related GTCC waste: (1) A defect in any spent fuel, HLW, or reactor-related GTCC waste storage structure, system, or component that is important to safety. (2) A significant reduction in the effectiveness of any spent fuel, HLW, or reactor-related GTCC waste storage confinement system during use. (3) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (d)	Non-emergency notifications: 24-hour reports. Each licensee shall notify the NRC within 24 hours after the discovery of any of the following events involving spent fuel, HLW, or reactor-related GTCC waste: (1) An event in which important to safety equipment is disabled or fails to function as designed when: (i) The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident; and (ii) No redundant equipment was available and operable to perform the required safety function. (2) For notifications made under this paragraph, the licensee may delay the notification to the NRC if the end of the 24-hour period occurs outside of the NRC's normal working day (i.e., 7:30 a.m. to 5:00 p.m. Eastern time), on a weekend, or a Federal holiday. In these cases, the licensee shall notify the NRC before 8:00 a.m. Eastern time on the next working day.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.

Category	Sub part	Section	Description	Comments
Operation	D	72.75 (e)	*Initial notification:* Reports made by licensees in response to the requirements of this section must be made as follows: 1) Licensees shall make reports required by paragraphs (a), (b), (c), or (d) of this section by telephone to the NRC Headquarters Operations Center. [Note: The commercial telephone number of the NRC Headquarters Operations Center is (301) 816-5100. Those licensees with an available Emergency Notification System (ENS) shall use the ENS to notify the NRC Headquarters Operations Center.] (2) When making a report under paragraphs (a), (b), (c), or (d) of this section, the licensee shall identify: (i) The Emergency Class declared; or (ii) Paragraph (b), "four-hour reports," paragraph (c), "eight- hour reports," or paragraph (d), "24-hour reports," as the paragraph of this section requiring notification of the non-emergency event. (3) To the extent that the information is available at the time of notification, the information provided in these reports must include: (i) The caller's name and call back telephone number; (ii) A description of the event, including date and time; (iii) The exact location of the event; (iv) The quantities and chemical and physical forms of the spent fuel, HLW, or reactor-related GTCC waste involved in the event; and (v) Any personnel radiation exposure data.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (f)	*Follow-up notification:* With respect to the telephone notifications made under paragraphs (a), (b), (c) or (d) of this section, in addition to making the required initial notification, each licensee shall during the course of the event: (1) Immediately report any further degradation in the level of safety of the ISFSI or MRS or other worsening conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made; or any change from one Emergency Class to another; or a termination of the Emergency Class. (2) Immediately report the results of ensuing evaluations or assessments of ISFSI or MRS conditions; the effectiveness of response or protective measures taken; and information related to ISFSI or MRS behavior that is not understood. (3) Maintain an open, continuous communication channel with the NRC Headquarters Operations Center upon request by the NRC.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.

Category	Sub part	Section	Description	Comments
Operation	D	72.75 (g)	*Preparation and submission of written reports.* Each licensee who makes an initial notification required by paragraphs (b)(1), (c)(1), (c)(2), or (d)(1) of this section shall also submit a written follow-up report to the Commission within 60 days of the initial notification. Written reports prepared pursuant to other regulations may be submitted to fulfill this requirement if the reports contain all the necessary information and the appropriate distribution is made. These written reports must be of sufficient quality to permit legible reproduction and optical scanning and must be submitted to the NRC in accordance with § 72.4. These reports must include the items described in § 72.75 (g)(1, 2, 3, 4, 5, 6, and 7):	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(1)	A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence;	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2)	A clear, specific, narrative description of the event that occurred so that knowledgeable readers conversant with the design of an ISFSI or MRS, but not familiar with the details of a particular facility, can understand the complete event. For the particular event, the narrative description must include the specific information, as appropriate, outlined in § 72.75 (g)(2)(i, ii, iii, iv, v, vi, vii, vii, ix, x, xi, and xii).	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2)(i)	The ISFSI or MRS operating conditions before the event.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (ii)	The status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (iii)	The dates and approximate times of occurrences.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (iv)	The cause of each component or system failure or personnel error, if known.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (v)	The failure mode, mechanism, and effect of each failed component, if known.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.

Category	Sub part	Section	Description	Comments
Operation	D	72.75 (g)(2) (vi)	A list of systems or secondary functions that were also affected for failures of components with multiple functions.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (vii)	For wet spent fuel storage systems only, after the failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (viii)	The method of discovery of each component or system failure or procedural error.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (ix)	For each human performance related root cause, the licensee shall discuss the cause(s) and circumstances.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (x)	For wet spent fuel storage systems only, any automatically and manually initiated safety system responses;	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (xi)	The manufacturer and model number (or other identification) of each component that failed during the event,	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(2) (xii)	The quantities and chemical and physical forms of the spent fuel, HLW, or reactor-related GTCC waste involved in the event.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(3)	An assessment of the safety consequences and implications of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(4)	A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(5)	Reference to any previous similar events at the same facility that are known to the licensee.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (g)(6)	The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the facility's characteristics.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.

Category	Sub part	Section	Description	Comments
Operation	D	72.75 (g)(7)	The extent of exposure of individuals to radiation or to radioactive materials without identification of individuals by name.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (h)	*Supplemental information* : The Commission may require the licensee to submit specific additional information beyond that required by paragraph (g) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing, and the licensee shall submit, as specified in § 72.4, the requested information as a supplement to the initial written report.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.
Operation	D	72.75 (i)	*Applicability* : The requirements of this section apply to: (1)(i) Licensees issued a specific license under § 72.40; and (ii) Licensees issued a general license under § 72.210, after the licensee has placed spent fuel on the ISFSI storage pad (if the ISFSI is located inside the collocated protected area, for a reactor licensed under part 50 of this chapter) or after the licensee has transferred spent fuel waste outside the reactor licensee's protected area to the ISFSI storage pad (if the ISFSI is located outside the collocated protected area, for a reactor licensed under part 50 of this chapter). (2) Those non-emergency events specified in paragraphs (b), (c), and (d) of this section that occurred within 3 years of the date of discovery.	Records, Reports, Inspections, and Enforcement - Reporting requirements for specific events and conditions.

Category	Sub part	Section	Description	Comments
Operation	D	72.76 (a)	Except as provided in paragraph (b) of this section, each licensee shall complete in computer-readable format and submit to the Commission a Material Balance Report and a Physical Inventory Listing Report as specified in the instructions in NUREG/BR–0007 and NMMSS Report D–24 ''Personal Computer Data Input for NRC Licensees.'' Copies of these instructions may be obtained either by writing to the U.S. Nuclear Regulatory Commission, Division of Fuel Cycle Safety, Safeguards, and Environmental Review, Washington, DC 20555–0001, or by e-mail to *RidsNmssFcss@nrc.gov* . These reports, as specified by § 74.13 or 40.64 of this chapter, provide information concerning the special nuclear material and/or source material possessed, received, transferred, disposed of, or lost by the licensee. Each report must be submitted within 60 days of the beginning of the physical inventory required by § 72.72(b). The Commission may, when good cause is shown, permit a licensee to submit Material Balance Reports and Physical Inventory Listing Reports at other times. Each licensee required to report material balance and inventory information as described in this part, shall resolve any discrepancies identified during the report review and reconciliation process within 30 calendar days of notification of a discrepancy identified by NRC. The Commission's copy of this report must be submitted to the address specified in the instructions. These prescribed, computer-readable forms replace the DOE/NRC Forms 742 and 742C previously submitted in paper form.	Records, Reports, Inspections, and Enforcement - Material status reports.
Operation	D	72.76 (b)	Any licensee who is required to submit routine material status reports pursuant to § 75.35 of this chapter (pertaining to implementation of the US/IAEA Safeguards Agreement) shall prepare and submit such reports only as provided in that section instead of as provided in paragraph (a) of this section.	Records, Reports, Inspections, and Enforcement - Material status reports.

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Category	Sub part	Section	Description	Comments
Operation	D	72.78 (a)	Except as provided in paragraph (b) of this section, whenever the licensee transfers or receives or adjusts the inventory, in any manner, of special nuclear material as specified by § 74.15 and/or source material as specified by § 40.64 of this chapter, the licensee shall complete in computer-readable format a Nuclear Material Transaction Report as specified in the instructions in NUREG/BR–0006 and NMMSS Report D–24, "Personal Computer Data Input for NRC Licensees." Copies of these instructions may be obtained either by writing to the U.S. Nuclear Regulatory Commission, Division of Fuel Cycle Safety, Safeguards, and Environmental Review, Washington, DC 20555–0001, or by e-mail to RidsNmssFcss@nrc.gov. Each licensee who transfers the material shall submit a Nuclear Material Transaction Report in computer-readable format as specified in the instructions no later than the close of business the next working day. Each licensee who receives the material shall submit a Nuclear Material Transaction Report in computer-readable format in accordance with instructions within ten (10) days after the material is received. Each ISFSI licensee who receives spent fuel from a foreign source shall complete both the supplier's and the receiver's portion of the Nuclear Material Transaction Report, verify the identity of the spent fuel, and indicate the results on the receiver's portion of the form. These prescribed computer-readable forms replace the DOE/NRC Form 741 which have been previously submitted in paper form.	Records, Reports, Inspections, and Enforcement - Nuclear material transaction reports.
Operation	D	72.78 (b)	Any licensee who is required to submit Nuclear Material Transactions Reports pursuant to § 75.34 of this chapter (pertaining to implementation of the US/IAEA Safeguards Agreement) shall prepare and submit the reports only as provided in that section instead of as provided in paragraph (a) of this section.	Records, Reports, Inspections, and Enforcement - Nuclear material transaction reports.
Operation	D	72.79 (a)	In response to a written request by the Commission, each applicant for a certificate of compliance or license and each recipient of a certificate of compliance or specific or general license shall submit facility information, as described in § 75.10 of this chapter, on Form N–71 and associated forms and site information on DOC/NRC Form AP–A and associated forms.	Records, Reports, Inspections, and Enforcement - Facility information and verification.
Operation	D	72.79 (b)	Shall submit location information described in § 75.11 of this chapter on DOC/NRC Form AP–1 and associated forms.	Records, Reports, Inspections, and Enforcement - Facility information and verification.

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Category	Sub part	Section	Description	Comments
Operation	D	72.79 (c)	Shall permit verification thereof by the International Atomic Energy Agency (IAEA) and take other action as necessary to implement the US/IAEA Safeguards Agreement, as described in Part 75 of this chapter.	Records, Reports, Inspections, and Enforcement - Facility information and verification.
Operation	D	72.80 (a)	Each licensee shall maintain any records and make any reports that may be required by the conditions of the license or by the rules, regulations, and orders of the Commission in effectuating the purposes of the Act.	Records, Reports, Inspections, and Enforcement - Other records and reports.
Operation	D	72.80 (b)	Each licensee shall furnish a copy of its annual financial report, including the certified financial statements, to the Commission. However, licensees who submit a Form 10-Q with the Securities and Exchange Commission or a Form 1 with the Federal Energy Regulatory Commission, need not submit the annual financial report or a certified financial statement under this paragraph.	Records, Reports, Inspections, and Enforcement - Other records and reports.
Operation	D	72.80 (c)	Records that are required by the regulations in this part or by the license conditions must be maintained for the period specified by the appropriate regulation or license condition. If a retention period is not otherwise specified, the above records must be maintained until the Commission terminates the license.	Records, Reports, Inspections, and Enforcement - Other records and reports.
Operation	D	72.80 (d)	Any record that must be maintained pursuant to this part may be either the original or a reproduced copy by any state of the art method provided that any reproduced copy is duly authenticated by authorized personnel and is capable of producing a clear and legible copy after storage for the period specified by Commission regulations.	Records, Reports, Inspections, and Enforcement - Other records and reports.
Operation	D	72.80 (e)	Before license termination, the licensee shall forward records required by § 20.2103(b)(4), of this chapter, and § 72.30(f) to the appropriate NRC Regional Office.	Records, Reports, Inspections, and Enforcement - Other records and reports.
Operation	D	72.80 (f)	If licensed activities are transferred or assigned in accordance with § 72.44(b)(1), the licensee shall transfer the records required by § 20.2103(b)(4), of this chapter, and § 72.30(f) to the new licensee and the new licensee will be responsible for maintaining these records until the license is terminated.	Records, Reports, Inspections, and Enforcement - Other records and reports.
Operation	D	72.80 (g)	Each specific licensee shall notify the Commission, in accordance with § 72.4, of its readiness to begin operation at least 90 days prior to the first storage of spent fuel, high-level waste, or reactor-related GTCC waste in an ISFSI or an MRS.	Records, Reports, Inspections, and Enforcement - Other records and reports.

Category	Sub part	Section	Description	Comments
Operation	D	72.82 (a)	Each licensee under this part shall permit duly authorized representatives of the Commission to inspect its records, premises, and activities and of spent fuel, high-level radioactive waste, or reactor-related GTCC waste in its possession related to the specific license as may be necessary to meet the objectives of the Act, including section 105 of the Act.	Records, Reports, Inspections, and Enforcement - Inspections and tests.
Operation	D	72.82 (b)	Each licensee under this part shall make available to the Commission for inspection, upon reasonable notice, records kept by the licensee pertaining to its receipt, possession, packaging, or transfer of spent fuel, high-level radioactive waste, or reactor-related GTCC waste.	Records, Reports, Inspections, and Enforcement - Inspections and tests.
Operation	D	72.82 (c)	(1) Each licensee under this part shall upon request by the Director, Office of Nuclear Material Safety and Safeguards or the appropriate NRC Regional Administrator provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets and janitorial services shall be furnished by each licensee. The office shall be convenient to and have full access to the installation and shall provide the inspector both visual and acoustic privacy. (2) For a site with a single storage installation the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary, and transient NRC personnel and will be generally commensurate with other office facilities at the site. A space of 250 sq. ft., either within the site's office complex or in an office trailer, or other onsite space, is suggested as a guide. For sites containing multiple facilities, additional space may be requested to accommodate additional full-time inspectors. The office space that is provided shall be subject to the approval of the Director, Office of Nuclear Material Safety and Safeguards or the appropriate NRC Regional Administrator. All furniture, supplies and Commission equipment will be furnished by the Commission. (3) Each licensee under this part shall afford any NRC resident inspector assigned to that site, or other NRC inspectors identified by the Regional Administrator as likely to inspect the installation, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection, and personal safety.	Records, Reports, Inspections, and Enforcement - Inspections and tests.
Operation	D	72.82 (d)	Each licensee shall perform, or permit the Commission to perform, such tests as the Commission deems appropriate or necessary for the administrator of the regulations in this part.	Records, Reports, Inspections, and Enforcement - Inspections and tests.

Category	Sub part	Section	Description	Comments
Operation	D	72.84 (a)	The Commission may obtain an injunction or other court order to prevent a violation of the provisions of-- (1) The Atomic Energy Act of 1954, as amended; (2) Title II of the Energy Reorganization Act of 1974, as amended; or (3) A regulation or order issued pursuant to those Acts.	Records, Reports, Inspections, and Enforcement - Violations.
Operation	D	72.84 (b)	The Commission may obtain a court order for the payment of a civil penalty imposed under section 234 of the Atomic Energy Act: (1) For violations of-- (i) Sections 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Atomic Energy Act of 1954, as amended; (ii) Section 206 of the Energy Reorganization Act; (iii) Any rule, regulation, or order issued pursuant to the sections specified in paragraph (b)(1)(i) of this section; (iv) Any term, condition, or limitation of any license issued under the sections specified in paragraph (b)(1)(i) of this section. (2) For any violation for which a license may be revoked under Section 186 of the Atomic Energy Act of 1954, as amended.	Records, Reports, Inspections, and Enforcement - Violations.
Operation	D	72.86 (a)	Section 223 of the Atomic Energy Act of 1954, as amended, provides for criminal sanctions for willful violation of, attempted violation of, or conspiracy to violate, any regulation issued under sections 161b, 161i, or 161o of the Act. For purposes of section 223, all the regulations in part 72 are issued under one or more of sections 161b, 161i, or 161o, except for the sections listed in paragraph (b) of this section.	Records, Reports, Inspections, and Enforcement - Criminal penalties.
Operation	D	72.86 (b)	The regulations in Part 72 that are not issued under sections 161b, 161i, or 161o for the purposes of section 223 are as follows: Secs. 72.1, 72.2, 72.3, 72.4, 72.5, 72.7, 72.8, 72.9, 72.13, 72.16, 72.18, 72.20, 72.22, 72.24, 72.26, 72.28, 72.32, 72.34, 72.40, 72.46, 72.56, 72.58, 72.60, 72.62, 72.84, 72.86, 72.90, 72.96, 72.108, 72.120, 72.122, 72.124, 72.126, 72.128, 72.130, 72.182, 72.194, 72.200, 72.202, 72.204, 72.206, 72.210, 72.214, 72.220, 72.230, 72.238, and 72.240.	Records, Reports, Inspections, and Enforcement - Criminal penalties.
Design	E	72.90 (a)	Site characteristics that may directly affect the safety or environmental impact of the ISFSI or MRS must be investigated and assessed.	Siting Evaluation Factors - General Considerations.
Design	E	72.90 (b)	Proposed sites for the ISFSI or MRS must be examined with respect to the frequency and the severity of external natural and man-induced events that could affect the safe operation of the ISFSI or MRS.	Siting Evaluation Factors - General Considerations.
Design	E	72.90 (c)	Design basis external events must be determined for each combination of proposed site and proposed ISFSI or MRS design.	Siting Evaluation Factors - General Considerations.

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Category	Sub part	Section	Description	Comments
Design	E	72.90 (d)	Proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI or MRS design shall be deemed unsuitable for the location of the ISFSI or MRS.	Siting Evaluation Factors - General Considerations.
Design	E	72.90 (e)	Pursuant to subpart A of part 51 of this chapter for each proposed site for an ISFSI and pursuant to sections 141 or 148 of NWPA, as appropriate (96 Stat. 2241, 101 Stat. 1330-235, 42 U.S.C. 10161, 10168) for each proposed site for an MRS, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and esthetic values.	Siting Evaluation Factors - General Considerations.
Design	E	72.90 (f)	The facility must be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.	Siting Evaluation Factors - General Considerations.
Design	E	72.92 (a)	Natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI or MRS. The important natural phenomena that affect the ISFSI or MRS design must be identified.	Siting Evaluation Factors - Design basis external natural events
Design	E	72.92 (b)	Records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.	Siting Evaluation Factors - Design basis external natural events
Design	E	72.92 (c)	Appropriate methods must be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.	Siting Evaluation Factors - Design basis external natural events
Design	E	72.94 (a)	The region must be examined for both past and present man-made facilities and activities that might endanger the proposed ISFSI or MRS. The important potential man-induced events that affect the ISFSI or MRS design must be identified.	Siting Evaluation Factors - Design basis man-induced events.
Design	E	72.94 (b)	Information concerning the potential occurrence and severity of such events must be collected and evaluated for reliability, accuracy, and completeness.	Siting Evaluation Factors - Design basis man-induced events.
Design	E	72.94 (c)	Appropriate methods must be adopted for evaluating the design basis external man-induced events, based on the current state of knowledge about such events.	Siting Evaluation Factors - Design basis man-induced events.

Category	Sub part	Section	Description	Comments
Design	E	72.96 (a)	An ISFSI which is owned and operated by DOE must not be located at any site within which there is a candidate site for a HLW repository. This limitation shall apply until such time as DOE decides that such candidate site is no longer a candidate site under consideration for development as a HLW repository.	Siting Evaluation Factors - Siting limitations.
Design	E	72.96 (b)	An MRS must not be sited in any State in which there is located any site approved for site characterization for a HLW repository. This limitation shall apply until such time as DOE decides that the candidate site is no longer a candidate site under consideration for development as a repository. This limitation shall continue to apply to any site selected for construction as a repository.	Siting Evaluation Factors - Siting limitations.
Design	E	72.96 (c)	If an MRS is located, or is planned to be located, within 50 miles of the first HLW repository, any Commission decision approving the first HLW repository application must limit the quantity of spent fuel or high-level radioactive waste that may be stored. This limitation shall prohibit the storage of a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal, or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel, in both the repository and the MRS until such time as a second repository is in operation.	Siting Evaluation Factors - Siting limitations.
Design	E	72.96 (d)	An MRS authorized by section 142(b) of NWPA (101 Stat. 1330-232, 42 U.S.C. 10162(b)) may not be constructed in the State of Nevada. The quantity of spent nuclear fuel or high-level radioactive waste that may be stored at an MRS authorized by section 142(b) of NWPA shall be subject to the limitations in § 72.44(g) of this part instead of the limitations in paragraph (c) of this section.	Siting Evaluation Factors - Siting limitations.
Design	E	72.98 (a)	The regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI or MRS must be identified.	Siting Evaluation Factors - Identifying regions around an ISFSI or MRS site.
Design	E	72.98 (b)	The potential regional impact due to the construction, operation or decommissioning of the ISFSI or MRS must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI or MRS activities.	Siting Evaluation Factors - Identifying regions around an ISFSI or MRS site.

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Category	Sub part	Section	Description	Comments
Design	E	72.98 (c)	Those regions identified pursuant to paragraphs (a) and (b) of this section must be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI or MRS.	Siting Evaluation Factors - Identifying regions around an ISFSI or MRS site.
Design	E	72.100 (a)	The proposed site must be evaluated with respect to the effects on populations in the region resulting from the release of radioactive materials under normal and accident conditions during operation and decommissioning of the ISFSI or MRS; in this evaluation both usual and unusual regional and site characteristics shall be taken into account.	Siting Evaluation Factors - Defining potential effects of the ISFSI or MRS on the region.
Design	E	72.100 (b)	Each site must be evaluated with respect to the effects on the regional environment resulting from construction, operation, and decommissioning for the ISFSI or MRS; in this evaluation both usual and unusual regional and site characteristics must be taken into account.	Siting Evaluation Factors - Defining potential effects of the ISFSI or MRS on the region.
Design	E	72.102 (a)	(1) East of the Rocky Mountain Front (east of approximately 104° west longitude), except in areas of known seismic activity including but not limited to the regions around New Madrid, MO, Charleston, SC, and Attica, NY, sites will be acceptable if the results from onsite foundation and geological investigation, literature review, and regional geological reconnaissance show no unstable geological characteristics, soil stability problems, or potential for vibratory ground motion at the site in excess of an appropriate response spectrum anchored at 0.2 g. (2) For those sites that have been evaluated under paragraph (a)(1) of this section that are east of the Rocky Mountain Front, and that are not in areas of known seismic activity, a standardized design earthquake (DE) described by an appropriate response spectrum anchored at 0.25 g may be used. Alternatively, a site-specific DE may be determined by using the criteria and level of investigations required by appendix A of part 100 of this chapter.	Siting Evaluation Factors - Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.
Design	E	72.102 (b)	West of the Rocky Mountain Front (west of approximately 104° west longitude), and in other areas of known potential seismic activity, seismicity will be evaluated by the techniques of appendix A of part 100 of this chapter. Sites that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided.	Siting Evaluation Factors, Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.

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Category	Sub part	Section	Description	Comments
Design	E	72.102 (c)	Sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion.	Siting Evaluation Factors - Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.
Design	E	72.102 (d)	Site-specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.	Siting Evaluation Factors - Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.
Design	E	72.102 (e)	In an evaluation of alternative sites, those which require a minimum of engineered provisions to correct site deficiencies are preferred. Sites with unstable geologic characteristics should be avoided.	Siting Evaluation Factors - Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.
Design	E	72.102 (f)	The design earthquake (DE) for use in the design of structures must be determined as follows: (1) For sites that have been evaluated under the criteria of appendix A of 10 CFR part 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant. (2) Regardless of the results of the investigations anywhere in the continental U.S., the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.	Siting Evaluation Factors - Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.

Category	Sub part	Section	Description	Comments
Design	E	72.103 (a)	(1) East of the Rocky Mountain Front (east of approximately 104° west longitude), except in areas of known seismic activity including but not limited to the regions around New Madrid, MO; Charleston, SC; and Attica, NY; sites will be acceptable if the results from onsite foundation and geological investigation, literature review, and regional geological reconnaissance show no unstable geological characteristics, soil stability problems, or potential for vibratory ground motion at the site in excess of an appropriate response spectrum anchored at 0.2 g. (2) For those sites that have been evaluated under paragraph (a)(1) of this section that are east of the Rocky Mountain Front, and that are not in areas of known seismic activity, a standardized design earthquake ground motion (DE) described by an appropriate response spectrum anchored at 0.25 g may be used. Alternatively, a site-specific DE may be determined by using the criteria and level of investigations required by paragraph (f) of this section. For a site with a co-located nuclear power plant (NPP), the existing geological and seismological design criteria for the NPP may be used. If the existing design criteria for the NPP is used and the site has multiple NPPs, then the criteria for the most recent NPP must be used.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (b)	West of the Rocky Mountain Front (west of approximately 104° west longitude), and in other areas of known potential seismic activity east of the Rocky Mountain Front, seismicity must be evaluated by the techniques presented in paragraph (f) of this section. If an ISFSI or MRS is located on an NPP site, the existing geological and seismological design criteria for the NPP may be used. If the existing design criteria for the NPP is used and the site has multiple NPPs, then the criteria for the most recent NPP must be used.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (c)	Sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (d)	Site-specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.

Category	Sub part	Section	Description	Comments
Design	E	72.103 (e)	In an evaluation of alternative sites, those which require a minimum of engineered provisions to correct site deficiencies are preferred. Sites with unstable geologic characteristics should be avoided.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)	Except as provided in paragraphs (a)(2) and (b) of this section, the DE for use in the design of structures, systems, and components must be determined by items describe in § 72.103 (f)(1, 2, and 3)	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)(1)	*Geological, seismological, and engineering characteristics* . The geological, seismological, and engineering characteristics of a site and its environs must be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the DE, and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. The size of the region to be investigated and the type of data pertinent to the investigations must be determined based on the nature of the region surrounding the proposed site. Data on the vibratory ground motion, tectonic surface deformation, nontectonic deformation, earthquake recurrence rates, fault geometry and slip rates, site foundation material, and seismically induced floods and water waves must be obtained by reviewing pertinent literature and carrying out field investigations. However, each applicant shall investigate all geologic and seismic factors (for example, volcanic activity) that may affect the design and operation of the proposed ISFSI or MRS facility irrespective of whether these factors are explicitly included in this section.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)(2)	*Geologic and seismic siting factors.* The geologic and seismic siting factors considered for design must include the item identified in §72.103 (f)(2)(i, ii, and iii) and other design conditions as stated in paragraph (f)(2)(iv) of this section.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Design	E	72.103 (f)(2)(i)	Determination of the Design Earthquake Ground Motion (DE). The DE for the site is characterized by both horizontal and vertical free-field ground motion response spectra at the free ground surface. In view of the limited data available on vibratory ground motions for strong earthquakes, it usually will be appropriate that the design response spectra be smoothed spectra. The DE for the site is determined considering the results of the investigations required by paragraph (f)(1) of this section. Uncertainties are inherent in these estimates and must be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis (PSHA) or suitable sensitivity analyses.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)(2)(ii)	Determination of the potential for surface tectonic and nontectonic deformations. Sufficient geological, seismological, and geophysical data must be provided to clearly establish if there is a potential for surface deformation.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)(2) (iii)	Determination of design bases for seismically induced floods and water waves. The size of seismically induced floods and water waves that could affect a site from either locally or distantly generated seismic activity must be determined.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)(2) (iv)	Determination of siting factors for other design conditions. Siting factors for other design conditions that must be evaluated include soil and rock stability, liquefaction potential, and natural and artificial slope stability. Each applicant shall evaluate all siting factors and potential causes of failure, such as, the physical properties of the materials underlying the site, ground disruption, and the effects of vibratory ground motion that may affect the design and operation of the proposed ISFSI or MRS.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.
Design	E	72.103 (f)(3)	Regardless of the results of the investigations anywhere in the continental U.S., the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.	Siting Evaluation Factors - Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.

Category	Sub part	Section	Description	Comments
Design, Operation	E	72.104 (a)	During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ as a result of exposure to: (1) Planned discharges of radioactive materials, radon and its decay products excepted, to the general environment, (2) Direct radiation from ISFSI or MRS operations, and (3) Any other radiation from uranium fuel cycle operations within the region.	Siting Evaluation Factors - Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS.
Design, Operation	E	72.104 (b)	Operational restrictions must be established to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations.	Siting Evaluation Factors - Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS.
Design, Operation	E	72.104 (c)	Operational limits must be established for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations to meet the limits given in paragraph (a) of this section.	Siting Evaluation Factors - Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS.
Design	E	72.106 (a)	For each ISFSI or MRS site, a controlled area must be established.	Siting Evaluation Factors - Controlled area of an ISFSI or MRS.
Design	E	72.106 (b)	Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or any extremity may not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel, high-level radioactive waste, or reactor-related GTCC waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.	Siting Evaluation Factors - Controlled area of an ISFSI or MRS.
Design	E	72.106 (c)	The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety.	Siting Evaluation Factors - Controlled area of an ISFSI or MRS.
Design	E	72.108	The proposed ISFSI or MRS must be evaluated with respect to the potential impact on the environment of the transportation of spent fuel, high-level radioactive waste, or reactor-related GTCC waste within the region.	Siting Evaluation Factors - Spent fuel or high-level radioactive waste transportation.

Category	Sub part	Section	Description	Comments
Design	F	72.120 (a)	As required by § 72.24, an application to store spent fuel or reactor-related GTCC waste in an ISFSI or to store spent fuel, high-level radioactive waste, or reactor-related GTCC waste in an MRS must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in § 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI or an MRS. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI or MRS.	General Design Criteria - General considerations.
Design	F	72.120 (b)	The ISFSI must be designed to store spent fuel and/or solid reactor-related GTCC waste. (1) Reactor-related GTCC waste may not be stored in a cask that also contains spent fuel. This restriction does not include radioactive materials that are associated with fuel assemblies (e.g., control rod blades or assemblies, thimble plugs, burnable poison rod assemblies, or fuel channels); (2) Liquid reactor-related GTCC wastes may not be received or stored in an ISFSI; and (3) If the ISFSI is a water-pool type facility, the reactor-related GTCC waste must be in a durable solid form with demonstrable leach resistance.	General Design Criteria - General considerations.
Design	F	72.120 (c)	The MRS must be designed to store spent fuel, solid high-level radioactive waste, and/or solid reactor-related GTCC waste. (1) Reactor-related GTCC waste may not be stored in a cask that also contains spent fuel. This restriction does not include radioactive materials associated with fuel assemblies (e.g., control rod blades or assemblies, thimble plugs, burnable poison rod assemblies, or fuel channels); (2) Liquid high-level radioactive wastes or liquid reactor-related GTCC wastes may not be received or stored in an MRS; and (3) If the MRS is a water-pool type facility, the high-level waste and reactor-related GTCC waste must be in a durable solid form with demonstrable leach resistance.	General Design Criteria - General considerations.

Category	Sub part	Section	Description	Comments
Design	F	72.120 (d)	The ISFSI or MRS must be designed, made of materials, and constructed to ensure that there will be no significant chemical, galvanic, or other reactions between or among the storage system components, spent fuel, reactor-related GTCC waste, and/or high level waste including possible reaction with water during wet loading and unloading operations or during storage in a water-pool type ISFSI or MRS. The behavior of materials under irradiation and thermal conditions must be taken into account.	General Design Criteria - General considerations.
Design	F	72.120 (e)	The NRC may authorize exceptions, on a case-by-case basis, to the restrictions in paragraphs (b) and (c) of this section regarding the commingling of spent fuel and reactor-related GTCC waste in the same cask.	General Design Criteria - General considerations.
Design. Quality Assurance	F	72.122 (a)	*Quality Standards* . Structures, systems, and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.	General Design Criteria - Overall requirements.
Design	F	72.122 (b)	*Protection against environmental conditions and natural phenomen* a. The design needs to address the items described in § 72.122 (b)(1, 2,3, and 4).	General Design Criteria - Overall requirements.
Design, Quality Assurance	F	72.122 (b)(1)	Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI or MRS and to withstand postulated accidents.	General Design Criteria - Overall requirements.

Category	Sub part	Section	Description	Comments
Design	F	72.122 (b)(2)	(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: (A) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (ii) The ISFSI or MRS also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel, high-level radioactive waste, or reactor-related GTCC waste or on to structures, systems, and components important to safety.	General Design Criteria - Overall requirements.
Design, Quality Assurance	F	72.122 (b)(3)	Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety.	General Design Criteria - Overall requirements.
Design	F	72.122 (b)(4)	If the ISFSI or MRS is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.	General Design Criteria - Overall requirements. See NUREG 1567, § 2.4.5 for acceptance criteria related to SITE CHARACTERISTICS and Subsurface Hydrology.
Design, Quality Assurance	F	72.122 (c)	*Protection against fires and explosions* . Structures, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI or MRS, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI or MRS must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.	General Design Criteria - Overall requirements.

Category	Sub part	Section	Description	Comments
Design, Quality Assurance	F	72.122 (d)	*Sharing of structures, systems, and components* . Structures, systems, and components important to safety must not be shared between an ISFSI or MRS and other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety functions, including the ability to return to a safe condition in the event of an accident.	General Design Criteria - Overall requirements.
Design	F	72.122 (e)	*Proximity of sites* . An ISFSI or MRS located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public.	General Design Criteria - Overall requirements.
Design, Quality Assurance	F	72.122 (f)	*Testing and maintenance of systems and components.* Systems and components that are important to safety must be designed to permit inspection, maintenance, and testing.	General Design Criteria - Overall requirements.
Design, Quality Assurance	F	72.122 (g)	*Emergency capability* . Structures, systems, and components important to safety must be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.	General Design Criteria - Overall requirements.
Design	F	72.122 (h)	*Confinement barriers and systems* . The design needs to address the items described in § 72.122 (h)(1, 2, 3, 4 and 5).	General Design Criteria - Overall requirements.
Design	F	72.122 (h)(1)	The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.	General Design Criteria - Overall requirements.
Design	F	72.122 (h)(2)	For underwater storage of spent fuel, high-level radioactive waste, or reactor-related GTCC waste in which the pool water serves as a shield and a confinement medium for radioactive materials, systems for maintaining water purity and the pool water level must be designed so that any abnormal operations or failure in those systems from any cause will not cause the water level to fall below safe limits. The design must preclude installations of drains, permanently connected systems, and other features that could, by abnormal operations or failure, cause a significant loss of water. Pool water level equipment must be provided to alarm in a continuously manned location if the water level in the storage pools falls below a predetermined level.	General Design Criteria - Overall requirements.

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Category	Sub part	Section	Description	Comments
Design	F	72.122 (h)(3)	Ventilation systems and off-gas systems must be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions.	General Design Criteria - Overall requirements.
Design	F	72.122 (h)(4)	Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.	General Design Criteria - Overall requirements.
Design	F	72.122 (h)(5)	The high-level radioactive waste and reactor-related GTCC waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.	General Design Criteria - Overall requirements.
Design	F	72.122 (i)	*Instrumentation and control systems.* Instrumentation and control systems for wet spent fuel and reactor-related GTCC waste storage must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that must remain operational under accident conditions must be identified in the Safety Analysis Report. Instrumentation systems for dry storage casks must be provided in accordance with cask design requirements to monitor conditions that are important to safety over anticipated ranges for normal conditions and off-normal conditions. Systems that are required under accident conditions must be identified in the Safety Analysis Report.	General Design Criteria - Overall requirements.
Design	F	72.122 (j)	*Control room or control area* . A control room or control area, if appropriate for the ISFSI or MRS design, must be designed to permit occupancy and actions to be taken to monitor the ISFSI or MRS safely under normal conditions, and to provide safe control of the ISFSI or MRS under off-normal or accident conditions.	General Design Criteria - Overall requirements.
Design	F	72.122 (k)	Utility or other services. The design needs to address the items described in § 72.122 (k)(1, 2, 3, and 4).	General Design Criteria - Overall requirements.

Category	Sub part	Section	Description	Comments
Design	F	72.122 (k)(1)	Each utility service system must be designed to meet emergency conditions. The design of utility services and distribution systems that are important to safety must include redundant systems to the extent necessary to maintain, with adequate capacity, the ability to perform safety functions assuming a single failure.	General Design Criteria - Overall requirements.
Design	F	72.122 (k)(2)	Emergency utility services must be designed to permit testing of the functional operability and capacity, including the full operational sequence, of each system for transfer between normal and emergency supply sources; and to permit the operation of associated safety systems.	General Design Criteria - Overall requirements.
Design	F	72.122 (k)(3)	Provisions must be made so that, in the event of a loss of the primary electric power source or circuit, reliable and timely emergency power will be provided to instruments, utility service systems, the central security alarm station, and operating systems, in amounts sufficient to allow safe storage conditions to be maintained and to permit continued functioning of all systems essential to safe storage.	General Design Criteria - Overall requirements.
Design	F	72.122 (k)(4)	An ISFSI or MRS which is located on the site of another facility may share common utilities and services with such a facility and be physically connected with the other facility; however, the sharing of utilities and services or the physical connection must not significantly: (i) Increase the probability or consequences of an accident or malfunction of components, structures, or systems that are important to safety; or (ii) Reduce the margin of safety as defined in the basis for any technical specifications of either facility.	General Design Criteria - Overall requirements.
Design	F	72.122 (l)	*Retrievability* . Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal.	General Design Criteria - Overall requirements.
Design	F	72.124 (a)	*Design for criticality safety* . Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.	General Design Criteria - Criteria for nuclear criticality safety.

Category	Sub part	Section	Description	Comments
Design	F	72.124 (b)	*Methods of criticality control* . When practicable, the design of an ISFSI or MRS must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.	General Design Criteria - Criteria for nuclear criticality safety.
Design	F	72.124 (c)	*Criticality Monitoring* . A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under this subpart is not required.	General Design Criteria - Criteria for nuclear criticality safety.
Design	F	72.126 (a)	*Exposure control* . Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to: (1) Prevent the accumulation of radioactive material in those systems requiring access; (2) Decontaminate those systems to which access is required; (3) Control access to areas of potential contamination or high radiation within the ISFSI or MRS; (4) Measure and control contamination of areas requiring access; (5) Minimize the time required to perform work in the vicinity of radioactive components; for example, by providing sufficient space for ease of operation and designing equipment for ease of repair and replacement; and 6) Shield personnel from radiation exposure.	General Design Criteria - Criteria for radiological protection.

Category	Sub part	Section	Description	Comments
Design	F	72.126 (b)	*Radiological alarm systems* . Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibration and testing their operability.	General Design Criteria - Criteria for radiological protection.
Design	F	72.126 (c)	*Effluent and direct radiation monitoring* . (1) As appropriate for the handling and storage system, effluent systems must be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions must be provided for these systems. A means of measuring the flow of the diluting medium, either air or water, must also be provided. (2) Areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.	General Design Criteria - Criteria for radiological protection.
Design	F	72.126 (d)	*Effluent control* . The ISFSI or MRS must be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in § 72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in § 72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.	General Design Criteria - Criteria for radiological protection.

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Category	Sub part	Section	Description	Comments
Design	F	72.128 (a)	*Spent fuel and high-level radioactive waste storage and handling systems* . Spent fuel storage, high-level radioactive waste storage, reactor-related GTCC waste storage and other systems that might contain or handle radioactive materials associated with spent fuel, high-level radioactive waste, or reactor-related GTCC waste, must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with-- (1) A capability to test and monitor components important to safety, (2) Suitable shielding for radioactive protection under normal and accident conditions, (3) Confinement structures and systems, (4) A heat-removal capability having testability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.	General Design Criteria - Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling.
Design	F	72.128 (b)	*Waste treatment* . Radioactive waste treatment facilities must be provided. Provisions must be made for the packing of site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.	General Design Criteria - Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling.
Design, Decommissioning	F	72.130	The ISFSI or MRS must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive wastes and contaminated materials at the time the ISFSI or MRS is permanently decommissioned.	General Design Criteria - Criteria for decommissioning.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Quality Assurance	G	72.140 (a)	*Purpose* . This subpart describes quality assurance requirements that apply to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems, and components, and decommissioning that are important to safety. As used in this subpart, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to control of the physical characteristics and quality of the material or component to predetermined requirements. The certificate holder and applicant for a CoC are responsible for the quality assurance requirements as they apply to the design, fabrication, and testing of a spent fuel storage cask until possession of the spent fuel storage cask is transferred to the licensee. The licensee and the certificate holder are also simultaneously responsible for these quality assurance requirements through the oversight of contractors and subcontractors.	Quality Assurance - Quality assurance requirements.
Quality Assurance	G	72.140 (b)	*Establishment of program* . Each licensee, applicant for a license, certificate holder, applicant for a CoC shall establish, maintain, and execute a quality assurance program satisfying each of the applicable criteria of this subpart, and satisfying any specific provisions which are applicable to the licensee's, applicant's for a license, certificate holder's, and applicant's for a CoC activities. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall execute the applicable criteria in a graded approach to an extent that is commensurate with the quality assurance requirements' importance to safety. The quality assurance program must cover the activities identified in this subpart throughout the life of the activity. For licensees, this includes activities from the site selection through decommissioning prior to termination of the license. For certificate holders, this includes activities from development of the spent fuel storage cask design through termination of the CoC.	Quality Assurance - Quality assurance requirements.

Category	Sub part	Section	Description	Comments
Quality Assurance	G	72.140 (c)	*Approval of program* . (1) Each licensee, applicant for a license, certificate holder, or applicant for a CoC shall file a description of its quality assurance program, including a discussion of which requirements of this subpart are applicable and how they will be satisfied, in accordance with Sec. 72.4. (2) Each licensee shall obtain Commission approval of its quality assurance program prior to receipt of spent fuel and/or reactor-related GTCC waste at the ISFSI or spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste at the MRS. Each licensee or applicant for a specific license shall obtain Commission approval of its quality assurance program before commencing fabrication or testing of a spent fuel storage cask. (3) Each certificate holder or applicant for a CoC shall obtain Commission approval of its quality assurance program before commencing fabrication or testing of a spent fuel storage cask.	Quality Assurance - Quality assurance requirements.
Quality Assurance	G	72.140 (d)	*Previously-approved programs* . A quality assurance program previously approved by the Commission as satisfying the requirements of Appendix B to part 50 of this chapter, subpart H to part 71 of this chapter, or subpart G to this part will be accepted as satisfying the requirements of paragraph (b) of this section, except that a licensee, applicant for a license, certificate holder, and applicant for a CoC who is using an Appendix B or subpart H quality assurance program shall also meet the recordkeeping requirements of Sec. 72.174. In filing the description of the quality assurance program required by paragraph (c) of this section, each licensee, applicant for a license, certificate holder, and applicant for a CoC shall notify the NRC, in accordance with Sec. 72.4, of its intent to apply its previously-approved quality assurance program to ISFSI activities or spent fuel storage cask activities. The notification shall identify the previously-approved quality assurance program by date of submittal to the Commission, docket number, and date of Commission approval.	Quality Assurance - Quality assurance requirements.

Category	Sub part	Section	Description	Comments
Quality Assurance	G	72.142 (a)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall be responsible for the establishment and execution of the quality assurance program. The licensee and certificate holder may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, but the licensee and the certificate holder shall retain responsibility for the program. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall clearly establish and delineate in writing the authority and duties of persons and organizations performing activities affecting the functions of structures, systems, and components which are important to safety. These activities include performing the functions associated with attaining quality objectives and the quality assurance functions.	Quality Assurance - Quality assurance organization.
Quality Assurance	G	72.142 (b)	The quality assurance functions are-- (1) Assuring that an appropriate quality assurance program is established and effectively executed; and (2) Verifying, by procedures such as checking, auditing, and inspection, that activities affecting the functions that are important to safety have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions.	Quality Assurance - Quality assurance organization.
Quality Assurance	G	72.142 (c)	The persons and organizations performing quality assurance functions shall report to a management level that ensures that the required authority and organizational freedom, including sufficient independence from cost and schedule considerations when these considerations are opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms, provided that the persons and organizations assigned the quality assurance functions have the required authority and organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program, at any location where activities subject to this section are being performed, must have direct access to the levels of management necessary to perform this function.	Quality Assurance - Quality assurance organization.

Category	Sub part	Section	Description	Comments
Quality Assurance	G	72.144 (a)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this subpart. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with these procedures throughout the period during which the ISFSI or MRS is licensed or the spent fuel storage cask is certified. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall identify the structures, systems, and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.	Quality Assurance - Quality assurance program.
Quality Assurance	G	72.144 (b)	The licensee, applicant for a license, certificate holder, and applicant for a CoC, through their quality assurance program(s), shall provide control over activities affecting the quality of the identified structures, systems, and components to an extent commensurate with the importance to safety and, as necessary, to ensure conformance with the approved design of each ISFSI, MRS, or spent fuel storage cask. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall ensure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall take into account the need for special controls, processes, test equipment, tools and skills to attain the required quality and the need for verification of quality by inspection and test.	Quality Assurance - Quality assurance program.

Category	Sub part	Section	Description	Comments
Quality Assurance	G	72.144 (c)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall base the requirements and procedures of their quality assurance program(s) on the following considerations concerning the complexity and proposed use of the structures, systems, or components: (1) The impact of malfunction or failure of the item on safety; (2) The design and fabrication complexity or uniqueness of the item; (3) The need for special controls and surveillance over processes and equipment; (4) The degree to which functional compliance can be demonstrated by inspection or test; and (5) The quality history and degree of standardization of the item.	Quality Assurance - Quality assurance program.
Quality Assurance	G	72.144 (d)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that suitable proficiency is achieved and maintained.	Quality Assurance - Quality assurance program.
Quality Assurance	G	72.144 (e)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall review the status and adequacy of the quality assurance program at established intervals. Management of other organizations participating in the quality assurance program must regularly review the status and adequacy of that part of the quality assurance program which they are executing.	Quality Assurance - Quality assurance program.
Quality Assurance, Design	G	72.146 (a)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to ensure that applicable regulatory requirements and the design basis, as specified in the license or CoC application for those structures, systems, and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled. Measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the structures, systems, and components which are important to safety.	Quality Assurance - Design Control.

Category	Sub part	Section	Description	Comments
Quality Assurance, Design	G	72.146 (b)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. The design control measures must provide for verifying or checking the adequacy of design by methods such as design reviews, alternate or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the licensee and certificate holder shall designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the licensee and certificate holder shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall apply design control measures to items such as the following: criticality physics, radiation, shielding, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections and tests.	Quality Assurance - Design Control.
Quality Assurance, Design	G	72.146 (c)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall subject design changes, including field changes, to design control measures commensurate with those applied to the original design. Changes in the conditions specified in the license or CoC require prior NRC approval.	Quality Assurance - Design Control.
Quality Assurance, Design	G	72.148	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the licensee, certificate holder, or by their contractors and subcontractors. To the extent necessary, the licensee, applicant for a license, certificate holder, and applicant for a CoC, shall require contractors or subcontractors to provide a quality assurance program consistent with the applicable provisions of this subpart.	Quality Assurance - Procurement document control.

Category	Sub part	Section	Description	Comments
Quality Assurance, Design	G	72.150	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.	Quality Assurance - Instructions, procedures, and drawings.
Quality Assurance, Design	G	72.152	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must ensure that changes to documents are reviewed and approved.	Quality Assurance - Document control.
Quality Assurance, Operation	G	72.154 (a)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to ensure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures must include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery.	Quality Assurance - Control of purchased material, equipment, and services.
Quality Assurance, Operation	G	72.154 (b)	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall have available documentary evidence that material and equipment conform to the procurement specifications prior to installation or use of the material and equipment. The licensee and certificate holder shall retain or have available this documentary evidence for the life of the ISFSI, MRS, or spent fuel storage cask. The licensee and certificate holder shall ensure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.	Quality Assurance - Control of purchased material, equipment, and services.
Quality Assurance, Operation	G	72.154 (c)	The licensee, applicant for a license, certificate holder, and applicant for a CoC, or a designee of either, shall assess the effectiveness of the control of quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services.	Quality Assurance - Control of purchased material, equipment, and services.

Category	Sub part	Section	Description	Comments
Quality Assurance, Operation	G	72.156	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures for the identification and control of materials, parts, and components. These measures must ensure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item as required, throughout fabrication, installation, and use of the item. These identification and control measures must be designed to prevent the use of incorrect or defective materials, parts, and components.	Quality Assurance - Identification and control of materials, parts, and components.
Quality Assurance, Operation	G	72.158	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to ensure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.	Quality Assurance - Control of special processes.
Quality Assurance, Operation	G	72.160	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish and execute a program for inspection of activities affecting quality by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examinations, measurements, or tests of material or products processed must be performed for each work operation where necessary to assure quality. If direct inspection of processed material or products cannot be carried out, indirect control by monitoring processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when quality control is inadequate without both. If mandatory inspection hold points that require witnessing or inspecting by the licensee's or certificate holder's designated representative, and beyond which work should not proceed without the consent of its designated representative, are required, the specific hold points must be indicated in appropriate documents.	Quality Assurance - Licensee inspection.

Category	Sub part	Section	Description	Comments
Quality Assurance, Operation	G	72.162	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish a test program to ensure that all testing, required to demonstrate that the structures, systems, and components will perform satisfactorily in service, is identified and performed in accordance with written test procedures that incorporate the requirements of this part and the requirements and acceptance limits contained in the ISFSI, MRS, or spent fuel storage cask license or CoC. The test procedures must include provisions to ensure that all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall document and evaluate the test results to ensure that test requirements have been satisfied.	Quality Assurance - Test control.
Quality Assurance, Operation	G	72.164	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to ensure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.	Quality Assurance - Control of measurement and test equipment.
Quality Assurance, Operation	G	72.166	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to control, in accordance with work and inspection instructions, the handling, storage, shipping, cleaning, and preservation of materials and equipment to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels must be specified and provided.	Quality Assurance - Handling, storage, and shipping control.
Quality Assurance, Operation	G	72.168 (a)	The licensee shall establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the ISFSI or MRS. These measures must provide for the identification of items which have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspections and tests.	Quality Assurance - Inspections, tests, and operating status.
Quality Assurance, Operation	G	72.168 (b)	The licensee shall establish measures to identify the operating status of structures, systems, and components of the ISFSI or MRS, such as tagging valves and switches, to prevent inadvertent operation.	Quality Assurance - Inspections, tests, and operating status.

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Category	Sub part	Section	Description	Comments
Quality Assurance, Operation	G	72.170	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to control materials, parts, or components that do not conform to their requirements in order to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.	Quality Assurance - Nonconforming materials, parts, and components.
Quality Assurance, Operation	G	72.172	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of a significant condition identified as adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action is taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.	Quality Assurance - Corrective action.
Quality Assurance, Operation	G	72.174	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall maintain sufficient records to furnish evidence of activities affecting quality. The records must include the following: design records, records of use, and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records must include closely related data such as qualifications of personnel, procedures, and equipment. Inspection and test records must, at a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any noted deficiencies. Records must be identifiable and retrievable. Records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety must be maintained by or under the control of the licensee or certificate holder until the NRC terminates the license or CoC.	Quality Assurance - Quality assurance records.

Category	Sub part	Section	Description	Comments
Quality Assurance, Operation	G	72.176	The licensee, applicant for a license, certificate holder, and applicant for a CoC shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, must be taken where indicated.	Quality Assurance - Audits.
Operation, Design	H	72.180	The licensee shall establish, maintain, and follow a detailed plan for physical protection as described in § 73.51 of this chapter. The licensee shall retain a copy of the current plan as a record until the Commission terminates the license for which the procedures were developed and, if any portion of the plan is superseded, retain the superseded material for 3 years after each change or until termination of the license. The plan must describe how the applicant will meet the requirements of § 73.51 of this chapter and provide physical protection during on-site transportation to and from the proposed ISFSI or MRS and include within the plan the design for physical protection, the lice safeguards contingency plan, and the security organization personnel training and qualification plan. The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with such requirements.	Physical Protection - Physical protection plan.
Design	H	72.182	The design for physical protection must show the site layout and the design features provided to protect the ISFSI or MRS from sabotage. It must include: (a) The design criteria for the physical protection of the proposed ISFSI or MRS; (b) The design bases and the relation of the design bases to the design criteria submitted pursuant to paragraph (a) of this section; and (c) Information relative to materials of construction, equipment, general arrangement, and proposed quality assurance program sufficient to provide reasonable assurance that the final security system will conform to the design bases for the principal design criteria submitted pursuant to paragraph (a) of this section.	Physical Protection - Design for physical protection.

Category	Sub part	Section	Description	Comments
Operation	H	72.184 (a)	The requirements of the licensee's safeguards contingency plan for responding to threats and radiological sabotage must be as defined in appendix C to part 73 of this chapter. This plan must include Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix, the first four categories of information relating to nuclear facilities licensed under part 50 of this chapter. (The fifth and last category of information, Procedures, does not have to be submitted for approval.)	Physical Protection - Safeguards contingency plan.
Operation	H	72.184 (b)	The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with appendix C to 10 CFR part 73 for effecting the actions and decisions contained in the Responsibility Matrix of the licensee's safeguards contingency plan. The licensee shall retain a copy of the current procedures as a record until the Commission terminates the license for which the procedures were developed and, if any portion of the procedures is superseded, retain the superseded material for three years after each change.	Physical Protection, Safeguards contingency plan.
Operation	H	72.186 (a)	The licensee shall make no change that would decrease the safeguards effectiveness of the physical security plan, guard training plan or the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix) contained in the licensee safeguards contingency plan without prior approval of the Commission. A licensee desiring to make a change must submit an application for a license amendment pursuant to § 72.56.	Physical Protection - Change to physical security and safeguards contingency plans.
Operation	H	72.186 (b)	The licensee may, without prior Commission approval, make changes to the physical security plan, guard training plan, or the safeguards contingency plan, if the changes do not decrease the safeguards effectiveness of these plans. The licensee shall maintain records of changes to any such plan made without prior approval for a period of three years from the date of the change, and shall, within two months after the change is made, submit a report addressed to Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, in accordance with § 72.4, containing a description of each change. A copy of the report must be sent to the Regional Administrator of the appropriate NRC Regional Office specified in appendix A to part 73 of this chapter.	Physical Protection - Change to physical security and safeguards contingency plans.

Category	Sub part	Section	Description	Comments
Operation	I	72.190	Operation of equipment and controls that have been identified as important to safety in the Safety Analysis Report and in the license must be limited to trained and certified personnel or be under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of equipment and controls that are important to safety must also be certified in such operations.	Training and Certification of Personnel - Operator requirments.
Licensing	I	72.192	The applicant for a license under this part shall establish a program for training, proficiency testing, and certification of ISFSI or MRS personnel. This program must be submitted to the Commission for approval with the license application.	Training and Certification of Personnel - Operator training and certification program.
Operation	I	72.194	The physical condition and the general health of personnel certified for the operation of equipment and controls that are important to safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, if appropriate provisions are made to accommodate such defect.	Training and Certification of Personnel - Physical requirements.
General	J	72.200 (a)	The Director, Office of Nuclear Material Safety and Safeguards, or the Director's designee shall provide to the Governor and legislature of any State in which an MRS authorized under the Nuclear Waste Policy Act of 1982, as amended, is or may be located, to the Governors of any contiguous States, to each affected unit of local government and to the governing body of any affected Indian tribe, timely and complete information regarding determinations or plans made by the Commission with respect to siting, development, design, licensing, construction, operation, regulation or decommissioning of such monitored retrievable storage facility.	Provision of MRS Information to State Governments and Indian Tribes - Provision of MRS information.
General	J	72.200 (b)	Notwithstanding paragraph (a) of this section, the Director or the Director's designee is not required to distribute any document to any entity if, with respect to such document, that entity or its counsel is included on a service list prepared pursuant to part 2 of this chapter.	Provision of MRS Information to State Governments and Indian Tribes - Provision of MRS information.
General	J	72.200 (c)	Copies of all communications by the Director or the Director's designee under this section must be made available at the NRC Web site, http://www.nrc.gov, and/or at the NRC Public Document Room, and must be furnished to DOE.	Provision of MRS Information to State Governments and Indian Tribes - Provision of MRS information.

Category	Sub part	Section	Description	Comments
Licensing	J	72.202	States, local governmental bodies and affected, Federally-recognized Indian Tribes may participate in license reviews as provided in Subpart C of Part 2 of this chapter.	Provision of MRS Information to State Governments and Indian Tribes - Participation in license reviews.
General	J	72.204	If the Governor and legislature of a State have jointly designated on their behalf a single person or entity to receive notice and information from the Commission under this part, the Commission will provide such notice and information to the jointly designated person or entity instead of the Governor and the legislature separately.	Provision of MRS Information to State Governments and Indian Tribes - Notice to States.
General	J	72.206	Any person who acts under this subpart as a representative for a State (or for the Governor or legislature thereof) or for an affected Indian tribe shall include in the request or other submission, or at the request of the Commission, a statement of the basis of his or her authority to act in such representative capacity.	Provision of MRS Information to State Governments and Indian Tribes - Representation.
Licensing	K	72.210	A general license is hereby issued for the storage of spent fuel in an independent spent fuel storage installation at power reactor sites to persons authorized to possess or operate nuclear power reactors under 10 CFR part 50 or 10 CFR part 52.	General License for Storage of Spent Fuel at Power Reactor Sites - General license issued.
Licensing	K	72.212 (a)	(1) The general license is limited to that spent fuel which the general licensee is authorized to possess at the site under the specific license for the site. (2)This general license is limited to storage of spent fuel in casks approved under the provisions of this part. (3) The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance shall commence upon the date that the particular cask is first used by the general licensee to store spent fuel, shall continue through any renewals of the Certificate of Compliance, unless otherwise specified in the Certificate of Compliance, and shall terminate when the cask’s Certificate of Compliance expires. For any cask placed into service during the final renewal term of a Certificate of Compliance, or during the term of a Certificate of Compliance that was not renewed, the general license for that cask shall terminate after a storage period not to exceed the length of the term certified by the cask’s Certificate of Compliance. Upon expiration of the general license, all casks subject to that general license must be removed from service.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.

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Category	Sub part	Section	Description	Comments
Operation	K	72.212 (b)	The general licensee must adhere to the items described in § 72.212 (b)(1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13 and 14).	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(1)	Notify the Nuclear Regulatory Commission using instructions in § 72.4 at least 90 days before first storage of spent fuel under this general license. The notice may be in the form of a letter, but must contain the licensee's name, address, reactor license and docket numbers, and the name and means of contacting a person responsible for providing additional information concerning spent fuel under this general license. A copy of the submittal must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office listed in appendix D to part 20 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(2)	Register use of each cask with the Nuclear Regulatory Commission no later than 30 days after using that cask to store spent fuel. This registration may be accomplished by submitting a letter using instructions in § 72.4 containing the following information: the licensee's name and address, the licensee's reactor license and docket numbers, the name and title of a person responsible for providing additional information concerning spent fuel storage under this general license, the cask certificate number, the CoC amendment number to which the cask conforms, unless loaded under the initial certificate, cask model number, and the cask identification number. A copy of each submittal must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office listed in appendix D to part 20 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(3)	Ensure that each cask used by the general licensee conforms to the terms, conditions, and specifications of a CoC or an amended CoC listed in § 72.214.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.

Category	Sub part	Section	Description	Comments
Operation	K	72.212 (b)(4)	In applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, register each such cask with the Nuclear Regulatory Commission no later than 30 days after applying the changes authorized by the amended CoC. This registration may be accomplished by submitting a letter using instructions in § 72.4 containing the following information: the licensee's name and address, the licensee's reactor license and docket numbers, the name and title of a person responsible for providing additional information concerning spent fuel storage under this general license, the cask certificate number, the CoC amendment number to which the cask conforms, cask model number, and the cask identification number. A copy of each submittal must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office listed in appendix D to part 20 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(5)	Perform written evaluations, before use and before applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which establish that: (i) The cask, once loaded with spent fuel or once the changes authorized by an amended CoC have been applied, will conform to the terms, conditions, and specifications of a CoC or an amended CoC listed in § 72.214; (ii) Cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion; and (iii) The requirements of § 72.104 have been met. A copy of this record shall be retained until spent fuel is no longer stored under the general license issued under § 72.210.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(6)	Review the Safety Analysis Report referenced in the CoC or amended CoC and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(5) of this section.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.

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Category	Sub part	Section	Description	Comments
Operation	K	72.212 (b)(7)	Evaluate any changes to the written evaluations required by paragraphs (b)(5) and (b)(6) of this section using the requirements of § 72.48(c). A copy of this record shall be retained until spent fuel is no longer stored under the general license issued under § 72.210.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Licensing, Operation	K	72.212 (b)(8)	Before use of the general license, determine whether activities related to storage of spent fuel under this general license involve a change in the facility Technical Specifications or require a license amendment for the facility pursuant to § 50.59(c) of this chapter. Results of this determination must be documented in the evaluations made in paragraph (b)(5) of this section.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(9)	Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee’s physical security plan pursuant to § 73.55 of this chapter with the following additional conditions and exceptions outlined in § 72.212 (b)(9)(i, ii, iii, iv, v, and vi).	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(9)(i)	The physical security organization and program for the facility must be modified as necessary to assure that activities conducted under this general license do not decrease the effectiveness of the protection of vital equipment in accordance with § 73.55 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(9) (ii)	Storage of spent fuel must be within a protected area, in accordance with § 73.55(e) of this chapter, but need not be within a separate vital area. Existing protected areas may be expanded or new protected areas added for the purpose of storage of spent fuel in accordance with this general license.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(9) (iii)	For the purpose of this general license, personnel searches required by § 73.55(h) of this chapter before admission to a new protected area may be performed by physical pat-down searches of persons in lieu of firearms and explosives detection equipment.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(9) (iv)	The observational capability required by § 73.55(i)(3) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of video surveillance technology.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(9) (v)	For the purpose of this general license, the licensee is exempt from requirements to interdict and neutralize threats in § 73.55 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.

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Category	Sub part	Section	Description	Comments
Operation	K	72.212 (b)(9) (vi)	Each general licensee that receives and possesses power reactor spent fuel and other radioactive materials associated with spent fuel storage shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 and the requirements of § 73.22 or § 73.23 of this chapter, as applicable.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(10)	Review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(11)	Maintain a copy of the CoC and, for those casks to which the licensee has applied the changes of an amended CoC, the amended CoC, and the documents referenced in such Certificates, for each cask model used for storage of spent fuel, until use of the cask model is discontinued. The licensee shall comply with the terms, conditions, and specifications of the CoC and, for those casks to which the licensee has applied the changes of an amended CoC, the terms, conditions, and specifications of the amended CoC, including but not limited to, the requirements of any AMP put into effect as a condition of the NRC approval of a CoC renewal application in accordance with § 72.240.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(12)	Accurately maintain the record provided by the CoC holder for each cask that shows, in addition to the information provided by the CoC holder, the following: (i) The name and address of the CoC holder or lessor; (ii) The listing of spent fuel stored in the cask; and (iii) Any maintenance performed on the cask.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(13)	Conduct activities related to storage of spent fuel under this general license only in accordance with written procedures.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (b)(14)	Make records and casks available to the Commission for inspection.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (c)	The record described in paragraph (b)(12) of this section must include sufficient information to furnish documentary evidence that any testing and maintenance of the cask has been conducted under an NRC-approved quality assurance program.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Operation	K	72.212 (d)	In the event that a cask is sold, leased, loaned, or otherwise transferred to another registered user, the record described in paragraph (b)(12) of this section must also be transferred to and must be accurately maintained by the new registered user. This record must be maintained by the current cask user during the period that the cask is used for storage of spent fuel and retained by the last user until decommissioning of the cask is complete.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.212 (e)	Fees for inspections related to spent fuel storage under this general license are those shown in § 170.31 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Conditions of general license issued under § 72.210.
Operation	K	72.214	This section contains the list of each approved spent fuel storage cask. For each approved cask, the following items are provided: Certificate Number; Initial Certificate Effective Date; Amendment Number # Effective Date; SAR Submitted by; SAR Title; Docket Number; Certificate Expiration Date; and Model Number.	General License for Storage of Spent Fuel at Power Reactor Sites - List of approved spent fuel storage casks.
Licensing, Operation	K	72.218 (a)	The notification regarding the program for the management of spent fuel at the reactor required by § 50.54(bb) of this chapter must include a plan for removal of the spent fuel stored under this general license from the reactor site. The plan must show how the spent fuel will be managed before starting to decommission systems and components needed for moving, unloading, and shipping this spent fuel.	General License for Storage of Spent Fuel at Power Reactor Sites - Termination of licenses.
Licensing, Operation	K	72.218 (b)	An application for termination of a reactor operating license issued under 10 CFR part 50 and submitted under § 50.82 of this chapter, or a combined license issued under 10 CFR part 52 and submitted under § 52.110 of this chapter, must contain a description of how the spent fuel stored under this general license will be removed from the reactor site.	General License for Storage of Spent Fuel at Power Reactor Sites - Termination of licenses.
Licensing, Operation	K	72.218 (c)	The reactor licensee shall send a copy of submittals under § 72.218(a) and (b) to the administrator of the appropriate Nuclear Regulatory Commission regional office shown in appendix D to part 20 of this chapter.	General License for Storage of Spent Fuel at Power Reactor Sites - Termination of licenses.
Operation	K	72.220	This general license is subject to the provisions of § 72.84 for violation of the regulations under this part.	General License for Storage of Spent Fuel at Power Reactor Sites - Violations.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	L	72.230 (a)	An application for approval of a spent fuel storage cask design must be submitted in accordance with the instructions contained in § 72.4. A safety analysis report describing the proposed cask design and how the cask should be used to store spent fuel safely must be included with the application.	Approval of Spent Fuel Storage Casks - Procedures for spent fuel storage cask submittals.
Licensing	L	72.230 (b)	Casks that have been certified for transportation of spent fuel under part 71 of this chapter may be approved for storage of spent fuel under this subpart. An application must be submitted in accordance with the instructions contained in § 72.4, for a proposed term not to exceed 40 years. A copy of the CoC issued for the cask under part 71 of this chapter, and drawings and other documents referenced in the certificate, must be included with the application. A safety analysis report showing that the cask is suitable for storage of spent fuel, for the term proposed in the application, must also be included.	Approval of Spent Fuel Storage Casks - Procedures for spent fuel storage cask submittals.
Licensing	L	72.230 (c)	*Public inspection* . An application for the approval of a cask for storage of spent fuel may be made available for public inspection under § 72.20.	Approval of Spent Fuel Storage Casks - Procedures for spent fuel storage cask submittals.
Licensing	L	72.230 (d)	*Fees* . Fees for reviews and evaluations related to issuance of a spent fuel storage cask Certificate of Compliance and inspections related to storage cask fabrication are those shown in § 170.31 of this chapter.	Approval of Spent Fuel Storage Casks - Procedures for spent fuel storage cask submittals.
Licensing	L	72.232 (a)	The certificate holder and applicant for a CoC shall permit, and make provisions for, the NRC to inspect the premises and facilities where a spent fuel storage cask is designed, fabricated, and tested.	Approval of Spent Fuel Storage Casks - Inspection and tests.
Licensing	L	72.232 (b)	The certificate holder and applicant for a CoC shall make available to the NRC for inspection, upon reasonable notice, records kept by them pertaining to the design, fabrication, and testing of spent fuel storage casks.	Approval of Spent Fuel Storage Casks - Inspection and tests.
Licensing	L	72.232 (c)	The certificate holder and applicant for a CoC shall perform, and make provisions that permit the NRC to perform, tests that the Commission deems necessary or or appropriate for the administration of the regulations in this part.	Approval of Spent Fuel Storage Casks - Inspection and tests.
Licensing	L	72.232 (d)	The certificate holder and applicant for a CoC shall submit a notification under § 72.4 at least 45 days prior to starting fabrication of the first spent fuel storage cask under a Certificate of Compliance.	Approval of Spent Fuel Storage Casks - Inspection and tests.
Licensing	L	72.234 (a)	The certificate holder and applicant for a CoC shall ensure that the design, fabrication, testing, and maintenance of a spent fuel storage cask comply with the requirements in § 72.236.	Approval of Spent Fuel Storage Casks - Conditions of approval.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	L	72.234 (b)	The certificate holder and applicant for a CoC shall ensure that the design, fabrication, testing, and maintenance of spent fuel storage casks are conducted under a quality assurance program that meets the requirements of subpart G of this part.	Approval of Spent Fuel Storage Casks - Conditions of approval.
Licensing	L	72.234 (c)	An applicant for a CoC may begin fabrication of spent fuel storage casks before the Commission issues a CoC for the cask; however, applicants who begin fabrication of casks without a CoC do so at their own risk. A cask fabricated before the CoC is issued shall be made to conform to the issued CoC before being placed in service or before spent fuel is loaded.	Approval of Spent Fuel Storage Casks - Conditions of approval.
Licensing	L	72.234 (d)	(1) The certificate holder shall ensure that a record is established and maintained for each spent fuel storage cask fabricated under the CoC. (2) This record must include: (i) The NRC CoC number; (ii) The spent fuel storage cask model number; (iii) The spent fuel storage cask identification number; (iv) Date fabrication was started; (v) Date fabrication was completed; (vi) Certification that the spent fuel storage cask was designed, fabricated, tested, and repaired in accordance with a quality assurance program accepted by NRC; (vii) Certification that inspections required by § 72.236(j) were performed and found satisfactory; and (viii) The name and address of the licensee using the spent fuel storage cask. (3) The certificate holder shall supply the original of this record to the licensees using the spent fuel storage cask. A current copy of a composite record of all spent fuel storage casks manufactured under a CoC, showing the information in paragraph (d)(2) of this section, must be initiated and maintained by the certificate holder for each model spent fuel storage cask. If the certificate holder permanently ceases production of spent fuel storage casks under a CoC, the certificate holder shall send this composite record to the Commission using instructions in § 72.4.	Approval of Spent Fuel Storage Casks - Conditions of approval.
Operation, Quality Assurance	L	72.234 (e)	The certificate holder and the licensees using the spent fuel storage cask shall ensure that the composite record required by paragraph (d) of this section is available to the Commission for inspection.	Approval of Spent Fuel Storage Casks - Conditions of approval.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Operation, Quality Assurance	L	72.234 (f)	The certificate holder shall ensure that written procedures and appropriate tests are established prior to use of the spent fuel storage casks. A copy of these procedures and tests must be provided to each licensee using the spent fuel storage cask.	Approval of Spent Fuel Storage Casks - Conditions of approval.
Design	L	72.236	The certificate holder and applicant for a CoC shall ensure that the requirements, described in (a, b, c, d, e, f, g, h, I, j, k, l, m, and n) of this section are met.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (a)	Specifications must be provided for the spent fuel to be stored in the spent fuel storage cask, such as, but not limited to, type of spent fuel (i.e., BWR, PWR, both), maximum allowable enrichment of the fuel prior to any irradiation, burn-up (i.e., megawatt-days/MTU), minimum acceptable cooling time of the spent fuel prior to storage in the spent fuel storage cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), the inerting atmosphere requirements.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (b)	Design bases and design criteria must be provided for structures, systems, and components important to safety.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (c)	The spent fuel storage cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (d)	Radiation shielding and confinement features must be provided sufficient to meet the requirements in §§ 72.104 and 72.106.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (e)	The spent fuel storage cask must be designed to provide redundant sealing of confinement systems.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (f)	The spent fuel storage cask must be designed to provide adequate heat removal capacity without active cooling systems.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (g)	The spent fuel storage cask must be designed to store the spent fuel safely for the term proposed in the application, and permit maintenance as required.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (h)	The spent fuel storage cask must be compatible with wet or dry spent fuel loading and unloading facilities.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.

Category	Sub part	Section	Description	Comments
Design	L	72.236 (i)	The spent fuel storage cask must be designed to facilitate decontamination to the extent practicable.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Operation, Quality Assurance	L	72.236 (j)	The spent fuel storage cask must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design, Quality Assurance	L	72.236 (k)	The spent fuel storage cask must be conspicuously and durably marked with-- (1) A model number; (2) A unique identification number; and (3) An empty weight.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Operation, Quality Assurance	L	72.236 (l)	The spent fuel storage cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Design	L	72.236 (m)	To the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Operation	L	72.236 (n)	Safeguards Information shall be protected against unauthorized disclosure in accordance with the requirements of § 73.21 and the requirements of § 73.22 or § 73.23 of this chapter, as applicable.	Approval of Spent Fuel Storage Casks - Specific requirements for spent fuel storage cask approval and fabrication.
Operation	L	72.238	A Certificate of Compliance for a cask model will be issued by NRC for a term not to exceed 40 years on a finding that the requirements in § 72.236(a) through (i) are met.	Approval of Spent Fuel Storage Casks - Issuance of an NRC Certificate of Compliance.
Licensing	L	72.240 (a)	The certificate holder may apply for renewal of the design of a spent fuel storage cask for a term not to exceed 40 years. In the event that the certificate holder does not apply for a cask design renewal, any licensee using a spent fuel storage cask, a representative of such licensee, or another certificate holder may apply for a renewal of that cask design for a term not to exceed 40 years.	Approval of Spent Fuel Storage Casks - Conditions for spent fuel storage cask renewal.
Licensing	L	72.240 (b)	The application for renewal of the design of a spent fuel storage cask must be submitted not less than 30 days before the expiration date of the CoC. When the applicant has submitted a timely application for renewal, the existing CoC will not expire until the application for renewal has been determined by the NRC.	Approval of Spent Fuel Storage Casks - Conditions for spent fuel storage cask renewal.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Licensing	L	72.240 (c)	The application must be accompanied by a safety analysis report (SAR). The SAR must include the following: (1) Design bases information as documented in the most recently updated final safety analysis report (FSAR) as required by § 72.248; (2) Time-limited aging analyses that demonstrate that structures, systems, and components important to safety will continue to perform their intended function for the requested period of extended operation; and (3) A description of the AMP for management of issues associated with aging that could adversely affect structures, systems, and components important to safety.	Approval of Spent Fuel Storage Casks - Conditions for spent fuel storage cask renewal.
Licensing	L	72.240 (d)	The design of a spent fuel storage cask will be renewed if the conditions in subpart G of this part and § 72.238 are met, and the application includes a demonstration that the storage of spent fuel has not, in a significant manner, adversely affected structures, systems, and components important to safety.	Approval of Spent Fuel Storage Casks - Conditions for spent fuel storage cask renewal.
Licensing	L	72.240 (e)	In approving the renewal of the design of a spent fuel storage cask, the NRC may revise the CoC to include terms, conditions, and specifications that will ensure the safe operation of the cask during the renewal term, including but not limited to, terms, conditions, and specifications that will require the implementation of an AMP.	Approval of Spent Fuel Storage Casks - Conditions for spent fuel storage cask renewal.
Operation, Quality Assurance	L	72.242 (a)	Each certificate holder or applicant shall maintain any records and produce any reports that may be required by the conditions of the CoC or by the rules, regulations, and orders of the NRC in effectuating the purposes of the Act.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (b)	Records that are required by the regulations in this part or by conditions of the CoC must be maintained for the period specified by the appropriate regulation or the CoC conditions. If a retention period is not specified, the records must be maintained until the NRC terminates the CoC.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (c)	Any record maintained under this part may be either the original or a reproduced copy by any state-of-the-art method provided that any reproduced copy is duly authenticated by authorized personnel and is capable of producing a clear and legible copy after storage for the period specified by NRC regulations.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.

A-2. DETAILED REQUIREMENTS of 10CFR72

Category	Sub part	Section	Description	Comments
Operation, Quality Assurance	L	72.242 (d)	Each certificate holder shall submit a written report to the NRC within 30 days of discovery of a design or fabrication deficiency, for any spent fuel storage cask which has been delivered to a licensee, when the design or fabrication deficiency affects the ability of structures, systems, and components important to safety to perform their intended safety function. The written report shall be sent to the NRC in accordance with the requirements of § 72.4. The report shall include the items described in § 72.242 (d)(1, 2, 3, 4, 5, and 6).	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (d)(1)	A brief abstract describing the deficiency, including all component or system failures that contributed to the deficiency and corrective action taken or planned to prevent recurrence.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (d)(2)	A clear, specific, narrative description of what occurred so that knowledgeable readers familiar with the design of the spent fuel storage cask, but not familiar with the details of a particular cask, can understand the deficiency. The narrative description shall include the following specific information as appropriate for the particular event: (i) Dates and approximate times of discovery; (ii) The cause of each component or system failure, if known; (iii) The failure mode, mechanism, and effect of each failed component, if known; (iv) A list of systems or secondary functions that were also affected for failures of components with multiple functions; (v) The method of discovery of each component or system failure; (vi) The manufacturer and model number (or other identification) of each component that failed during the event; (vii) The model and serial numbers of the affected spent fuel storage casks; and (viii) The licensees that have affected spent fuel storage casks.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (d)(3)	An assessment of the safety consequences and implications of the deficiency. This assessment shall include the availability of other systems or components that could have performed the same function as the components and systems that were affected.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (d)(4)	A description of any corrective actions planned as a result of the deficiency, including those to reduce the probability of similar occurrences in the future.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.

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Category	Sub part	Section	Description	Comments
Operation, Quality Assurance	L	72.242 (d)(5)	Reference to any previous similar deficiencies at the same facility that are known to the certificate holder.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Operation, Quality Assurance	L	72.242 (d)(6)	The name and telephone number of a person within the certificate holder's organization who is knowledgeable about the deficiency and can provide additional information.	Approval of Spent Fuel Storage Casks - Recordkeeping and reports.
Licensing, Operation	L	72.244	Whenever a certificate holder desires to amend the CoC (including a change to the terms, conditions or specifications of the CoC), an application for an amendment shall be filed with the Commission fully describing the changes desired and the reasons for such changes, and following as far as applicable the form prescribed for original applications.	Approval of Spent Fuel Storage Casks - Application for amendment of a certificate of compliance.
Licensing, Operation	L	72.246	In determining whether an amendment to a CoC will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of an initial CoC.	Approval of Spent Fuel Storage Casks - Issuance of amendment to a certificate of compliance.
Licensing, Operation	L	72.248 (a)	Each certificate holder for a spent fuel storage cask design shall update periodically, as provided in paragraph (b) of this section, the final safety analysis report (FSAR) to assure that the information included in the report contains the latest information developed. (1) Each certificate holder shall submit an original FSAR to the Commission, in accordance with § 72.4, within 90 days after the spent fuel storage cask design has been approved pursuant to § 72.238. (2) The original FSAR shall be based on the safety analysis report submitted with the application and reflect any changes and applicant commitments developed during the cask design review process. The original FSAR shall be updated to reflect any changes to requirements contained in the issued Certificate of Compliance (CoC).	Approval of Spent Fuel Storage Casks - Safety analysis report updating.
Licensing, Operation	L	72.248 (b)	Each update shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the certificate holder or prepared by the certificate holder pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last update to the FSAR under this section. The update shall include the effects of: [Note: Effects of changes includes appropriate revisions of descriptions in the FSAR such that the FSAR (as updated) is complete and accurate.]	Approval of Spent Fuel Storage Casks - Safety analysis report updating.

Category	Sub part	Section	Description	Comments
Licensing, Operation	L	72.248 (c)	(1) The update of the FSAR must be filed in accordance with § 72.4. If the update is filed on paper, then it should be filed on a page-replacement basis; if filed electronically, it should be filed on a full replacement basis. See Guidance for Electronic Submissions to the Commission at http://www.nrc.gov/site-help/e-submittals.html. (2) A paper update filed on a page-replacement basis must include a list that identifies the current pages of the FSAR following page replacement. If the update is filed electronically on a full replacement basis, it must include a list of changed pages. (3) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both); (4) The update shall include: (i) A certification by a duly authorized officer of the certificate holder that either the information accurately presents changes made since the previous submittal, or that no such changes were made; and (ii) An identification of changes made by the certificate holder under the provisions of § 72.48, but not previously submitted to the Commission; (5) The update shall reflect all changes implemented up to a maximum of 6 months prior to the date of filing; (6) Updates shall be filed every 24 months from the date of issuance of the CoC; and (7) The certificate holder shall provide a copy of the updated FSAR to each general and specific licensee using its cask design.	Approval of Spent Fuel Storage Casks - Safety analysis report updating.
Licensing, Operation	L	72.248 (d)	The updated FSAR shall be retained by the certificate holder until the Commission terminates the certificate.	Approval of Spent Fuel Storage Casks - Safety analysis report updating.
Licensing, Operation	L	72.248 (e)	A certificate holder who permanently ceases operation, shall provide the updated FSAR to the new certificate holder or to the Commission, as appropriate, in accordance with § 72.234(d)(3).	Approval of Spent Fuel Storage Casks - Safety analysis report updating.

Categories:
Construction
Decommissioning
Design

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Category	Sub part	Section	Description	Comments

General
Licensing
Operaton
Quality Assurance

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
General	1.4	**GENERAL DESCRIPTION** **Acceptance Criteria**	72.22, 72.24, 72.44	The general description should enable all reviewers, regardless of their specific review assignments, to obtain a basic understanding of the principal function and design features of the proposed installation. Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation, (Dry Storage)," provides guidance regarding information that should be included in the general description. Because much of the information relevant to this initial aspect of the review is presented in more detail in other chapters of this Standard Review Plan (SRP), this chapter focuses on familiarization with the system and should be consistent with the remaining sections of the SAR.
Design	2.4	**SITE CHARACTERISTICS** **Acceptance Criteria**	72.24, 72.40, 72.90, 72.92, 72.94, 72.96, 72.98, 72.100, 72.102, 72.122	The specific acceptance criteria for methods used to identify design criteria are presented in the appropriate parts of this section. No specific acceptance criteria for factors such as atmospheric dispersion or population location are applied in assessing the impacts. Rather, the applicant must supply accurate information so that realistic impacts can be estimated.
Design	2.4.1	**SITE CHARACTERISTICS** **Geography and Demography**	72.24, 72.90, 72.96, 72.98, 72.100, 72.122	10 CFR 72.90, 72.98, 72.100, and 72.122 require that the SAR contain information about the site geography, population, and water and land uses. The criteria given here indicate the kind and degree of detail of information required in an application before a reviewer can validate its adequacy for use in an impact analysis.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.1.1	**SITE CHARACTERISTICS** **Geography and Demography** Site Location	72.24, 72.90, 72.96	Information on site location of the proposed ISFSI and nearby facilities should clearly describe the location by stating the site's host State and county, and its latitude, longitude, and Universal Transverse Mercator coordinates. Maps and aerial photographs of the site should be presented with radial coverage extending a minimum of 8 km (5 mi) from the site. A detailed map of the site area should clearly show adjacent buildings, roads, railroads, transmission lines, wetlands, and surface water bodies. The reviewer should be aware of the limitations on ISFSI and MRS siting which are listed in 10 CFR 72.96, and the potential changes to these limitations which may have been enacted by Congress.
Design	2.4.1.2	**SITE CHARACTERISTICS** **Geography and Demography** Site Description	72.24, 72.90, 72.104	A site map should clearly indicate the site boundary and the controlled area (if different from the site boundary), controlled area access points, and the distances from the boundary to significant features of the installation. The SAR should discuss the applicant's legal responsibilities for the properties described, such as ownership, lease, or easements. Topographic maps should reveal the site topography and surface drainage patterns as well as roads, railroads, transmission lines, wetlands, and surface water bodies on the site. Vegetative cover and surface soil characteristics should be described to facilitate evaluation of fire hazards and erosion. Other activities conducted by the applicant within the controlled area should be identified, and potential interactions with ISFSI operation discussed.
Design	2.4.1.3	**SITE CHARACTERISTICS** **Geography and Demography** Population Distribution and Trends	72.24, 72.90	Current population data and projections should be presented. A sector map of population should divide the area within a 8-km (5-mi) radius of the site by concentric circles with radii of 1.5, 3, 5, 6.5 and 8 km (approximately 1, 2, 3, 4, and 5 miles), and by 22.5-degree segments, each segment centered on one of the 16 compass points. Current and projected populations in each sector should be given. The population data should be overlain on a base map which shows any cities or towns. The maximally exposed individual(s) should be specifically identified and a rationale for their selection (e.g., nearest well, closest person downwind in the predominant wind direction)presented.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.1.4	**SITE CHARACTERISTICS** **Geography and Demography** Land and Water Use	72.24, 72.90	Use of land and water within an 8-km (5-mi) radius should be described. Residential, farming, dairy, industrial, and recreational uses of land and water should be presented in sufficient detail to allow estimates of concentrations of radionuclides to populations from any airborne or liquid effluents.
Design	2.4.2	**SITE CHARACTERISTICS** **Nearby Industrial, Transportation, and Military Facilities**	72.24, 72.40, 72.90, 72.94 (a), 72.96, 72.98, 72.100, and 72.122	10 CFR 72.94 requires that the region be examined for man-made facilities that might endanger the proposed ISFSI or MRS. The SAR should indicate the locations of nearby industrial, transportation, military, and nuclear installations on a map which clearly shows their distance and relationship to the ISFSI. All facilities within an 8-km (5-mi) radius and all relevant facilities at greater distances should be included. For each facility, the products or materials produced, stored, or transported should be described, and any potential hazards to the ISFSI from activities or materials at the facilities should be discussed. Any effect of these facilities on the specific ISFSI design basis should also be discussed.
Design	2.4.3	**SITE CHARACTERISTICS** **Meteorology**	72.24, 72.40, 72.90 (a), 72.92, 72.98, and 72.122	10 CFR 72.90 requires that site characteristics affecting the safety of the proposed ISFSI or MRS must be assessed. The SAR should describe the meteorological conditions at the site and vicinity. Conditions which influence the design and operation of the facility should be identified, and sources of all information should be stated. Enough information should be provided to permit NRC staff to independently evaluate atmospheric diffusion characteristics of the site area. Enough information should also be provided to permit NRC staff to determine the basis for the high winds (either straight line or tornado winds) and high temperature used in the design basis.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.3.1	**SITE CHARACTERISTICS** **Meteorology** Regional Climatology	72.90 (a), 72.92, 72.122	The SAR should describe the climate of the region, including temperature, precipitation, relative humidity, general airflow, pressure patterns, cloud cover, average wind speeds, and prevalent wind direction. Ranges and seasonal variations of these parameters should be discussed. Climate characteristics attributable to terrain should be mentioned. Data on the frequency, intensity, and duration of severe weather should be presented. For example, the SAR should address: temperature, wind, and precipitation extremes; hurricanes, tropical storms, tornadoes, lightning strikes; and snow, ice, and hail storms. Data sources and reliability should be discussed. The rationale for the design basis winds and temperature should be stated in the application.
Design	2.4.3.2	**SITE CHARACTERISTICS** **Meteorology** Local Meteorology	72.90 (a), 72.92, 72.122	The description of local meteorology should summarize data on temperature, wind speed and direction, and relative humidity collected onsite as well as at nearby weather stations. The representativeness of data collected offsite should be discussed. If offsite data adequately represent onsite conditions, then onsite data may not be necessary. For the purpose of evaluating atmospheric diffusion, topographic maps at two different scales should be provided. One shouldshow detailed topographic features, as modified by the facility, within an 8-km (5-mi) radius of the site. A smaller-scale map should show topography out to a 16-km (10-mi) radius. This map should be accompanied by profiles of maximum elevation over distance from the center of the installation out to 16-km (10-mi) for each of the 22.5 degree compass-point sectors.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.3.3	**SITE CHARACTERISTICS** **Meteorology** Onsite Meteorological Measurement Program	72.90, 72.92, 72.122	The meteorological data collected onsite should be reviewed to ensure its adequacy for NRC staff to conduct independent atmospheric dispersion estimates for both postulated accidents and expected routine releases of gaseous effluents. The meteorological data should be provided in the form of joint frequency distributions of wind speed and wind direction by atmospheric stability class. The SAR should state the measurements made, the locations and elevations of measurements, descriptions of the instruments used, instrument performance specifications, calibration and maintenance procedures, and data analysis procedures. Any onsite program and any programs to be used during operations to estimate offsite concentrations of airborne effluents should be described in conformity to Regulatory Guide 1.23, "Onsite Meteorological Programs," criteria for an acceptable onsite meteorological measurements program, and its format for presenting stability class data. If no onsite measurement program exists, the applicant should provide justification for using data from nearby stations.
Design	2.4.4	**SITE CHARACTERISTICS** **Surface Hydrology**	72.24, 72.40, 72.90, 72.92, 72.98 (c)(2), 72.122	10 CFR 72.98 requires that the present and future uses of land and water within the region be investigated. The SAR should contain adequate information for an independent review of all surface hydrology-related design bases, performance requirements, and operating procedures important to safety.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.4.1	**SITE CHARACTERISTICS** **Surface Hydrology** Hydrologic Description	72.98 (c)(2)	The SAR should characterize the surface hydrologic features of the region, area, and site, because this information is the basis for hydrologic engineering analyses. The location, size, and hydrologic characteristics of all streams, rivers, lakes, and adjacent shore regions which influence or may influence the site or facilities under severe hydrologic conditions, should be described. Topographic maps of the area and the site should be provided to give a clear understanding of these features. A map of the site area should indicate any proposed change to the natural drainage features. If the site is vulnerable to river flooding, any river control structures upstream or downstream of the site should be identified. The SAR should identify the sources of the hydrologic information, the types of data collected, and the methods and frequency of collection. The SAR should also list the structures important to safety, including their exterior accesses, and equipment and systems which may be affected by hydrologic features. The SAR should note any surface waters which could potentially be affected by normal or accidental effluents from the site. A listing of any population groups which use such surface waters as a potable water supply should be provided, as well as the size of these population groups, location, and water-use rates.
Design	2.4.4.2	**SITE CHARACTERISTICS** **Surface Hydrology** Floods		The SAR should adequately support any claim that the proposed site is flood-dry, that is, with structures important to safety so high above potential sources of flooding that safety is obvious or can be documented with little analysis, as indicated in American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.8-1981. The SAR should adequately support any claim that the proposed site is flood-dry, that is, with structures important to safety so high above potential sources of flooding that safety is obvious or can be documented with little analysis, as indicated in American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.8-1981.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.4.3	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers	72.98 (c)(2)	The SAR must consider the PMF on adjacent streams and rivers in its detailed flood analysis. If the SAR did not follow the approach for assessing PMFs in ANSI/ANS 2.8-1981, then it should describe the alternative approach used. The steps taken to derive the probable maximum precipitation (PMP) over the applicable drainage area, the precipitation losses, the amount of runoff, and the PMF should be shown. Drainage basins should be identified on a topographic map. The estimated discharge hydrograph for the PMF at the site and, if applicable, a similar hydrograph without the effects of an upstream reservoir should be included. The conversion of the PMF peak discharge into water elevation at the site should be described. Wind-wave activity which could coincide with the PMF should be discussed. Finally, the locations and associated water levels for which PMF determinations have been made should be summarized.
Design	2.4.4.4	**SITE CHARACTERISTICS** **Surface Hydrology** Potential Dam Failures (Seismically Induced)	72.98 (c)(2)	If potential dam failures are necessary to identify flood design bases, then the SAR should discuss the effects of potential seismically induced dam failures (both upstream and downstream) on the water levels of streams and rivers. Descriptions of existing or proposed dams and reservoirs which could influence conditions at the site should be provided and include seismic design criteria for dams. The potential dam failure modes which lead to the most critical consequences for the site (flood or low reservoir level) should be described. Domino-type dam failures from floodwaves should be considered where applicable. Finally, the reliability of the water level estimate should be addressed.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.4.5	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding	72.98 (c)(2)	If the site is at risk of inundation from surge or seiche flooding, these hazards should be described. Water bodies which could impact the site should be described, and the surge and seiche history of the site should be provided. The frequency and magnitudes of potential causes of surges, such as hurricanes, wind storms, squall lines, and other mechanisms should be described. A graph of the calculated maximum surge hydrograph should be provided. The potentially coincident wind-generated waves and the possibility of wave oscillation at natural frequencies should be described. Estimates of potential wave runup, erosion, and sedimentation, and any site facilities designed to guard against these processes, should be described.
Design	2.4.4.6	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding	72.98 (c)(2)	If the site abuts a coastal area, the hazards posed by tsunami should be analyzed. The history of tsunami in the region--be it recorded, translated, or inferred from the geologic record--should be analyzed. The analysis should include all potential tsunami generators, such as specific faults, fault zones, volcanoes, and potential landslide areas. The maximum tsunami height from these causes should be estimated at the source, in deep water offshore from the site, and onshore. A probable maximum tsunami should be derived from these analyses. Near-shore routing, wave breaking, bore formation, and resonance effects of this tsunami should be discussed. Any structures designed to protect against tsunami flooding should be described.
Design	2.4.4.7	**SITE CHARACTERISTICS** **Surface Hydrology** Ice Flooding	72.98 (c)(2)	If the site is not subject to flooding caused by ice jams, a brief statement of explanation should be provided. If the site is subject to ice-jam flooding, an analysis of this hazard should be provided. The history of ice jam formation in the region and the location of ice-generating mechanisms relative to the facility should be described. Any structures designed to protect against flooding from ice jams should also be described.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.4.8	**SITE CHARACTERISTICS** **Surface Hydrology** Flood Protection Requirements	72.98 (c)(2)	The static and dynamic consequences of all types of flooding on each storage structure and component important to safety should be described if the previous flooding analyses indicate that the structure or component is subject to flooding. The design bases required to ensure that all structures and components can survive all design flood conditions should be included.
Design	2.4.4.9	**SITE CHARACTERISTICS** **Surface Hydrology** Environmental Acceptance of Effluents	72.98 (c)(2)	The ability of the surface water and ground water environment to disperse, dilute, or concentrate normal and inadvertent liquid releases of radioactive effluents for the full range of anticipated operating conditions, including accident scenarios leading to worst-case releases, should be described. All potential surface water and ground water pathways by which radionuclides could reach existing and potential water users should be identified. Any potential for water recirculation, sediment concentration, or hydraulic short-circuiting of cooling ponds should be assessed in anticipation of normal or accidental releases of radionuclides.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.5	SITE CHARACTERISTICS Subsurface Hydrology	72.24, 72.98, and 72.122 (b)(4)	10 CFR 72.122 requires that measures be taken to preclude the transport of radioactive materials to the environment through subsurface characteristics. The SAR should contain adequate information for an independent review of all subsurface hydrology-related design bases and compliance with dose radiological exposure standards. If the site is located over an aquifer which is a source of well water, the groundwater aquifer(s) beneath the site, the associated hydrologic units, and their recharge and discharge areas should be described. The results of a survey of groundwater users, well locations, source aquifers, water uses, static water levels, pumping rates, and drawdown should be provided. A water table contour map showing surface water bodies, recharge and discharge areas, and locations of monitoring wells to detect leakage from storage structures should also be provided. Information on monitoring wells should include: wellhead elevation, screened interval, installation method, and representative hydrochemical analyses. An analysis bounding the potential groundwater contamination from site operations should be provided. A graph of time versus radionuclide concentration at the closest existing or potential downgradient well should be included.
Design	2.4.6	SITE CHARACTERISTICS Geology and Seismology	72.24, 72.40, 72.90, 72.92, 72.98, 72.102, 72.103 (f), and 72.122	10 CFR 72.102 (103) requires that the SAR describe the geological and seismological setting of the site and surrounding region. Conditions which may influence the design and operation of the facility should be identified, and sources of all information should be stated. Enough information for an independent evaluation of the potential ground vibrations and the seismic and fault displacement hazards at the site area should be provided. Design bases for ground vibration, surface faulting, subsurface material stability, and slope stability should also be provided.

Category	NUREG 1567 Section	NUREG 1567 <u>SECTION TITLE</u> Subsection Title	10 CFR Section	Criteria Description
Design	2.4.6.1	<u>**SITE CHARACTERISTICS**</u> **Geology and Seismology** Basic Geologic and Seismic Information	72.102, 72.103 (f)	Basic geologic and seismic characteristics of the site and vicinity should be provided. The geologic history of the area should describe its lithologic, stratigraphic, and structural conditions. A large-scale geologic map of the site area showing the surface geology and the location of major facilities should be provided. A stratigraphic column and cross-sections should also be provided. Planar and linear features of structural significance such as folds, faults, synclines, anticlines, basins, and domes should be identified on a geologic map showing bedrock surface contours. A description of the site geomorphology should include areas of potential landsliding or subsidence, and a topographic map showing geomorphic features and principal site facilities should be provided. The results of pertinent geophysical investigations in the area, such as seismic refraction, seismic reflection, aeromagnetic, or geoelectrical surveys, should also be provided. The SAR should evaluate geologic features from an engineering geology perspective. Detailed static and dynamic engineering properties of soil and rock underlying the site should be provided, with the results integrated to provide a comprehensive understanding of the surface and subsurface conditions. A small-scale map should show major features of the installation and the locations of all borings, trenches, and excavations. Small-scale cross-sections should demonstrate relationships between major foundations and subsurface materials, structures, and the water table. Finally, any physical evidence concerning the behavior of surficial site materials during previous earthquakes should be presented.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.6.2	**SITE CHARACTERISTICS** **Geology and Seismology** Ground Vibration	72.102, 72.103 (f)	The design basis ground vibration and a rationale for its selection should be presented and explained. The rationale should list historical earthquakes which could have affected the site, their dates, epicenter locations, and magnitudes. This listing of events is not constrained by distance and may include entries for distant structures, such as the New Madrid fault system. All faults and epicenters should be displayed on maps of appropriate scales. The fault map should include all potentially significant faults or parts of faults within 161 km (100 mi), regardless of capability. All capable faults (as defined in 10 CFR Part 100, Appendix A) which may be of significance in establishing the design basis ground vibration for the site should be identified and adequately described. The maximum ground vibration at the site should be derived from the potential earthquakes from all capable faults and from floating earthquakes (FEs, those not associated with a previously identified structure).
Design	2.4.6.3	**SITE CHARACTERISTICS** **Geology and Seismology** Surface Faulting	72.102, 72.103 (f)	Surface faulting at the site and underlying tectonic structures which have caused or might cause faulting should be described. The capability of any mapped faults 300 m (1000 ft) or longer within 8 km (5 mi) of the site should be described. Those judged capable should be described in detail, with special attention to their displacement history and their relationship to any regional tectonic structures.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Design	2.4.6.4	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials	72.102, 72.103 (f)	The stability of the rock (defined as having a shear wave velocity of at least 1166 m/s [3500 ft/s]) and soil beneath the foundations of the facility structures while subjected to the design basis ground vibration should be described. The geologic features which could affect the foundations, such as areas of potential uplift or collapse, or zones of deformation, alteration, structural weakness, or irregular weathering, should be described. The static and dynamic engineering properties of the materials underlying the site, as well as the physical properties of foundation materials should be described. A plot plan showing the locations of all borings, trenches, seismic lines, piezometers, geologic cross-sections, and excavations, with all installation structures superimposed, should be provided. Plans and profiles showing the extent of excavations and backfill, as well as compaction criteria, should be provided. The water table history and anticipated groundwater conditions beneath the site during facility construction and operation should be described. Analyses of soil and rock responses to dynamic loading should be provided, and potential liquefaction beneath the site should be discussed. Criteria, references, or methods of design used, along with safety factors, should be discussed.
Design	2.4.6.5	**SITE CHARACTERISTICS** **Geology and Seismology** Slope Stability	72.102, 72.103 (f)	The stability of all natural and man-made slopes, both cut and fill, the failure of which could adversely affect the site, should be described. Cross-sections of the slopes and a summary of the static and dynamic properties of embankment and foundation soil and rock underlying the slopes, should be provided. The design criteria and analyses used to determine slope stability should be described.

A-2. DETAILED REQUIREMENTS OF NUREG-1567

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	3.4	**OPERATION SYSTEMS** **Acceptance Criteria**	72.24, 72.40, 72.44, 72.104, 72.122, 72.124, 72.126, 72.128, 72.150, 72.166	This section identifies the acceptance criteria used for the operation systems review. Information on systems may be fully described functionally at the Safety Analysis Report (SAR) chapters oriented on physical design and specific safety-related functions (such as installation and structural design, thermal, criticality, and confinement), since detailed information need only be included once in the SAR. The primary purpose of this chapter is a review of the functional description of the systems operations, flowsheets showing sequences of operations and controls, and drawings showing proper functioning of each system. Additional description of the information that should be in the SAR is provided for each of the review areas.
Licensing, Operation	3.4.1	**OPERATION SYSTEMS** **Operation Description**	72.24, 72.40, 72.44, 72.104, 72.122, 72.124, 72.126, 72.128, 72.150	Operation description relates to the overall storage functions and operation of the installation. The applicant should provide an overview of operations. Acceptable criteria for operation system descriptions are given in NUREG-1536, Chapter 8, Section IV, items 1 through 6.
Licensing, Operation	3.4.2	**OPERATION SYSTEMS** **Spent Fuel and High-Level Waste Handling Systems**	72.24, 72.104, 72.124, 72.128, 72.150, 72.166	The regulatory requirements given in 10 CFR 72.124, 10 CFR 72.128, 10 CFR 72.150, and 10 CFR 72.166 address the information to be included in a license application. The SAR should include information as described in Regulatory Guide 3.48 Section 5.2 on spent fuel (and high-level waste if for an MRS) handling systems. The descriptions of the spent fuel or high-level waste handling systems must be clear. The functions of transfer from transportation vehicles, receipt inspection, and initial decontamination should be addressed if the operations are performed independently of a 10 CFR 50 license review. The transfer facility and its use should be described, including its use during the stages of operation of the ISFSI. Spent fuel and highlevel waste handling systems in a pool facility used for wet transfer is addressed in a following section.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	3.4.3	**OPERATION SYSTEMS** **Other Operating Systems**	72.24, 72.44, 72.104, 72.122, 72.124, 72.126, 72.128, 72.150, 72.166	The scope of this section is taken to be all operating systems important to safety that are not covered in Sections 3.4.1 (Operation Description) and 3.4.2 (Spent Fuel and High-Level Waste Handling Systems) except that instrumentation and controls are covered in 3.4.4 and analytical sampling is covered in 3.4.6. “Other operating systems” and “auxiliary systems” that are important to safety should be as described in Regulatory Guide 3.48 Sections 4.3 and 5.3 and noted in the narrative descriptions or flowcharts describing the operation of the ISFSI. 10 CFR 72.122 requires that the SAR include clear descriptions of the systems and system equipment and controls used to assure safety. These items must be consistent with other parts of the SAR. Examples of “other operating systems” that may be classified as important to safety include ventilation and off-gas systems, electrical systems, air supply systems, steam supply and distribution systems, water supply systems, fire protection systems, air sampling systems, decontamination systems, and systems related to chemical hazards.
Licensing, Operation	3.4.4	**OPERATION SYSTEMS** **Operation Support Systems**	72.24, 72.44, 72.122(i)	10 CFR 72.122 requires that the SAR include information on operation support systems, primarily instrumentation and control (I&C) systems and component spares or alternative equipment. These items should be as described in Regulatory Guide 3.48 Section 5.4. This information should include an analysis or other acceptable basis for determination that operation support systems important to safety remain operational under accident-level conditions. The SAR should include clear descriptions of the operation support systems and descriptions of equipment and controls used to assure safety, which are consistent with other parts of the SAR.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	3.4.5	**OPERATION SYSTEMS** **Control Room and Control Area**	72.122 (j)	10 CFR 72.122 requires that the SAR include a discussion of how a control room and control room areas permit the installation to operate safely under normal, off-normal, and accident conditions. The SAR should include clear descriptions of the control room and control area. The NRC has accepted omission of a control room for ISFSI operations that have not involved control of operations within a pool or use of a powered cooling system for material in storage. A control room and redundancy for control of functions important to safety in a separate control area is acceptable for ISFSI with pool facilities.
Licensing, Operation	3.4.6	**OPERATION SYSTEMS** **Analytical Sampling**	72.24, 72.44, 72.122	The SAR should include a discussion of the provisions for obtaining samples for analysis necessary to ensure that the ISFSI is operating within prescribed limits. The SAR should include a description of the facilities and equipment available to perform the required tests.
Licensing, Operation	3.4.7	**OPERATION SYSTEMS** **Shipping Cask Repair and Maintenance**	72.128 (a)	The SAR should contain a description of the shipping cask repair and maintenance facilities. The operation of these facilities, including provision for contamination control and occupational exposure minimization, should also be included. Note that the ownership, maintenance, and use of a shipping cask for shipping nuclear material by an ISFSI or MRS licensee is governed by the requirements of 10 CFR 71 only.
Licensing, Operation	3.4.8	**OPERATION SYSTEMS** **Pool and Pool Facility Systems**	72.24, 72.44, 72.104, 72.122, 72.124, 72.126, 72.128, 72.150	For ISFSI or MRS with a pool, the pool facility and the associated equipment constitute the principal capability for handling the subject radioactive material outside its storage confinement barrier or with that barrier open. The SAR should include clear descriptions of the pool and pool facility systems and descriptions of pool facility equipment and controls used to assure safety, which are consistent with other parts of the SAR. Section 9.1.2 of NUREG-0800 presents pool and pool facility systems requirements for a 10 CFR 50 license and should be used as guidance in the design of a 10 CFR 72 facility. Because a pool facility used only for wet transfer presents unique requirements, specific criteria are not provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4	**SSC AND DESIGN CRITERIA EVALUATION** **Acceptance Criteria**	72.2, 72.3, 72.6, 72.24, 72.102, 72.104, 72.106, 72.120, 72.122, 72,124, 72.128, 72.130, 72.144, 72.182, 72.236	This section identifies the acceptance criteria used for the various review areas. The acceptance criteria are based on regulatory requirements, Regulatory Guides, and staff judgments.
Licensing, Design	4.4.1	**SSC AND DESIGN CRITERIA EVALUATION** **Materials To Be Stored**	72.2, 72.3, 72.6, 72.120	No text provided.
Licensing, Design	4.4.1.1	**SSC AND DESIGN CRITERIA EVALUATION** **Materials To Be Stored** Spent Fuel (Part 1 of 3)	72.2(a)(1) and (a)(2), 72.3, 72.6(b), 72.120(b)	The regulatory requirements given in 10 CFR 72.2 (a)(1) and (a)(2) identify power reactor spent fuel as material to be stored. 10 CFR 72.6 (b) states that the general license to store spent fuel or high-level radioactive waste may be issued without regard to quantity. 10 CFR 72.120 (b) discusses the acceptable form, i.e., solid fuel or high-level radioactive waste. The applicant must provide information on the spent fuel to be stored including, but not limited to, reactor type (e.g., Boiling Water Reactor, Pressurized Water Reactor, etc.), fuel manufacturer and model designation and number, fuel physical characteristics, fuel cladding material, thermal characteristics, radionuclide characteristics (e.g., gamma and neutron source terms), and history and census, including burnup, initial enrichment, and cooling time. The applicant must also provide information on the ranges of parameters of the spent fuel to be stored. Bounding parameters for further fuel storage should be listed. Continued in Part 2 of 3.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4.1.1	**SSC AND DESIGN CRITERIA EVALUATION** **Materials To Be Stored** Spent Fuel (Part 2 of 3)	72.2(a)(1) and (a)(2), 72.3, 72.6(b), 72.120(b)	In the SAR, the applicant must specify if damaged fuel is to be stored at the ISFSI. Damaged fuel should be canned for storage and transportation. The purpose of canning is to confine gross fuel particles to a known, subcritical volume during off-normal and accident conditions, and to facilitate handling and retrievability. As proof that the fuel is undamaged, the applicant, at a minimum, should review the fuel records and verify that the fuel was undamaged. Also, the applicant should specify that prior to loading, the fuel assemblies will receive an external visual examination for any obvious damage. For fuel assemblies where reactor records are not available, the applicant should provide alternate information which provides reasonable assurance that the fuel is undamaged or that damaged fuel loaded in a storage or transportation cask is canned in addition to the external visual examination for any obvious damage. Rod cluster control assemblies, burnable poison (rod) assemblies, thimble plugging assemblies, and primary and secondary source assemblies are materials associated with the storage of spent fuel assemblies. Title 10, Code of Federal Regulations (10 CFR), Section 72.3, “Definitions,” states, “...Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.” The applicant should define the range and types of spent fuel or other radioactive materials that the DCSS [dry cask storage system] is designed to store. For DCSSs that will be used to store activated components associated with a spent fuel assembly, the applicant should specify the types and amounts of radionuclides, heat generation, and the relevant source strengths and radiation energy spectra permitted for storage in the DCSS. Continued in Part 3 of 3.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4.1.1	**SSC AND DESIGN CRITERIA EVALUATION** **Materials To Be Stored** Spent Fuel (Part 3 of 3)	72.2(a)(1) and (a)(2), 72.3, 72.6(b), 72.120(b)	Specifically, the applicant should describe: • The design bases source term (radiological and thermal components). The source term should be based on a saturation value for activation of cobalt impurities or on cobalt activation from a specified maximum burn-up and minimum cool time. The applicant should describe other activation products, as appropriate. • The effects of gas generation must be considered in the design pressure for the cask, including (1) the release of gas from additional components, and (2) the volume occupied by additional components on the cask internal pressure. • Additional weight and length of the proposed material must be considered in the structural and stability analyses. • The thermal analysis must consider (1) the added heat from these components, and (2) the effects of heat transfer within and to/from the fuel assembly by the addition or absence of these components. This would ultimately affect the maximum predicted cladding temperature. • In terms of a criticality evaluation, absent direct physical measurements, the applicant should not take credit for any negative reactivity from residual neutron absorbing material remaining in the control components. A bounding analysis would assume that no control components are present. Credit for water displacement may be taken provided adequate structural integrity and placement under accident conditions is demonstrated. Also, the applicant may need to consider the effects of displacing borated water, if applicable.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4.1.2	**SSC AND DESIGN CRITERIA EVALUATION** **Materials To Be Stored** High-Level Radioactive Waste	72.2, 72.3, 72.6, 72.120 (b) and 72.124	The regulatory requirements given in 10 CFR 72.3 define high-level radioactive waste and 10 CFR 72.120 (b) establish that the spent fuel or solid high-level waste are the acceptable waste forms. Liquid high-level radioactive waste is not acceptable for storage. Furthermore, if a pool type facility is proposed, the solidified waste form shall be a durable solid with demonstrable leach resistance. The applicant must provide information on the waste form, proposed storage package, characteristics of any encapsulation material, radionuclide characteristics, heat generation rate, and history and census. The Safety Analysis Report (SAR) must also include both the ranges of parameters of the known material to be stored and the bounding parameters of any additional materials that may be stored.
Licensing, Design. Quality Assurance	4.4.2	**SSC AND DESIGN CRITERIA EVALUATION** **Classification of Structures, Systems, and Components**	72.3, 72.24 (n), 72.144 (a) and (c)	The applicant must identify all SSCs important to safety and provide a rationale for the identification. SSCs are classified into two broad categories: important to safety or not. The NRC review involves both categories; however, SSCs important to safety are reviewed in greater depth. Acceptance criteria for classification of SSCs important to safety are discussed in 10 CFR 72.3, 10 CFR 72.24 (n), and 10 CFR 72.144 (a) and (c). The chapter on Installation Design and Structural Review discusses five areas of review which generally include SSCs identified as important to safety. These areas of review are: confinement structures, systems, and components; pool confinement facilities; and reinforced concrete structures; other SSCs important to safety; and other SSCs subject to NRC approval. Similarly the chapters on Thermal Evaluation, Radiation Shielding Evaluation, Criticality Evaluation, Confinement Evaluation, Waste Confinement, Radiation Protection, and Decommissioning have review areas that must be considered in identifying SSCs important to safety.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design. Quality Assurance	4.4.3	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety**	72.122, 72,124, 72.126, 72.128, 72.130, 72.144, 72.182, 72.236	No text provided.
Licensing, Design. Quality Assurance	4.4.3.1	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** General (Part 1 of 2)	72.24 (c)(1), (c)(2), and (c)(4), 72.106 (a) and (c), 72.120 (a) and (b), 72.122 (a) through (l), 72.144, and 72.182 (a) and (b)	The regulatory requirements for design bases and general design criteria are given in 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.106 (a) and (c); 10 CFR 72.120 (a) and (b); 10 CFR 72.122 (a) through (l); 10 CFR 72.144; and 10 CFR 72.182 (a), and (b). The applicant must identify design criteria and design bases for all SSCs determined to be important to safety. The basic design criteria for SSCs which are important to safety shall: maintain subcriticality, maintain confinement, ensure radiation rates and doses for workers and public do not exceed acceptable levels and remain as low as is reasonably achievable (ALARA), maintain retrievability, and provide for heat removal (as necessary to meet the above criteria). Acceptance criteria for the specific design criteria are discussed in detail in each of the chapters.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design. Quality Assurance	4.4.3.1	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** General (Part 2 of 2)	72.24 (c)(1), (c)(2), and (c)(4), 72.106 (a) and (c), 72.120 (a) and (b), 72.122 (a) through (l), 72.144, and 72.182 (a) and (b)	The principal design criteria and bases should include the following items: • Normal design conditions and parameters, including site-specific environmental conditions such as ambient temperature, humidity, and insolation; and operational parameters such as maximum load capacity of cranes and handling equipment; and maximum dimensions of the casks or other critical equipment to be handled. • Off-normal design conditions and parameters, including site-specific environmental conditions such as ambient temperatures and insolation, and operational parameters which do not approach accident conditions. • Accident design events, including site-specific environmental conditions such as tornado wind velocities, tornado pressure drop, maximum wind velocities, design basis earthquake, peak explosive over pressure, peak flood elevation, and accident design events such as maximum dose rates associated with hypothetical accidents including a cask drop or loss of pool coolant. Codes and standards and other detailed criteria applicable for ISFSI and MRS SSCs important to safety are presented or referenced in the Standard Review Plan (SRP) chapters addressing structural evaluation, thermal evaluation, shielding evaluation, nuclear criticality safety, confinement, waste management and decommissioning. The FSRP chapter on site evaluation addresses review of site characteristics that must be included in design criteria and bases for natural phenomena.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design. Quality Assurance	4.4.3.2	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** Structural	72.24 (c)(1), (c)(2), (c)(3), (c)(4) and n, 72.102 (a), (b), (c), (d), (e), and (f), 72.120 (a) and (b), 72.122 (a), (b)(1), (b)(2) and b(3), (c), (d), (e), and (f), 72.120 (a) and (b), 72.122 (a), (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), and (k)	The regulatory requirements for structural aspects of SSCs important to safety are given in 10 CFR 72.24 (c)(1), (c)(2), (c)(3), and (n); 10 CFR 72.102 (a), (b), (c), (d), (e), and (f); 10 CFR 72.120 (a) and (b); and 10 CFR 72.122 (a), (b)(1), (b)(2) and (b)(3), (c), (d), (f), (g), (h), (i), (j), and (k). The applicant must present the structural design criteria and design bases for the proposed ISFSI or MRS. The structural design criteria and bases presented by the applicant for an ISFSI or MRS must address the design magnitudes of loads and limits derived from site characteristics and analyses of normal, off-normal, and accident-level conditions. The design bases presented by the applicant must include dead load, live load, lateral rail pressure, thermal loads, wind loads, accident loads, earthquake loads, and flood loads. Design bases guidance for tornado protection are given in Regulatory Guides 1.76, “Design Basis Tornado for Nuclear Power Plants,” and 1.117, “Tornado Design Classification.” Guidance for flood protection is given in Regulatory Guides 1.59, “Design Basis Floods for Nuclear Power Plants,” and 1.102, “Flood Protection for Nuclear Power Plants.” Guidance for protection against seismic events is given in Regulatory Guides 1.29, “Seismic Design Classification,” 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” 1.92, “Combing Modal Responses and Spatial Components in Seismic ResponseAnalysis,” and 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.”

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4.3.3	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** Thermal	72.122 (a), (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), and (i), 72.128 (a)(4)	The regulatory requirements relating to design bases and design criteria for thermal considerations are given in 10 CFR 72.122 (a), (b)(1), (b)(2) and (b)(3), (c), (d), (f), (g), (h), and (i); and 10 CFR 72.128 (a)(4). The applicant must identify thermal design criteria and bases. These criteria and bases must recognize the site temperature range and the specific materials used in ISFSI or MRS components. Another aspect of thermal design criteria and design bases is fire protection. Guidance for fire protection is given in Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants."
Licensing, Design	4.4.3.4	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safe**ty Shielding and Confinement	72. 24 (c)(1), (c)(2) and (c)(4), 72.104 (a), (b), and (c), 72.106 (a), (b), and (c); 72.122 (a), (b), (c), (d), (e), (f), (g), (h), and (i), 72.126 (a), (b), (c) and (d), 72.128 (a) and (b)	The regulatory requirements for shielding and confinement are given in 10 CFR 72. 24 (c)(1), (c)(2) and (c)(4); 10 CFR 72.104 (a), (b), and (c); 10 CFR 72.106 (a), (b), and (c); 10 CFR 72.122 (a), (b), (c), (d), (e), (f), (g), (h), and (i); 10 CFR 72.126 (a), (b), (c) and (d); and 10 CFR 72.128 (a) and (b). The applicant must identify shielding and confinement design criteria and design bases. These criteria and bases should discuss any proposed compliance with Regulatory Guides 8.5, "Criticality and Other Interior Evacuation Signals;" 8.25, "Air Sampling in the Workplace;" 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses;" and 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4.3.5	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** Criticality	72.124 (a), (b), and (c)	The regulatory design bases and design criteria for criticality safety are given in 10 CFR 72.124 (a), (b), and (c). The application must identify nuclear criticality safety design criteria and design bases. These criteria and bases should discuss any proposed compliance with Regulatory Guides.
Licensing, Design, Decom-missioning	4.4.3.6	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** Decommissioning	72.130	10 CFR 72.130 outlines the regulatory requirements for decommissioning considerations. The applicant must identify any decommissioning design criteria and design bases. The application must also discuss compliance with any relevant Regulatory Guides. Planning for decommissioning and design guidance for facilitating decommissioning are addressed in the FSRP chapter on decommissioning.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	4.4.3.7	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for SSCs Important to Safety** Retrieval	72.122 (a), (b)(1), (b)(2), and (b)(3), (c), (f), and (h), 72.122 (l)	General regulatory requirements for retrieval capability are given in 10 CFR 72.122 (a), (b)(1), (b)(2), and (b)(3), (c), (f), and (h). Retrievability is specifically outlined in 10 CFR 72.122 (l). The applicant must include design criteria and design bases for retrieval. The design criteria and bases for the ISFSI or MRS storage system must recognize the need for facilities, equipment, and procedures for the removal of spent fuel or solidified high-level radioactive waste from storage systems, and the transfer of this material into another storage system or a transportation cask. The design developed in compliance with the criteria must be able to retrieve spent fuel or the solidified high-level waste following normal and off-normal design conditions. Specific retrieval facilities, equipment, and procedures for post accident conditions are not required to be described in the SAR because of the wide variety of possible post-accident conditions that may occur. The design must accommodate the retrieval of spent fuel or solid HLW following design basis accidents. The design and procedures for retrieval must be such that the operations can be conducted in compliance with the requirements of 10 CFR Part 20.
Licensing, Design	4.4.4	**SSC AND DESIGN CRITERIA EVALUATION** **Design Criteria for Other SSCs**	72.24 (a), (b), (c), (d), (e), (f), (g), (h), and (l), Appro-priate Subparts of E & F	Design criteria and bases for other SSCs not important to safety should meet the general regulatory requirements as given in 10 CFR 72.24 (a), (b), (c), (d), (e), (f), (g), (h), (l), and the appropriate requirements as given in 10 CFR 72, Subparts E and F. The applicant must identify design criteria and bases for SSCs not important to safety. These design criteria and bases for ISFSI and MRS SSCs that are not important to safety may be adequately defined by statements in the SAR identifying the design codes and standards to be met in design and construction. Greater definition is typically appropriate for SSCs that interface with SSCs important to safety.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4	**INSTALLATION AND STRUCTURAL Acceptance Criteria** (Part 1 of 2)	72.24, 72.40, 72.82, 72.102, 72.106, 72.120, 72.122, 72.128, 72.236	This section identifies the acceptance criteria used for the structural evaluation. Acceptability of the design of the structures, systems, and components as described in the SAR is based on compliance with requirements and Regulatory Guides determined by independent calculations and staff judgments. The design of the SSCs are acceptable if the integrated design meets the general and specific criteria discussed below. The license approval process for ISFSI and MRS is a one-step licensing process rather than a two-step process as exemplified by 10 CFR Part 50 for a reactor license. Thus, the evaluation of the SAR and the supporting materials for an ISFSI license is the sole occasion in the design and construction sequence that the design and proposed construction are comprehensively reviewed by the NRC staff. The result is that the depth of information required for individual SSCs important to safety is greater for ISFSI and MRS than would be required for similar SSCs in the application for a construction permit under 10 CFR 50.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4	**INSTALLATION AND STRUCTURAL EVALUATION Acceptance Criteria** (Part 2 of 2)	72.24, 72.40, 72.82, 72.102, 72.106, 72.120, 72.122, 72.128, 72.236	The confinement systems, including pool facilities, reinforced concrete structures, and other SSCs, which are important to safety or subject to NRC approval, must to have sufficient structural capability to withstand the worst-case loads under accident conditions and natural phenomena events. This may be verified by the reviewer of the SAR, first by verifying acceptable design criteria and then by verifying acceptable analyses, which ensure that the structures preclude: • unacceptable risk of criticality • unacceptable release of radioactive materials • unacceptable radiation levels • impairment of ready retrievability of stored material Provided that a certified cask system has not been modified, the use of a certified cask design can be used to satisfy a part of the requirements for the facility license application by reference. Site facilities and infrastructure of concern to the NRC are to have the descriptions, design criteria, and safety analyses as appropriate to safety reviewed. These could include the pool and pool facility SSCs, the waste facilities, space for NRC use, and other elements of the site physical infrastructure.
Licensing, Design	5.4.1	**INSTALLATION AND STRUCTURAL EVALUATION Confinement Structures, Systems, and Components**	72.24, 72.40, 72.82, 72.102, 72.106, 72.120, 72.122, 72.128, 72.236	No text provided.

Category	NUREG 1567 Section	NUREG 1567 <u>SECTION TITLE</u> Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.1.1	<u>**INSTALLATION AND STRUCTURAL EVALUATION**</u> **Confinement Structures, Systems, and Components** Description of Confinement Structures	72.24 (a) and (b), 72.82 (c)(2), and 72.106 (a), (b), and (c)	10 CFR 72.24 (a) and (b), 10 CFR 72.82 (c)(2), and 10 CFR 72.106 (a), (b), and (c) outline the contents of the application, which include design descriptions in sufficient detail to support findings in the SER. For confinement SSCs the application must include text descriptions, drawings, figures, tables and specifications that would fully define the structural features of the confinement SSCs. For a site-specific ISFSI, the application may involve use of a cask certified under 10 CFR 72, Subpart L, including the SAR for the certified cask system by reference. Additional information relating to the cask should also be provided, including the applicant's evaluations that establish that site parameter limits are within the bounds of those established as limiting conditions as set forth in the Certificate of Compliance. If actual site parameters exceed the bounds of those assumed in the safety analysis submitted for the certified cask system or exceed specified conditions of compliance, then the SAR submitted with the application must fully address those areas affected by the variations. If the design of the proposed cask system is not identical to the certified cask system, the SAR shall include a full description of the cask system (drawings and construction or fabrication specifications), a description of all changes to the certified design, and analyses that show the proposed design satisfies the criteria for the proposed installation.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 1 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	The regulatory requirements given in 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.40 (a)(1); 10 CFR 72.120 (a), and (b); 10 CFR 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l); and 10 CFR 72.128 (a) and (b) identify acceptable design criteria. The NRC generally considers the design criteria identified below to be acceptable to meet the structural requirements of 10 CFR 72 for storage confinement casks. **General Structural Requirements** The confinement structures are to have sufficient structural capability so that every cross section of the structure can withstand the worst-case loads and successfully preclude the unacceptable risk of criticality, unacceptable release of radioactive materials to the environment, unacceptable radiation dose to the public or workers, and significant impairment of ready retrievability of the stored nuclear material. Confinement of radioactive material must be maintained under normal, off-normal, and accident conditions. These criteria do not require that all confinement systems and other structures important to safety survive all design basis accidents and extreme natural phenomena without any permanent deformation or other damage. Some load combination expressions for accident events, for structures important to safety, permit stress levels that exceed yield. These scenarios should be shown to be acceptable by computations, analyses, and/or tests acceptable to the NRC.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 2 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	Structures important to safety are not required to survive accident events and conditions to the extent that they remain suited for use for the life of the ISFSI or MRS without inspection, repair, or replacement. However, confinement structures are required to maintain confinement integrity under all accident conditions. The NRC does not accept breach of the storage confinement. If the life of structures important to safety may be degraded by design basis events, requirements and procedures for determination and correction of the degradation, or other acceptable remedial action must be provided. Spent fuel cladding must be protected against gross rupture caused by degradation resulting from normal, off-normal, or accident conditions. The cask and any racks for positioning stored fuel or waste material within the cask must not deform under credible loading conditions to the extent that the subcritical condition or the retrievability of the fuel would be jeopardized. The cask must be analyzed to show that it will not slide, tip over, or drop in its storage condition as a result of a credible natural phenomenon event, including tornado winds and tornado missiles, earthquakes, and floods. This criterion is to preclude damage to an entire array. A tip-over or drop is always to be assessed as a bounding condition during handling operations.

Category	NUREG 1567 Section	NUREG 1567 <u>SECTION TITLE</u> Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**<u>INSTALLATION AND STRUCTURAL EVALUATION</u>** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 3 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	Radiation shielding for the cask system, required for protection of the public or workers, must not degrade under normal or off-normal conditions. The shielding function may be acceptably degraded by a design basis event (e.g., loss of liquid neutron shielding resulting from a drop accident). However, the loss of function must be readily apparent. **Applicable Codes and Standards** The applicant must identify the design codes and standards intended for confinement structures. The structural design, fabrication, and testing of the confinement system must comply with an acceptable code or standard. Use of codes and standards that have been accepted by the NRC expedites the evaluation process. The alternative use of other codes and standards may require extensive NRC review and may delay the evaluation process. An accepted code for design, fabrication, and testing of steel confinement casks is Section III of the ASME Boiler and Pressure Vessel Code (ASME B&PVC). The NRC has accepted use of either Section NB or NC. The NRC has accepted use of Sections NF and NG of the ASME B&PVC, Section III, Division 1 for cask system components used within the confinement cask but not integrated with it. This includes the “basket” which is a structure used inside casks to restrain and position fuel assemblies. Other design codes or standards may be acceptable depending on their application.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 4 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	The NRC accepts use of Regulatory Guides 7.11, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m),” and 7.12, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1m) But Not Exceeding 12 Inches (0.3m),” as bases for determining the potential for brittle fracture. These Regulatory Guides also incorporate a portion of NUREG/CR 1815 by reference. The reviewer should be aware of those portions of NUREG/CR 1815 which are excluded by Regulatory Guides 7.11 and 7.12. The fatigue limits of the cask structural materials may be based on the provisions of the ASME B&PVC, Section III or the guidance provided in Regulatory Guide 7.6, “Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels.” Since casks are typically not subjected to cyclic loads, fatigue may not be a significant concern. **Cask Closure Welds After Fuel Loading** The following special considerations are generally accepted by the NRC for the dry storage canister top end closure welds which are made after the canister has been loaded with spent nuclear fuel assemblies. All other dry storage canister bottom end closure welds and shell welds should be designed, fabricated, examined, and tested to the requirements of the appropriate subsections of the ASME Section III Code.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 5 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	The top end closure welds are to be helium leak tested. No hydrostatic or pressure tests are required if a minimum margin of safety equal to or greater than 1.5 against design pressure was demonstrated by analysis. The closure weld joint may be either a full thickness penetration weld or a partial penetration groove weld. For a partial penetration groove weld, the maximum clearance between the closure plate and the enclosure shell should be small enough to ensure a good weld and should not exceed the clearance allowed in the weld procedure qualification. The minimum depth of the groove shall be equal to or larger than the enclosure shell thickness. The weld strength of the closure joint is based on the nominal weld area and the design stress intensity values for the weaker of the two materials jointed. However, the minimum ultimate tensile strength of the weld metal should equal or exceed the base metal strength to preclude weld metal failure. For dry storage canisters made from austenitic stainless steels Type 304, 304L, 304LN, 316, 316L, or 316LN, the top end closure weld may be examined by either the ultrasonic methods (UT) or progressive liquid penetrant (PT) examinations by the followin two options: 1) If UT is used, the UT acceptance criteria shall be the same as NB-5332 for preservice examination. 2) If PT is used, the examination shall be performed progressively on the root layer; the lesser of one half of the welded joint thickness, or ½ inch intervals thereafter; and the final surface. In addition, a stress reduction factor of 0.8 shall be applied to the weld strength of the joint.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 6 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	For dry storage canisters made from austenitic stainless steels other than the Type 304 or 316 materials listed above, the top end closure weld may be examined by PT as described above for Type 304 and 316, except that the thickness and number of intermediate layers to be examined shall be determined by a fracture mechanics assessment of the weld considering the specific geometry, material properties, and loadings. The maximum thickness of each weld pass deposit and PT layer shall not exceed the allowable critical flaw size for a 360 degree circumferential flaw. For dry storage canisters made from ferritic steels, the top end closure weld should be examined by UT and the following five criteria: 1) The critical flaw size and the critical design stress values shall be determined by the linear elastic fracture mechanics methodology specified in ASME Code, Section XI using the applicable service temperature and material properties. 2) The UT must be performed in accordance with pre-qualified procedures and methods. The UT examination methodology should be demonstrated to be reasonably accurate and consistently able to detect flaw sizes less than the critical flaw size determined by linear elastic fracture mechanics. 3) The UT examination must be performed by tested and certified operators.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	5.4.1.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Design Criteria for Confinement Structures (Part 7 of 7)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), and 72.128 (a) and (b)	4) The welding processes, weld inspection criteria, and weld personnel qualifications shall be in conformance with the ASME Code. The welding process and technique used should be evaluated to preclude hydrogen induced cracking. 5) As an alternative, progressive surface examinations, utilizing PT or magnetic particle examination (MT), are permitted only if unusual design and loading conditions exist. PT or MT must be performed after sufficiently small intervals to ensure that flaws equal to the critical flaw size will be detected. In addition, a stress reduction factor of 0.8 shall be used for the weld strength of the closure joint to account for imperfections or flaws potentially missed by progressive surface examinations. Critical flaw sizes for ferritic steels are generally small. Therefore, PT or MT must be performed on many layers of the weld and this alternative may become unacceptable, due to ALARA concerns. The weld design should provide a sufficient safety margin and should be approved by the NRC on a case-by-case basis.
Licensing, Design	5.4.1.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Material Properties (Part 1 of 2)	72.24 (c)(3)	Acceptable criteria for materials used in all structural components and systems are given in 10 CFR 72.24 (c)(3). The applicant must identify standards for materials and properties used in analyses. The information provided on materials must be consistent with the application of the accepted design criteria, codes, standards, and specifications selected for the storage cask system. For example, if the ASME B&PVC, Section III is used for the design criteria, the materials selected for the cask must be consistent with those allowed by the particular Section of the ASME B&PVC used for design. Acceptable requirements are ASME-adopted specifications given in ASME B&PVC, Section II, Part A "Ferrous Metals," Part B "Nonferrous Metals," Part C "Welding Rods, Electrodes, and Filler Metals," and Part D "Properties." NUREG-1536 provides additional guidance regarding the use of the ASME B&PVC requirements for material properties and specifications.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.1.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Material Properties (Part 2 of 2)	72.24 (c)(3)	Compatibility of materials and coatings to be used with the environments to be experienced must be established. This includes compatibility with fluids during loading and unloading operations that may occur on-site. Compatibility verification should specifically include potential reactions in the presence of liquids that may be used in conjunction with loading, unloading, decontamination, wet transfer operations, electrolytes, and water. Reactions may include chemical and galvanic actions, the possibility of production of explosive or toxic gas, and/or degradation. The SAR should include tables with material properties and allowable stresses and strains associated with temperature, as appropriate. Appropriate corrosion allowances should be established and used in the structural analyses. The potential for brittle fracture must be reviewed. The potential for brittle fracture of some components important to safety has resulted in conditions of use that preclude transfer operations during extremely low temperatures.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.1.4	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Component**s Structural Analysis	72.24 (d)(1), (d)(2), and (i), 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l)	Requirements for acceptable structural analysis are given in 10 CFR 72.24 (d)(1), (d)(2), and (i), as well as 10 CFR 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l). The applicant must provide analyses of load combinations for normal, off-normal, and accident conditions. The applicant must provide design analyses with adequate detail so that they may be readily audited to permit determination of the sources of expressions used, values of material properties, data from other supporting calculations and assumptions. ANSI N45.2.11 provides guidance for preparation of design analyses which is acceptable to the NRC. The design analysis for confinement SSCs shall identify all loading conditions and combinations of loadings. The analysis shall establish the design internal and external pressures, the design temperatures, and all the design mechanical loads. The analysis shall identify all combinations of design loads which can occur simultaneously. The specification shall establish service loadings (with appropriate service limits), which are discussed as normal, off-normal, and accident conditions in this SRP. For comparison purposes, normal service corresponds to Service Levels A and B of the ASME B&PVC, Section III; and accident service corresponds to Service Level D.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.1.5	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Buckling of Irradiated Fuel Under Bottom End Drop Conditions (Part 1 of 2)		Fuel rod buckling analyses under bottom end drop conditions have traditionally been performed to demonstrate integrity of the fuel following a cask drop accident. The analytical method described by Lawrence Livermore National Laboratory (LLNL) in report UCID-21246, is a simplified approach. The analytical method assumes that buckling occurs when a fuel rod segment between the bottom two spacer grids reaches the Euler buckling limit. The analytical method uses material properties for irradiated cladding, considers the weight of the cladding, but neglects the weight of fuel pellets. The NRC considers that, in addition to the weight of the cladding, end drop analyses should include the weight of fuel pellets and irradiated material properties. With the weight of the fuel pellets included, the analytical method of UCID-21246 yields highly conservative results. The analytical methods in UCID-21246 used to demonstrate fuel integrity following a cask drop accident yield a large margin to the point of actual failure. The calculated onset of buckling does not imply fuel or cladding failure. Where such analyses yield too conservative results, the applicant may use more realistic analyses of dynamic fuel behavior. If the cladding stress remains below yield strength, the fuel integrity is assured.
Licensing, Design	5.4.1.5	**INSTALLATION AND STRUCTURAL EVALUATION** **Confinement Structures, Systems, and Components** Buckling of Irradiated Fuel Under Bottom End Drop Conditions (Part 2 of 2)		If the applicant uses the analytical approach described in UCID-21246 for axial buckling to assess fuel integrity for the cask drop accident, the analysis should use the irradiated material properties and should include the weight of fuel pellets. Alternately, an analysis of fuel integrity which considers the dynamic nature of the drop accident and any restraints on fuel movement resulting from cask design is acceptable if it demonstrates that the cladding stress remains below yield. If a finite element analysis is performed, the analytical model may consider the entire fuel rod length with intermediate supports at each grid support (spacer). Irradiated material properties and weight of fuel pellets should be included in the analysis.

Category	NUREG 1567 Section	NUREG 1567 <u>SECTION TITLE</u> Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.2	**<u>INSTALLATION AND STRUCTURAL EVALUATION</u>** **Pool and Pool Confinement Facilities**	72.24, 72.40, 72.82, 72.102, 72.106, 72.120, 72.122, 72.128	The pool and pool confinement facilities provide a capability that may be essential to the conduct of ISFSI and MRS loading for storage and unloading functions and that may be needed for retrievability (see guidance in SRP Sections 3.4.8 and 4.4.3.7). The pool and pool confinement facilities are considered to include those systems important to safety that provide for wet transfer, loading, unloading, and temporary holding or long-term storage of spent fuel, high-level waste, and/or other radioactive materials associated with spent fuel or high-level waste storage. Other ISFSI or MRS equipment that may be used within and outside the pool facility, or that are used for lifting or transfer within the facility but are not installed cranes or conveyance systems, are addressed as “other SSCs important to safety” or “other SSCs.” The safety function of the pool and associated equipment is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The ISFSI and MRS pools and pool facilities should be designed as though they were to be in constant use for in-pool storage and wet transfer for the life of the ISFSI/MRS license. However, it is anticipated that the actual use of the pool facility may differ from the use of the spent fuel pool at a reactor facility. Therefore, limited or part-time use of the pool should be well-described in the SAR. The use status of the pool facility may have a major impact on the generation of radioactive and other waste. The design may also need to provide for conversion to standby mode or decontamination and decommissioning (D&D) while the rest of the ISFSI or MRS remains in use for dry storage.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.2.1	**INSTALLATION AND STRUCTURAL EVALUATION** **Pool and Pool Confinement Facilities** Description of Pool Facilities	72.24 (a) and (b), 72.40 (a)(3), 72.82 (c)(2), 72.106 (a), (b), and (c)	10 CFR 72.24(a) and (b), 10 CFR 72.40(a)(3), 10 CFR 72.82(c)(2), and 10 CFR 72.106(a), (b), and (c) address the descriptive information to be included in a license application. The application must describe pool facilities in sufficient detail to support a detailed review and evaluation. This would include text, descriptions, drawings, flow diagrams, figures, tables, and specifications to fully define the systems and features of the pool facilities. The NRC accepts use of existing pool and pool confinement facilities that are licensed under 10 CFR 50 for ISFSI or MRS, if concerns for possible sharing of SSCs between separately licensed facilities are satisfied (10 CFR 72.3 (included with definition of ISFSI), 72.24 (a), 72.40 (a)(3), and 72.122 (d)). The existing pool and pool confinement facilities may continue to be licensed under 10 CFR 50, or they may be re-licensed as elements of a wet storage and/or dry storage ISFSI.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.2.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Pool and Pool Confinement Facilitie**s Design Criteria (Part 1 of 2)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236 (b), (e), (f), (g), and (k)	The regulatory requirements given in 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.40 (a)(1); 10 CFR 72.120 (a), and (b); 10 CFR 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l); 10 CFR 72.128 (a) and (b); and 10 CFR 72.236 (b), (e), (f), (g), and (k) identify acceptable design criteria. Design criteria for important to safety facilities in 10 CFR 72 are fully applicable to pool and pool confinement facilities. Pool and pool confinement facilities should meet the criteria for structural integrity for similar facilities constructed at a power reactor which must comply with 10 CFR 50. These criteria are principally as stated in 10 CFR 50, Appendix A, General Design Criteria 61, “Fuel Storage and handling and radioactivity control.” Some portions of the General DesignCriteria 62, “Prevention of criticality in fuel storage and handling,” and General Design Criteria 63, “Monitoring fuel and waste storage” apply. Additionally, the General Design Criteria 2, 4, and 5 apply to the design of pool facilities. See NUREG-0800 Sections 9.1.2, Spent Fuel Storage and 9.1.3, Spent Fuel Pool Cooling and Cleanup System for specific acceptance criteria, which derives from 10 CFR Part 50, Appendix A. The intended usage of the pool and pool facilities may be used in the development of design requirements. Should the intended usage be long-term storage of spent nuclear fuel, the NRC accepts design of elements of the pool facility in accordance with ANSI/ANS 57.2. Should the intended usage be short term or primarily to facilitate wet transfer operations, the NRC accepts design of elements of the pool facility in accordance with ANSI/ANS 57.7. Regardless of whether ANSI/ANS 57.2 or 57.7 is used, it should be noted that 10 CFR 72.2 requires that spent fuel be aged for at least one year after discharge from the core.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.2.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Pool and Pool Confinement Facilities** Design Criteria (Part 2 of 2)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236 (b), (e), (f), (g), and (k)	The NRC accepts design of the pool liquid containment SSCs as required for Quality Group B (per Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants”) that are licensed under 10 CFR 50. This quality group requires design to not less than the requirements of ASME B&PVC, Section III, Class 2 (Division 1, Section NC). The NRC accepts design of ISFSI and MRS pool facility cooling and make-up water systems (as required) for Quality Group C. This quality group requires design to not less than the requirements of ASME B&PVC, Section III, Class 3 (Division 1, Section ND). The NRC accepts the guidance for reactor facility pools provided by Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” for ISFSI and MRS pool facilities. The principal criteria for pool facility design included in Regulatory Guide 1.13 are to: • prevent loss of water from the pool that would uncover the radioactive material • protect the radioactive material from mechanical damage • provide capability for limiting the potential offsite exposures in the event of a significant release of radioactivity from the subject materials.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.2.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Pool and Pool Confinement Facilities** Material Properties	72.24 (c)(3)	Acceptable criteria for materials used in all structural components and systems are given in 10 CFR 72.24 (c)(3). The applicant must identify materials and material properties to be used in the design. The information describing material properties must be consistent with the application of the accepted design criteria, codes, standards and specifications for the structural components of the pool facility. For example, if pool components forming the primary hydraulic containment or water level control, such as piping, pumps, valves, holding tanks, or filters are designed according to the ASME B&PVC Section III, then the materials selected must be consistent with those allowed by the particular Section of the design code. If the pool is housed in a reinforced concrete building designed according to ACI 349, then material properties should be consistent with the ACI 349 Code. If steel structures are to American Institute of Steel Constructions (AISC) standards, then the steel should have material properties from the Steel Construction Manual. In addition to the criteria given in 10 CFR 72.24 (c)(3), materials wetted by the pool water should be reviewed for compatibility and chemical stability. The selection of materials should be such that there are no potential mechanisms that will: (1) alter the location of any fixed neutron absorbers used in the design of the storage racks, and/or (2) cause physical distortion of the structures designed to retain the stored fuel assemblies in a fixed location.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.2.4	**INSTALLATION AND STRUCTURAL EVALUATION** **Pool and Pool Confinement Facilities** Structural Analysis	72.24 (d)(1), and (d)(2), (i), 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l)	Requirements for acceptable structural analysis are given in 10 CFR 72.24 (d)(1), and (d)(2), (i), as well as 10 CFR 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l). Design analyses should be prepared such that they may be readily audited to permit determination of the sources of expressions used, values of material properties, data from other supporting calculations, and assumptions. ANSI N45.2.11 provides guidance for preparation of design analyses which is acceptable to the NRC. The design specification for SSCs comprising the pool and the pool facilities shall identify all loading conditions and combinations of loadings. The specification shall establish the design internal and external pressures, the design temperatures, and all the design mechanical loads. The specification shall identify all combinations of design loads which can occur simultaneously. The specification shall establish service loadings (with appropriate service limits), which are discussed as normal, off-normal, and accident conditions in this SRP. ANSI/ANS 57.2 and ANSI/ANS 57.7 provide guidance for establishing design loads and structural analysis methods. Design codes are discussed.
Licensing, Design	5.4.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures**	72.24, 72.40, 72.82, 72.102, 72.106, 72.120, 72.122, 72.128	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.1	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Description of Concrete Structures	72.24 (a) and (b), 72.82 (c)(2), 72.106 (a), (b), and (c)	10 CFR 72.24 (a) and (b), 10 CFR 72.82(c)(2), and 10 CFR 72.106(a), (b), and (c) outline the contents of the application, which includes design descriptions in sufficient detail to support a detailed review and evaluation. Concrete structures may have roles in providing radiological shielding, forming ventilation passages, weather enclosures, structural supports, access denial, foundations, earth retention, anchorages, floors, walls, movable shields, and protection against natural phenomena and accidents. The applicant must fully describe any reinforced concrete structures. The description should include text descriptions, drawings, figures, tables, and specifications that would fully define the structural features of the reinforced concrete structures. Concrete structures may be cast in place, cast at the site, or cast elsewhere. Concrete structures may also be combinations of cast in place and precast sections that are integrated by bolting, welding, fitting, grouting, or placing additional concrete at the site. They may also include concrete that may be cast as part of a composite confinement cask with metallic liner. A metallic liner of a composite confinement cask, its closures, or its internal components should be designed as required for confinement SSCs (5.4.1).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Design Criteria (Part 1 of 5)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	The regulatory requirements given in 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.40 (a)(1); 10 CFR 72.120 (a), and (b); 10 CFR 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l); 10 CFR 72.128 (a) and (b); and 10 CFR 72.236(b), (e), (f), (g), and (k) identify acceptable design criteria. The structural design of the concrete structures shall withstand the effects of credible accident conditions and natural phenomena events without impairment of their capability to perform safety functions. The principal safety functions include maintaining subcriticality, containing radioactive material, providing radiation shielding for the public and workers, and maintaining retrievability of the stored material. The NRC has accepted special criteria for selection of components of reinforced concrete that may be exposed to elevated temperatures in normal or off-normal conditions. These criteria are given in the SRP Section 6.5.2.3. The acceptability of loads and stresses associated with thermal conditions is analyzed as part of the structural analysis. Concrete pads that support confinement casks in storage are not “pavements.” They should be designed and constructed as foundations under the applicable code (ACI 318 or ACI 349).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Design Criteria (Part 2 of 5)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	**Codes and Standards** ANSI/ANS 57.9 is generally applicable to ISFSI design and construction (with exceptions for confinement casks). Table 3-1 of NUREG-1536 includes extracts of ANSI/ANS 57.9 that are especially applicable to concrete structure design and construction. The table also includes corresponding evaluation guidance for review of the SAR documentation. The NRC has not accepted use of a set of criteria that has been derived by selection of criteria from more than one code. However, the NRC has accepted use of ACI 349 for design and material selection for concrete structures important to safety (but not as confinement cask), but has allowed the optional use of ACI 318 for construction, as described in this Section.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Design Criteria (Part 3 of 5)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	There are codes other than those discussed herein that may be applicable to the design and construction of the concrete elements of ISFSI and MRS. It is acceptable that such codes (e.g., the National Fire Protection Association (NFPA) Electric, Life Safety and Lightning Protection Codes) be included in the design by reference in the SAR documentation. Where designs of structures subject to approval are also covered by such other codes, the review should include evaluation of compliance with those codes. The NRC accepts use of ACI 349 for design, material selection and specification, and construction of all concrete structures that are not within the scope of ACI 359; except that additional or more stringent requirements given in ANSI/ANS 57.9, as incorporated by reference in NRC Regulatory Guide 3.60, “Design of an Independent Fuel Storage Installation (Dry Storage),” must also be met. Use of ACI 318 for construction of structures designed and with materials selected in accordance with ACI 349 is acceptable.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Design Criteria (Part 4 of 5)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	The following identifies the portions of ACI 349 and ASTM standards that are applicable to design (including material selection and metal embedments) that must be met by those applicants that choose to use ACI 318 for construction. The paragraph references are as in ACI 349-90. Unlisted and excepted sections cover construction requirements, for which the NRC accepts substitution of ACI 318. Chapter 1, “General Requirements”, Section 1.1 and 1.5 (less references to construction), Section 1.2, Section 1.4 Chapter 2, “Definitions”, All Chapter 3, “Materials, All, except Section 3.1, 3.2.3, 3.3.4, 3.5.3.2, 3.6.7, 3.7 Chapter 4, “Concrete Quality”, Section 4.1.4 Chapter 6, “Form work, Embedded Pipes, and Construction Joints”, Section 6.3.6(k), 6.3.8 Chapter 7, “Details of Reinforcement”, All Chapter 8, “Analysis and Design” - General Considerations, All Chapter 9, All Chapters 10-19, All Appendix A, All Appendix B, “Steel Embedments,” All, but note that the load combinations and load variation requirements of ANSI/ANS 57.9 must be met in addition to those of ACI 349 Section 9.2 cited at Section B.3.2 (given in Table 3-1 of NUREG-1536) Appendix C, “Special Provisions for Impulsive and Impactive Effects”, All, except that the load combinations and load variation requirements of ANSI/ANS 57.9 must be met in addition to those of ACI 349 Section 9.2 (given in Table 3-1 of NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.2	**INSTALLATION AND STRUCTURAL EVALUATION Reinforced Concrete Structures** Design Criteria (Part 5 of 5)	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	**Concrete Containments** ACI 359, Section CC, is acceptable for prestressed and reinforced concrete that is an integral component of a radioactive material containment vessel that must, in operation or in testing, withstand internal pressure. Application of ACI 359 is based on the containment function, regardless of whether the concrete structure is fixed or portable, or where the concrete structure is fabricated. ACI 359 also applies to structural concrete supports that are constructed as an integral part of the containment. If ACI 359 is applicable to an ISFSI/MRS structure, it is applicable for the full design, material selection, fabrication, and construction of that structure. The NRC has not accepted the substitution of elements of ACI 349 or ACI 318 for any portion of ACI 359 for an ISFSI/MRS structure. Structures for which ACI 359 is applicable shall also meet the minimum functional requirements of ANSI/ANS 57.9, where specific requirements in the subject area are not included in ACI 359.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Material Properties (Part 1 of 2)	72.24 (c)(3)	Acceptable criteria for materials used in all structural components and systems are given in 10 CFR 72.24 (c)(3). The information describing material properties must be consistent with the application of the accepted design criteria. For concrete structures as referenced in ACI 349-90, this would include ASTM standard specifications applicable to design and material specifications: A 36, A 53, A 82, A 184, A 185, A 242, A 416, A 421, A 496, A 497, A 500, A 501, A 572, A 588, A 615, A 706, A 722, C 33, C 144, C 150, C 595, and C 637. **Fabrication and Construction** Selection and validation of concrete mix to meet design requirements is considered to be a construction function. Specification of cement type, aggregates, and special requirements for durability and elevated temperatures is considered to be a design or material selection function, and therefore, to be governed by ACI 349 (ACI 359 if applicable).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Material Properties (Part 2 of 2)	72.24 (c)(3)	The following identifies sections of ACI 318, Building Code Requirements for Reinforced Concrete (chapters, appendix, and paragraphing per ACI 318-89) that have been accepted by the NRC for construction of ISFSI concrete structures that are not within the scope of ACI 359. Chapter 1, “General Requirements”, Section 1.1.1, 1.1.2, 1.1.3, and 1.1.5 (less references to design and material properties); Section 1.3 Chapter 2, “Definitions”, use ACI 349 Chapter 2 Chapter 3, “Materials”, Section 3.1, Section 3.8 (except delete A 616 and A 617) Chapter 4, “Durability Requirements”, All Chapter 5, “Concrete Quality, Mixing, and Placing”, All Chapter 6, “Form work, Embedded Pipes, and Construction Joints”, All (less references to design and material properties, these are governed by ACI 349) ASTM standard specifications acceptable for construction and associated testing are: C 31, C 39, C 42, C 94, C 109, C 172, C 192, C 260, C 494, C 496, C 685, and C 1017. The following standards relating to construction are identified in ACI 349 and may be used: C 88, C 131, C 289, and C 441.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.3.4	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Structural Analysis (Part 1 of 2)	72.24 (d)(1), and (d)(2), (i), 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l).	Requirements for acceptable structural analysis are given in 10 CFR 72.24 (d)(1), and (d)(2), (i), as well as 10 CFR 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l). Design analyses should be prepared such that they may be readily audited to permit determination of the sources of expressions used, values of material properties, data from other supporting calculations, and assumptions. ANSI N45.2.11 provides guidance for preparation of design analyses which is acceptable to the NRC. The design specification for concrete structures shall identify all loading conditions and combinations of loadings. The specification shall establish the design internal and external pressures, the design temperatures, and all the design mechanical loads. The specification shall identify all combinations of design loads which can occur simultaneously. The specification shall establish service loadings (with appropriate service limits), which are discussed as normal, offnormal, and accident conditions in this SRP.
Licensing, Design	5.4.3.5	**INSTALLATION AND STRUCTURAL EVALUATION** **Reinforced Concrete Structures** Structural Analysis (Part 2 of 2)	72.24 (d)(1), and (d)(2), (i), 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l).	The NRC accepts strength design as presented in the current ACI 349 for concrete structures important to safety that are not within the scope of ACI 359. ACI 359 is based on allowable stress design. Load definitions and load combinations shown in Table 3-1 of NUREG-1536 have been accepted by the NRC for analysis of steel and reinforced concrete ISFSI and MRS structures important to safety. The load combinations are as included or derived from ANSI/ANS 57.9 and ACI 349. Load combinations to be used for concrete structures designed in accordance with ACI 359 should be as given in ACI 359 (Section CC3230).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.4	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety**	72.24, 72.40, 72.82, 72.102, 72.106, 72.120, 72.122, 72.128	No text provided.
Licensing, Design	5.4.4.1	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Description of Other SSCs Important to Safety	72.24 (a) and (b), 72.82(c)(2), 72.106(a), (b), and (c)	10 CFR 72.24 (a) and (b), 10 CFR 72.82(c)(2), and 10 CFR 72.106(a), (b), and (c) outline the contents of the application, which includes design descriptions in sufficient detail to support findings in the SER. For other SSCs important to safety this would include text descriptions, drawings, figures, tables, and specifications that would fully define the structural features of the items identified.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.4.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Design Criteria	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	The regulatory requirements given in 10 CFR 72.24(c)(1), (c)(2), and (c)(4); 10 CFR 72.40 (a)(1); 10 CFR 72.120 (a), and (b); 10 CFR 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l); 10 CFR 72.128 (a) and (b); and 10 CFR 72.236(b), (e), (f), (g), and (k) identify acceptable design criteria. **Codes and Standards** The NRC accepts use of ANSI/ANS 57.9 and the codes and standards cited therein as the basic references for ISFSI structures important to safety that are not designed in accordance with the ASME B&PVC Section III. The principal included references applicable to steel structures and components are the following: • AISC, "Specification for Structural Steel Buildings - Allowable Stress Design and PlasticDesign" • AISC, "Code of Standard Practice for Steel Buildings and Bridges" • AWS D 1.1, "Structural Welding Code-Steel" • ASCE 7, "Minimum Design Loads for Buildings and Other Structures," however, note that the load combinations of ANSI/ANS 57.9 are to be used
Licensing, Design	5.4.4.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Material Properties	72.24 (c)(3)	Acceptable criteria for materials used in all structural components and systems are given in 10 CFR 72.24 (c)(3).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.4.4	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Structural Analysis	72.24 (d)(1), and (d)(2), (i), 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l)	Requirements for acceptable structural analysis are given in 10 CFR 72.24 (d)(1), and (d)(2), (i), as well as 10 CFR 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l). Design analyses should be prepared such that they may be readily audited to permit determination of the sources of expressions used, values of material properties, data from other supporting calculations, and assumptions. ANSI N45.2.11 provides guidance for preparation of design analyses which is acceptable to the NRC. The design specification for concrete structures shall identify all loading conditions and combinations of loadings. The specification shall establish the design internal and external pressures, the design temperatures, and all the design mechanical loads. The specification shall identify all combinations of design loads which can occur simultaneously. The specification shall establish service loadings (with appropriate service limits), which are discussed as normal, offnormal, and accident conditions in this SRP.
Licensing, Design	5.4.5	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs**	72.24, 72.40, 72.82, 72.106, 72.120, 72.122, 72.128	No text provided.
Licensing, Design	5.4.5.1	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Description of Other SSCs	72.24 (a) and (b), 72.82 (c)(2), 72.106 (a), (b), and (c)	10 CFR 72.24 (a) and (b), 10 CFR 72.82 (c)(2), and 10 CFR 72.106 (a), (b), and (c) outlines the contents of the application, which includes design descriptions in sufficient detail to support findings in the SER. For other SSCs subject to NRC approval this would include text descriptions, drawings, figures, tables and specifications that would fully define the structural features of the items identified.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.5.2	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Design Criteria	72.24 (c)(1), (c)(2), and (c)(4), 72.40 (a)(1), 72.120 (a), and (b), 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l), 72.128 (a) and (b), 72.236(b), (e), (f), (g), and (k)	The regulatory requirements given in 10 CFR 72.24(c)(1), (c)(2), and (c)(4); 10 CFR 72.40 (a)(1); 10 CFR 72.120 (a), and (b); 10 CFR 72.122 (a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l); 10 CFR 72.128 (a) and (b); and 10 CFR 72.236(b), (e), (f), (g), and (k) identify acceptable design criteria. **Codes and Standards** The principal structural codes and standards for SSCs which are not important to safety but which are subject to NRC approval include: • ASCE 7 • Uniform Building Code (UBC) • AISC, "Specification for Structural Steel Buildings, Allowable Stress Design and Plastic Design" • AISC, "Code of Standard Practice" • ASME B&PVC, Section VIII The above include acceptable load definitions and load combinations. Load definitions and load combinations shown in Table 3-1 of NUREG-1536 have been accepted by the NRC for analysis of steel and reinforced concrete ISFSI structures important to safety. These may also be used for structures not important to safety.
Licensing, Design	5.4.5.3	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Material Properties	72.24 (c)(3)	Acceptable criteria for materials used in all structural components and systems are given in 10 CFR 72.24 (c)(3).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	5.4.5.4	**INSTALLATION AND STRUCTURAL EVALUATION** **Other SSCs Important to Safety** Structural Analysis	72.24 (d)(1), and (d)(2), (i), 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l)	Requirements for acceptable structural analysis are given in 10 CFR 72.24 (d)(1), and (d)(2), (i), as well as 10 CFR 72.122 (b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l). Design analyses should be prepared such that they may be readily audited to permit determination of the sources of expressions used, values of material properties, data from other supporting calculations, and assumptions. ANSI N45.2.11 provides guidance for preparation of design analyses which is acceptable to the NRC. The design specification for all other SSCs subject to NRC approval shall identify all loading conditions and combinations of loadings. The specification shall establish the design internal and external pressures, the design temperatures, and all the design mechanical loads. The specification shall identify all combinations of design loads which can occur simultaneously. The specification shall establish service loadings (with appropriate service limits), which are discussed as normal, off-normal, and accident conditions in this SRP. Load combinations for analysis of structures not important to safety but subject to NRC approval should be as given in acceptable codes and standards. The load combinations given in ACI 318 or the Uniform Building Code (UBC) are appropriate for SSCs not important to safety.
Licensing, Design	6.4	**THERMAL EVALUATION** **Acceptance Criteria**	72.92, 72.122, 72.128	This section identifies the acceptance criteria used for the thermal evaluation review. Specific acceptance criteria are delineated in this section.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	6.4.1	**THERMAL EVALUATION** **Decay Heat Removal Systems**	72.122 (h), 72.128 (a)	The spent fuel cladding must be protected during storage against degradation that leads to gross fuel rupture (10 CFR 72.122(h)). Decay heat removal systems shall have testability and reliability consistent with their importance to safety (10 CFR 72.128(a)). The applicant must provide a description of the proposed heat removal system. The description must describe the mechanisms for removing decay heat including any active components or operator actions necessary for operation during normal, off-normal, and accident conditions. If the decay heat removal system is for a pool, the description must address the layout of piping and equipment, control systems for managing flow, and instrumentation systems for monitoring water conditions. The applicant must provide evidence that the decay heat removal system will operate reliably under normal, off-normal, and accident conditions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	6.4.2	**THERMAL EVALUATION** **Material Temperature Limits**	72.128(a)	SSCs important to safety shall be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions so as to support the performance of the intended safety function (10 CFR 72.128(a). The applicant must identify the temperature limits for fuel cladding, solidified waste packages, and materials used for SSCs that are important to safety. The applicant shall also provide a basis for the temperature limits. The temperature limits for fuel cladding should include consideration of mechanisms that can lead to gross cladding rupture. Fuel cladding temperature during dry storage shall be maintained below the expected damage-threshold temperatures for normal conditions and a minimum of 20 years dry storage for ISFSI or MRS design and environmental conditions. The fuel cladding temperature should also generally be maintained below 570oC (1058oF) for short-term off-normal, short-term accident, and fuel transfer operations (e.g., vacuum drying of the cask or dry transfer) (PNL-4835).
Licensing, Design	6.4.3	**THERMAL EVALUATION** **Thermal Loads and Environmental Conditions**	72.92, 72.122(b)	The applicant must identify the design basis thermal loads from the spent fuel or high-level waste, as well as the thermal loads associated with insolation and the site parameters that determine the rate at which heat can be removed from the ISFSI or MRS (10 CFR 72.92). The heat removal system must accommodate the decay heat of the spent fuel or high-level waste and the site normal, off-normal, and accident thermal conditions (10 CFR 72.122(b)).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	6.4.4	**THERMAL EVALUATION** **Analytical Methods, Models, and Calculations** (Part 1 of 2)	72.122(h), 72.128 (3)	The applicant shall present a thermal analysis that demonstrates the ability to manage design basis heat loads and have the various materials remain within temperature limits. The analysis shall be conducted for normal, off-normal, and accident conditions. The analysis shall also present temperature and temperature gradient information that is necessary to support the structural analysis. The applicant shall identify the codes or analytical methods used for thermal analysis and discuss the basis for the parameters selected for the analysis. For each fuel assembly-type proposed for storage, the dry storage system shall ensure a very low probability (e.g., 0.5 percent), per fuel rod, of cladding breach during long-term (e.g., 40 year) storage (10 CFR 72.122(h), PNL-6189). The maximum internal pressure of the cask shall remain within its design pressure for normal, off-normal, and accident conditions assuming 1 percent, 10 percent, and 100 percent ruptured fuel rods respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods (10 CFR 72.128(3), 10 CFR 72.122(h)).
Licensing, Design	6.4.4	**THERMAL EVALUATION** **Analytical Methods, Models, and Calculations** (Part 2 of 2)	72.122(h), 72.128 (3)	Under the conditions where any of the cask component or fuel cladding temperatures are close (within 5%) to their limiting values during an accident or the maximum normal operating pressure is within 10% of its design basis pressure, or any other special considerations affected by fission gas concentrations, the applicant should analyze the potential impact of the fission gas in the cask on the cask component and fuel cladding temperature limits and the internal cask pressure. The pool system shall be designed so that, for all postulated events, the pool water level is maintained at a level above the top of the active fuel to ensure adequate decay heat removal and shielding (ANS/ANS 57.7).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	6.4.5	**THERMAL EVALUATION** **Fire and Explosion Protection**	72.122(c), 72.122(j)	Spent fuel assemblies, other radioactive materials, and SSCs important to safety shall have adequate protection against fires and explosions to minimize and control the release of radioactive material to the environment (10 CFR 72.122(c)). Measures for fire prevention, fire detection, fire suppression, and fire containment for the protection of the spent fuel assemblies and SSCs important to safety shall be provided. 10 CFR 72.122(c) requires that: • SSCs important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. • Non-combustible and heat resistant materials must be used wherever practical throughout the ISFSI or MRS, particularly in locations vital to the maintenance of safety control functions. • Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on SSCs important to safety. • The design of the ISFSI or MRS must include provisions to protect against adverse effects that might result from the operation or failure of the fire suppression system. In addition, 10 CFR 72.122(j) requires that a control room or control area, if appropriate for the ISFSI or MRS design, must be designed to permit occupancy and actions to be taken to monitor the safety of the ISFSI or MRS under normal conditions and to provide safe control of the ISFSIor MRS under off-normal or accident conditions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4	**SHIELDING EVALUATION** **Acceptance Criteria**	20.1201, 20.1301, 20.1302, 72.24, 72.104, 72.126, 72.128	The information submitted in the SAR must be of sufficient scope and detail to allow for a thorough evaluation of proposed shielding, including the performance of independent dose rate estimates. All applicable regulatory requirements must be satisfied, and the methods for determining compliance must be acceptable to NRC. The following sections provide criteria for acceptability of SAR informational content and the details and method of evaluation of the proposed shielding features. Primary guidance related to the information to be included in the SAR is provided by Regulatory Guide 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation, (Dry Storage)” and NUREG-1536, Chapter 5. The guidance in this section summarizes and supplements the guidance provide by those sources.
Licensing, Design	7.4.1	**SHIELDING EVALUATION** **Contained Radiation Sources**	72.126	10 CFR 72.24 describes the required contents of the application. To meet those requirements, the SAR must describe each type of contained radiation source used as a basis for shield design calculations. The physical and chemical form, source geometry, radionuclide content, and estimated curie value and bases for estimation must be described in a manner suitable for use as input for shielding calculations.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.1.1	**SHIELDING EVALUATION** **Contained Radiation Sources** Gamma Sources	72.126	A tabulation of radiological characteristics for each gamma-ray source type must be provided, including isotopic composition and photon yields by X- and gamma-ray energy group. The SAR must specify gamma source terms for both spent fuel and activated materials. The energy group structure from the source term calculation must correspond to that of the cross-section set of the shielding calculation. The computer methodology or database application used to compute source term strength must be specifically identified. The SAR must describe the extent to which radioactivity may be induced by interactions involving neutrons originating in the stored materials. The SAR must provide source term descriptions for induced radioactivity and the bases (assumptions and analytical methods) used for their estimation. Alternatively, the SAR may describe the bases for excluding induced radioactivity source terms.
Licensing, Design	7.4.1.2	**SHIELDING EVALUATION** **Contained Radiation Sources** Neutron Sources	72.126	The SAR must describe the neutron source terms and tabulate the neutron yield by energy group. The SAR must describe the bases used to determine the source terms.
Licensing, Design	7.4.2	**SHIELDING EVALUATION** **Storage and Transfer Systems**	72.24, 72.104, 72.126, 72.128	10 CFR 72.126 and 72.128 require that the storage and handling systems requiring shielding be described. The SAR must provide design criteria and descriptions of design features for shielded containers.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.2.1	**SHIELDING EVALUATION** **Storage and Transfer Systems** Design Criteria	20.1201, 20.1301, 20.1302, 72.104	10 CFR 20.1201, 20.1301, 20.1302, and 10 CFR 72.104 provide dose rate criteria for occupational exposure and for members of the public. The principal design criteria (presented in SAR Section 3) must specify the criteria that have been used as a basis for protection against direct radiation. Design criteria must include the identification of maximum dose rates for each type of shielded container (transfer cask, storage cask, etc.). Design dose rates must also be specified for occupancy areas and correlated with occupancy times and distance to sources. An estimate of collective doses (person-rem per year) must be provided for each occupancy area and for various operations (see Chapter (Section) 11, Radiation Protection). The design dose rates must consider ALARA objectives. The SAR must identify choices between otherwise comparable alternatives affected by ALARA considerations and show that further reduction of collective doses from direct radiation is not practicable.
Licensing, Design	7.4.2.2	**SHIELDING EVALUATION** **Storage and Transfer Systems** Design Features	72.24, 72.104, 72.126, 72.128	The SAR must describe the transfer and storage systems, including the use of shielding, to reduce direct radiation dose rates. The SAR must describe various uses of shielding features at the proposed ISFSI or MRS, including any of the following that apply: • Shielding provided by the radioactive material being stored • Neutron capture provided by borated water in casks and storage pools, and by borated materials incorporated into casks • Gamma and neutron shielding provided by the structural and nonstructural materials (e.g.,lead) forming the walls and ends of the storage or transfer casks • Temporarily positioned shielding used during operations for preparing the storage confinement cask for storage or retrieval, and/or during transfer into the storage position at the storage location • Shielding provided by pool facility interior and exterior walls • Shielding provided by natural or man-made earth barriers between the radioactive material and the area beyond the controlled area boundary.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.3	**SHIELDING EVALUATION** **Shielding Composition and Details**	72.126	10 CFR 72.24 requires that the application include information relative to materials and arrangements of all structures, systems, and components important to safety.
Licensing, Design	7.4.3.1	**SHIELDING EVALUATION** **Shielding Composition and Details** Composition and Material Properties	72.126	The SAR must describe the composition of shielding materials and geometries. The SAR must give material compositions, densities and references for these data, for all materials used. The SAR must give references to the source of the data and the validation for the data for nonstandard materials (e.g., proprietary neutron shield material). The SAR must describe the potential for shielding material to experience changes in material properties at temperature extremes. The SAR should give and reference temperature sensitivities of shielding materials.
Licensing, Design	7.4.3.2	**SHIELDING EVALUATION** **Shielding Composition and Details** Shielding Details	72.126	The SAR must describe the geometric arrangement of shielding. The SAR must use illustrations to identify the spatial relationships among sources, shielding, and design dose rate locations. The SAR must clearly indicate the physical dimensions of sources and shielding materials. The SAR must identify penetrations, voids, or irregular geometries that provide potential paths for gamma or neutron streaming. These potential streaming paths must be clearly identifiable on submitted drawings. The SAR must describe design features used to minimize streaming through these penetrations. The SAR must clearly state any differences between shielding features during normal or off-normal conditions and accident level conditions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.4	**SHIELDING EVALUATION** **Analysis of Shielding Effectiveness**	20.1201, 20.1301, 20.1302, 72.24, 72.104, 72.126, 72.128	The SAR must describe the computational models, data, and assumptions used in evaluating shielding effectiveness, and must provide dose rate estimates for areas of concern.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.4.1	**SHIELDING EVALUATION** **Analysis of Shielding Effectiveness** Computational Methods and Data	20.1201, 20.1301, 20.1302, 72.24, 72.104, 72.126, 72.128	The SAR must identify the computer models used in evaluating shielding for each significant radiation source identified in Section 7.4.1 and reference the appropriate documentation. For each computer program used, the SAR must provide test problem solutions that demonstrate substantial similarity to solutions from other sources (hand calculations, published literature results, etc.). The SAR must provide a summary that compares the test problem solutions in either graphical or numeric form. These solutions may be referenced and need not be submitted in the SAR if the references are widely available or have been previously submitted to the NRC for the same model and version. The SAR must clearly present the data used as input for computational purposes. The SAR must identify any differences between actual material properties or physical dimensions and those used in the analytical method (e.g., for simplifying the computational process). The SAR must defend any simplifying assumptions by showing that the approach used will result in conservative (bounding) estimates. The SAR must state the basis for the flux-to-dose-rate conversion in its shielding analysis, including conversions that are done by a code using its own data library. The NRC accepts flux-to-dose rate conversion factors in ANSI/ANS 6.1.1. The SAR must include a representative computer code input file used in type of shielding computation performed for the installation.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.4.2	**SHIELDING EVALUATION** **Analysis of Shielding Effectiveness** Dose Rate Estimates (Part 1 of 2)	20.1201, 20.1301, 20.1302, 72.24, 72.104, 72.126, 72.128	The SAR evaluation of shielding effectiveness must include estimates of dose rates in representative areas around the storage and transfer systems. The SAR estimates must account for such factors as distance to occupied areas, duration of operations, expected occupancy rates, contributions from radionuclide releases and other factors . These criteria are identified and evaluated in the radiation protection evaluation described in Chapter (Section) 11. The criteria below relate primarily to the completeness of information provided in the SAR. The SAR must clearly indicate the physical locations on and around storage or transfer casks for which dose rate calculations have been performed. These locations must include points on or in the immediate vicinity of cask surfaces where workers will perform operations during loading, retrieval, handling operations, and any projected maintenance and surveillance. For storage casks with labyrinthine air flow passages, the SAR must include dose rate estimates for the air inlets and air outlets. The SAR must identify points that have the highest calculated dose rates. The SAR must include dose rate estimates for all onsite areas at which workers will be exposed to elevated dose rates. The SAR must compute dose rates within restricted areas in enough detail to estimate doses received by workers performing ISFSI or MRS functions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	7.4.4.2	**SHIELDING EVALUATION** **Analysis of Shielding Effectiveness** Dose Rate Estimates (Part 2 of 2)	20.1201, 20.1301, 20.1302, 72.24, 72.104, 72.126, 72.128	The SAR must present dose rate estimates for representative points on the perimeter of the controlled area and at locations beyond the controlled area boundary. The SAR must specify these estimates with respect to distance and direction in a manner that will allow for estimation of population dose within a 5-mile (8-km) radius of the site. These dose rates must include contributions from both direct line-of-sight and air-scattered radiation emerging from casks or other shielded sources. For storage confinement casks, the SAR must calculate the dose rate at 1 meter from the cask surface for off-normal events and conditions that result in a significant dose rate increase. The model used for these calculations must be consistent with the expected condition of the cask after the event.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4	**CRITICALITY EVALUATION** **Acceptance Criteria**	72.40, 72.124	This section identifies the acceptance criteria used for the criticality review. Four types of criteria are described. The first describes criticality design criteria and features including required conditions, assumptions, and scenarios. The second identifies the requirements for the specifications regarding stored nuclear material that are acceptable to the NRC. The third describes the features of criticality analysis models which are acceptable to the NRC. The fourth identifies the features of applicant criticality analyses including the specific computer program, benchmarks, and multiplication factor determination which constitute an acceptable submittal for criticality safety. Under the right conditions, certain isotopes of selected heavy elements, especially uranium and plutonium, have the ability to split or fission after absorbing a neutron and release energy along with several new neutrons from this fission process. The fission process can be self-sustaining or even grow by a chain reaction, which can produce as many or more neutrons than are absorbed. In criticality terminology, the term, k-effective or k_{eff} is the net ratio of neutrons produced per neutron absorbed in a mass of fissionable material. A k_{eff} of 1.0 indicates a critical mass whereas a k_{eff} of less than 1.0 is an indication of a subcritical condition.
Licensing, Design	8.4.1	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features**	72.40, 72.124	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.1.1	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features** Criteria (Part 1 of 3)	72.40, 72.124	The regulatory requirements given in 10 CFR 72.40 and 10 CFR 72.124 identify acceptable design criteria. The NRC generally considers the design criteria identified below to be acceptable to meet the criticality requirements of 10 CFR 72 for storage confinement casks: • The multiplication factor, $k_{eff,}$ including all biases and uncertainties at a 95 percent confidence level, must not exceed 0.95 under all credible normal, off-normal, and accident conditions and events. • Conditions for criticality safety (satisfaction of the limit on multiplication factor, k_{eff}) of subject radioactive material while at the Independent Spent Fuel Storage Installations (ISFSI) or Monitored Retrievable Storage (MRS) must include: — no burnup credit. (The conservative assumption of fresh unburned fuel provides a worst case criticality analysis; however, 10 CFR 72.3 requires that spent fuel have been irradiated and cooled at least one year as a condition for storage.) Alternately, burnup credit may be taken using the guidelines described in section 8.4.5 of this SRP. — no credit taken for flammable neutron absorbers or for any solid poisons that may melt or lose any significant mass from the original solid form by melting or vaporization at any of the temperature and pressure conditions that may be experienced while in use — no credit taken for liquid neutron shielding material (except that k_{eff} for the situation of a loaded confinement cask with liquid that serves as both shielding and absorber and is used in the confinement cask during loading operations or in the pool shall be based on presence of the water and bounding level(s) of poison)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.1.1	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features** Criteria (Part 2 of 3)	72.40, 72.124	— no more than 75 percent credit for fixed neutron absorbers, unless comprehensive fabrication acceptance tests capable of verifying the presence and uniformity of the neutron absorber are implemented — determination and use of optimum (i.e., most reactive) moderator density • The multiplication factor limit on k_{eff}, must be met for all conditions and events while at the ISFSI and MRS. This does not require determination of k_{eff} for every situation. However, it must be demonstrated that the situations that have the highest keff have been analyzed and that thereby the normal, off-normal, and accident and conditions with the lowest margins of safety have been analyzed; or are enveloped by the analyses conducted and included in the SAR and its supporting documentation (ANSI/ANS 8.17-1984). Conditions and events to be considered include, but are not limited to, the following: — in dry storage — in temporary or prolonged storage in the pool — during cask loading and unloading operations, including: —— transferring subject radioactive material to or from the storage cask or a transfer container (and possible drops, and including possible selection of the wrong material for loading); —— lift and translation movements (and possible drops); —— dewatering and charging operations, including situations in which a cask could be full of (unborated) steam; or with a partial fill of liquid water (borated if applicable to the procedures) and the remainder steam. — during all on-site transfer, and transportation operations — during and following possible drops and other accident events including credible flood from natural or man-made causes — during earthquakes

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.1.1	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features** Criteria (Part 3 of 3)	72.40, 72.124	— during and following a non-mechanistic cask tip-over (if a more severe event is not identified by accident analysis) — with the stored material in the densest configuration permitted by the basket or other separators and with the most conservative assumptions for tolerance stack-ups • At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, must occur before an accidental criticality is possible (ANSI/ANS 8.1-1983). For analysis, "accidental criticality" is defined as exceeding k_{eff} of 0.95 with a confidence level less than 95% and a 95% probability that k_{eff} =0.95 will not be exceeded. • Criticality safety of the design must be based on favorable geometry (preferred), permanent fixed neutron absorbing materials (poisons), or both. • Where solid neutron-absorbing materials are used, the design must provide a means to verify their initial efficacy, such as manufacturer's data or in-situ measurements (ANSI/ANS 8.21). Chapter 6 of NUREG-1536 provides a basis for accepting the 20-year continued efficacy of fixed neutron poisons. • Unless it is shown that all spent fuel to be stored will be contained within completely intact cladding, the occurrence of pinholes and cracks in the cladding (and water fill of the voids within the cladding) must be assumed for the criticality analysis if it results in a higher keff. The water fill in the fuel-to-cladding gap should be assumed to be unborated since this is conservative from a criticality safety viewpoint. The aforementioned criteria should be specifically incorporated into the SAR and applicable supporting criticality calculations.

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Licensing, Design	8.4.1.2	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features** Features (Part 1 of 3)	72.40, 72.124 (b)	The regulatory requirements given in 10 CFR 72.124(b) identify acceptable design criteria for criticality control. The NRC generally considers the design criteria identified below to be acceptable to meet the criticality control requirements of 10 CFR 72 for storage confinement casks. The NRC accepts use of borated water in the pool and during cask loading and unloading (of subject radioactive material to or from the storage confinement cask) operations as a means of criticality control if the minimum boron content is a technical specification. If borated water is used for criticality control, then administrative controls and/or design features must be used to ensure: (1) that pool boron concentration is maintained throughout the pool; and, (2) that accidental flooding of a cask with unborated water cannot occur (as in retrieval operations or in an interrupted loading operation requiring cask reflooding in anticipation of off-loading or additional operations on or about the cask). The alternative is that the criticality analysis assume accidental flooding with unborated water. If borated water is not to be used, the SAR should specify if any dummies (for fuel rods or other fissionable items to be stored) are to be used in storage positions within the confinement cask or pool to displace water. Credit for this displacement of water in the criticality analysis requires acceptable evidence that the storage positions not occupied by the subject radioactive materials will always be occupied by the dummies.

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Licensing, Design	8.4.1.2	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features** Features (Part 2 of 3)	72.40, 72.124 (b)	Borated water and any other liquids are not acceptable as a means of criticality control for a cask in dry storage. This includes use of any credit in criticality analysis for presence (outside the cask confinement barrier) of a liquid that may provide neutron shielding. Presence and optimum (most reactive) moderator density of the liquid shall be assumed if it increases k_{eff}. If more than one certified or licensed basket design of the same supplier could fit in the cask, the type basket to be used should be among the data stamped on the plate on the exterior of the storage confinement cask. The NRC accepts comparative neutron flux measurement made external to a confinement cask as a positive means for verifying continued efficacy of solid neutron absorbing materials incorporated in the storage cask system, if the following are acceptable: • testing procedures, • instrumentation, • accuracies, and • determination of baseline measurements for subsequent use in comparisons.

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Licensing, Design	8.4.1.2	**CRITICALITY EVALUATION** **Criticality Design Criteria and Features** Features (Part 3 of 3)	72.40, 72.124 (b)	The NRC has accepted a requirement for acceptance testing of the poisons during fabrication as a positive means for verifying continued efficacy of solid neutron absorbing materials incorporated in the storage cask system. This testing should show that: (1) the material is not subject to degradation from physical or chemical actions that may occur over the system life, (2) the material will not be degraded by time-integrated gamma radiation emitted by the spent fuel fission products, and (3) the small neutron flux from spontaneous fission and subcritical multiplication results in a negligible depletion of poison material over the storage period. Inclusion of evidence of satisfactory similar use of the material for a 20-year period is desirable. Tolerances for structures, systems, and components (SSCs) material, fabrication, and assembly can be important in identifying worst case (lowest margin of safety) geometries, material compositions, and densities. The tolerances for the properties and construction of all SSCs involved in criticality analyses should be used in the analyses and must then be also identical or conservatively bounded by the tolerances shown in the definition of the ISFSI or MRS design. The analyses should be based on the most conservative combination of tolerances.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.2	**CRITICALITY EVALUATION** **Stored Material Specifications** (Part 1 of 2)	72.124 (a)	The regulatory requirements given in 10 CFR 72.124(a) identify acceptable design criteria for stored material specifications. The NRC generally considers the design criteria identified below to be acceptable to meet the criticality requirements of 10 CFR 72 for storage confinement casks. The stored material specifications must include the ranges of properties of concern for criticality analysis (guidance on data required is provided in Standard Review Plan for Spent Fuel Dry Storage Facilities [FSRP] Section 4.4.1). Stored material characteristics of probable concern for the various criticality analyses include those listed below. These should be stated for each known type of spent fuel to be stored and for other radioactive material to be stored for which criticality analysis is appropriate. Radioactive materials which, due to their atomic properties and/or physical maximum densities of the solid material are not of criticality concern should be identified as such, as rationale for not including criticality analyses. Data identified below that are not required for the analytical approach used by the applicant should still be provided as they may be needed for confirmatory and independent analyses by the NRC (performed as part of the evaluation effort): • fuel manufacturer identity and spent fuel design/model • maximum U^{235} enrichment of fuel and type [e.g., 4.2% enrichment, Boiling Water Reactor (BWR)] of fuel assemblies • the maximum fuel pin enrichment [for BWR or Pressurized Water Reactor (PWR) fuel]

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Licensing, Design	8.4.2	**CRITICALITY EVALUATION** **Stored Material Specifications** (Part 2 of 2)	72.124 (a)	• for other radioactive materials that are to be stored and may be fissionable — isotopes present and their densities — means by which densities are limited — geometric data on the configuration (e.g., racks, basket) holding the materials including tolerances and uncertainties, and neutron absorption material integral to the configuration — characteristics (materials, densities, geometries, tolerances, uncertainties) of any encapsulation used to provide confinement and structural support during handling and when within the storage confinement barrier
Licensing, Design	8.4.3	**CRITICALITY EVALUATION** **Analytical Means**	72.124 (a) and (b)	The regulatory requirements given in 10 CFR 72.124(a) and (b) identify acceptable design criteria. The NRC generally considers the design criteria identified below to be acceptable to meet the criticality requirements of 10 CFR 72 for storage confinement casks.
Licensing, Design	8.4.3.1	**CRITICALITY EVALUATION** **Analytical Means** Model Configuration (Part 1 of 2)	72.124 (a) and (b)	The model used for the criticality evaluation must adequately describe normal, off-normal, and accident conditions analyzed. The model must provide for the most reactive tolerance combinations and any steady state elastic deformation of the positioning (racks or basket) or plastic deformation of the structure that could result from accident events. The dimensions and materials of the model used for the criticality analysis should be the same as those in the design definition (elsewhere in the SAR). If there are differences, the model must be shown to be conservative (result in a higher keff). The NRC accepts substitution of ordinary water for end sections and support structures of the fuel in the model. Substitution of borated water for other materials is not acceptable unless it can be shown to be conservative. Sufficient conditions must be modeled and analyzed to ensure that the highest keff have been determined and that conditions and configurations not analyzed are enveloped by those analyzed.

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Licensing, Design	8.4.3.1	**CRITICALITY EVALUATION** **Analytical Means** Model Configuration (Part 2 of 2)	72.124 (a) and (b)	The model should reflect conservative assumptions in all variables. This includes (but may not be limited to) variables identified below: • location of fissionable material within positioning basket or other framework (e.g., fuel rods would be positioned to be tight against the dividing spacing material and closest to the center of the array for vertical storage and as close to center as permitted by gravity, or potentially caused by in transit vibration for a horizontal cask position) • fuel density • U^{235} enrichment of fuel • manufacturing tolerances must be assumed to be at their most conservative (i.e.,maximum reactivity) value within the allowed tolerance band. No statistical combination of uncertainties is allowed for manufacturing tolerances • flooding in the fuel rod pellet-to-clad gap region The NRC accepts use of a heterogeneous model of each fuel rod. If, instead, the model homogenizes the entire fuel assembly (over the volume of the assembly), the applicant must acceptably demonstrate that the homogenized model is conservative relative to a heterogeneous model. This may be done by using both homogeneous and heterogeneous models in a criticality computation or by benchmarking to an acceptable (number and relevance) set of criticality experiments. The criticality analysis model must be described in sufficient detail, either in the SAR or supporting calculations, to show conformance to the requirements in this section.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.3.2	**CRITICALITY EVALUATION** **Analytical Means** Material Properties	72.124 (a) and (b)	The material compositions and densities must be provided for all materials used in the calculational model. The sources of the properties should be referenced. The amount and geometry of fixed poison used in the criticality analysis should be no more than the minimums to be verified by acceptance testing. Validation of the poison concentration is addressed in acceptance testing. An appropriate set of cross-sections should be determined and identified, and the sources should be referenced. Cross-sections may be obtained with the criticality computer codes or developed independently from another source. For multigroup calculations, the spectrum of the neutron flux used to construct the group cross-sections must be similar to that of the cask. Cross-sections and the computer program must be benchmarked by comparison to experimental data.
Licensing, Design	8.4.4	**CRITICALITY EVALUATION** **Applicant Criticality Analysis**	72.40, 72.124	The regulatory requirements given in 10 CFR 72.40 and 10 CFR 72124(a) identify acceptable design criteria. The NRC generally considers the design criteria identified below to be acceptable to meet the criticality requirements of 10 CFR 72 for storage confinement casks. The SAR must include criticality analyses for the most reactive cases for the items and materials that approval of the application will allow to be stored. These must be demonstrated to include or envelop the loadings and situations that have the highest values of k_{eff}.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.4.1	**CRITICALITY EVALUATION** **Applicant Criticality Analysis** Computer Program (Part 1 of 2)	72.40, 72.124	The SAR must identify the computer program and cross-section used in criticality analyses. The NRC has accepted both Monte Carlo and deterministic computer codes for criticality calculations. Monte Carlo codes are generally more suited to three-dimensional geometry, and therefore, more are widely used to evaluate spent fuel cask designs. Two acceptable Monte Carlo codes are SCALE/KENO (NUREG/CR-0200 and CCC-619) and MCNP (LANL, Dec 1993). KENO is a multi-group code that is part of the SCALE sequence. MCNP permits use of continuous cross sections. The NRC has accepted use of MICROX and DTFX (deterministic computer codes). If a multigroup treatment is used, the neutron spectrum of the cask must be appropriately considered. In addition to selecting a cross-section set collapsed with an appropriate flux spectrum, a more detailed processing of the energy-group cross-sections is also required to properly account for resonance absorption and self-shielding. The use of KENO as part of the SCALE sequence provides for such processing directly.
Licensing, Design	8.4.4.1	**CRITICALITY EVALUATION** **Applicant Criticality Analysis** Computer Program (Part 2 of 2)	72.40, 72.124	Some cross-section sets (e.g., HANSEN-ROACH) include data for fissile and fertile nuclides (based on a potential scattering cross-section) that can be input by the user. If a stand-alone version of KENO is used, potential scattering must be properly considered. [Note: The “working-format” library, commonly distributed with the Version 4.0 of SCALE/KENO to enable calculations of the manual’s sample problems, is not acceptable for criticality calculations of actual systems (NRC Information Notice 91-26). The NRC has accepted “27 Group NDF4” cross-section library in SCALE-4.1 PC KENO Va for criticality calculations.] Multigroup cross-section sets may be used in analyses of cask models with separate regions of water and steam or variations in the boron concentration. These involve use of different flux spectra in different regions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.4.2	**CRITICALITY EVALUATION** **Applicant Criticality Analysis** Multiplication Factor	72.40, 72.124	Variation in results of different computations of k_{eff} for different situations and with different codes and models should be rationalized and explained. Sensitivity parametric analyses may be used to provide the required demonstration that the highest keff with confidence level of 95% (with a k_{eff} • 0.95) have been determined. Certain cases which may require a lower value than 0.95 for maximum allowable keff are discussed in Chapter 6, Section V.4.b of NUREG-1536. For verification of Monte Carlo calculations, the number of neutron histories and convergence criteria should be appropriate. As the number of neutron histories increases, the mean value for k_{eff} should approach some fixed value, and the standard deviation associated with each mean value should decrease. Depending on the code used, a number of diagnostic calculations are generally available to demonstrate adequate convergence and adequate statistical variation. For deterministic codes, a convergence limit may be prescribed in the input. The selection of a proper convergence limit and achievement of this limit must be described and demonstrated in either the SAR or supporting criticality calculations.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.4.3	**CRITICALITY EVALUATION** **Applicant Criticality Analysis** Benchmark Comparisons (Part 1 of 2)	72.40, 72.124	Computer codes for criticality calculations must be benchmarked against critical experiments. Benchmark comparisons must be documented in appropriate calculations and/or the SAR. Benchmark comparisons can validate the computer code, its use on a specific geometric configuration, the neutron cross-sections used in the analysis, and consistency in modeling. Benchmark comparisons should be made by the analyst(s) and organization that will be performing the actual criticality analysis to qualify the analyst and computer environment. The calculated k_{eff}s and confidence levels of the base criticality computations must be adjusted to include the appropriate bias (the average of the differences between results and measurement) and uncertainties determined from the benchmark comparisons. The benchmark experiments should be relevant to the actual situation analyzed (ANSI/ANS 8.1-1983). No critical benchmark experiment will precisely match the fissile material, moderation, neutron poisoning, and configuration in the actual situation. However, the applicant can perform a proper benchmark analysis by selecting experiments that adequately represent the actual situation and fissionable material features and parameters important to reactivity. Key features and parameters that should be considered in selecting appropriate critical experiments for spent fuel include type of fuel, enrichment, hydrogen to uranium ratio or moderator to uranium ratio for graphite moderated fuel designs (rod diameter and pitch), fuel and cladding chemical composition, reflector, neutron energy spectrum, and poisoning. The applicant must justify the suitability of the critical experiments chosen to benchmark the criticality code and calculations. UCID- 21830 provides guidance for benchmarking and contains a substantial bibliography of benchmark experiments and validation testing.

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Licensing, Design	8.4.4.3	**CRITICALITY EVALUATION** **Applicant Criticality Analysis** Benchmark Comparisons (Part 2 of 2)	72.40, 72.124	Detailed guidance on determining a code bias from benchmark experiments has not been formalized. Multiple applicable benchmark experiments should be analyzed. The results of these benchmark calculations should be converted to a bias for application to the criticality computations. Simply using an average of the biases from a number of benchmark calculations is typically not considered to be sufficient, particularly if one benchmark yields results that are significantly different from the others. Benchmark comparisons must also be checked for bias trends with respect to parametric variations (such as pitch-to-rod-diameter ratio, assembly separation, reflector material, neutron absorber material, etc.). UCID- 21830 provides some guidance for this, but other methods have also been considered appropriate. The calculated statistical uncertainties of both benchmark and cask analyses also need to be addressed for Monte Carlo codes. The uncertainties should be applied to at least the 95% confidence level and 95% probability, as a general rule, if the acceptability of the result depends on small differences between large values. A sufficient number of neutron histories can readily be used so that the treatment of these uncertainties should not significantly affect the results. Only biases determined by benchmark comparisons that increase keff or lower the confidence level should be applied. If the benchmark calculation for a critical experiment results in a neutron multiplication that is greater than unity, it should not be used in a manner that would reduce the k_{eff} calculated for the cask. Critical experiments using a different fissionable isotope than that intended for the ISFSI or MRS should not be included in this benchmark comparison.

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Licensing, Design	8.4.5	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** (Part 1 of 2)	72.40, 72.124	Unirradiated reactor fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transport and storage casks. As the fuel is irradiated in the reactor, the nuclide composition changes. Ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is typically termed burnup credit. Extensive investigations have been performed both within the United States and by other countries in an effort to understand and document the technical issues related to burnup credit. Much of this work has been considered in the development of the U.S. Department of Energy's Topical Report (TR) on Actinide-Only Burnup Credit for Pressurized Water Reactor (PWR) Spent Nuclear Fuel Packages (DOE/RW-0472).
Licensing, Design	8.4.5	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** (Part 2 of 2)	72.40, 72.124	The technical information provided in the literature and in the various TR revisions, together with the initial confirmatory analyses by the U.S. Nuclear Regulatory Commission (NRC) research program, have provided a sufficient basis for the staff to proceed with acceptance of a burnup credit approach in the criticality safety analysis of PWR spent fuel casks as discussed in the Recommendations below. Although insights gained from reviewing the TR submittals form a part of the basis for the staff's position, the NRC has not endorsed the TR or its supporting documentation. The following recommendations provide a cask-specific basis for granting burnup credit, based on actinide composition. The NRC's staff will issue additional guidance and/or recommendations as information is obtained from its research program on burnup credit and as experience is gained through future licensing activities. Except as specified in the following recommendations, the application of burnup credit does not alter the current guidance and recommendations provided by the NRC staff for criticality safety analysis of transport and storage casks.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.5.1	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** Limits for the Licensing Basis	72.40, 72.124	***Recommendation*** : The licensing-basis analysis performed to demonstrate criticality safety should limit the amount of burnup credit to that available from actinide compositions associated with PWR irradiation of UO_2 fuel to an assembly-average burnup value of 40 GWd/MTU or less. This licensing-basis analysis should assume an out-of-reactor cooling time of five years and should be restricted to intact assemblies that have not used burnable absorbers. The initial enrichment of the fuel assumed for the licensing-basis analysis should be no more than 4.0 wt% ^{235}U unless a loading offset is applied. The loading offset is defined as the minimum amount by which the assigned burnup loading value (see Recommendation 8.4.5.5) must exceed the burnup value used in the licensing safety basis analysis. The loading offset should be at least 1 GWd/MTU for every 0.1 wt% increase in initial enrichment above 4.0 wt%. In any case, the initial enrichment shall not exceed 5.0 wt%. For example, if the applicant performs a safety analysis that demonstrates an appropriate subcritical margin for 4.5 wt% fuel burned to the limit of 40 GWd/MTU, then the loading curve (see Recommendation 8.4.5.4) should be developed to ensure that the assigned burnup loading value is at least 45 GWd/MTU (i.e., a 5 GWd/MTU loading offset resulting from the 0.5 wt% excess enrichment over 4.0 wt%). Applicants requesting use of actinide compositions associated with fuel assemblies, burnup values, or cooling times outside these specifications, or applicants requesting a relaxation of the loading offset for initial enrichments between 4.0 and 5.0 wt%, should provide the measurement data and/or justify extrapolation techniques necessary to adequately extend the isotopic validation and quantify or bound the bias and uncertainty.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	8.4.5.2	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** Code Validation	72.40, 72.124	***Recommendation*** : The applicant should ensure that the analysis methodologies used for predicting the actinide compositions and determining the neutron multiplication factor (k-effective) are properly validated. Bias and uncertainties associated with predicting the actinide compositions should be determined from benchmarks of applicable fuel assay measurements. Bias and uncertainties associated with the calculation of k-effective should be derived from benchmark experiments that represent important features of the cask design and spent fuel contents. The particular set of nuclides used to determine the k-effective value should be limited to that established in the validation process. The bias and uncertainties should be applied in a way that ensures conservatism in the licensing safety analysis. Particular consideration should be given to bias uncertainties arising from the lack of critical experiments that are highly prototypical of spent fuel in a cask.
Licensing, Design	8.4.5.3	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** Licensing-Basis Model Assumptions	72.40, 72.124	***Recommendation*** : The applicant should ensure that the actinide compositions used in analyzing the licensing safety basis (as described in Recommendation 8.4.5.1) are calculated using fuel design and in-reactor operating parameters selected to provide conservative estimates of the k-effective value under cask conditions. The calculation of the k-effective value should be performed using cask models, appropriate analysis assumptions, and code inputs that allow adequate representation of the physics. Of particular concern should be the need to account for the axial and horizontal variation of the burnup within a spent fuel assembly (e.g., the assumed axial burnup profiles), the need to consider the more reactive actinide compositions of fuels burned with fixed absorbers or with control rods fully or partly inserted, and the need for a k-effective model that accurately accounts for local reactivity effects at the less-burned axial ends of the fuel region.
Licensing, Design	8.4.5.4	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** Loading Curve	72.40, 72.124	***Recommendation*** : The applicant should prepare one or more loading curves that plot, as a function of initial enrichment, the assigned burnup loading value above which fuel assemblies may be loaded in the cask. Loading curves should be established based on a 5-year cooling time and only fuel cooled at least five years should be loaded in a cask approved for burnup credit.

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Licensing, Design	8.4.5.5	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** Assigned Burnup Loading Value	72.40, 72.124	***Recommendation*** : The applicant should describe administrative procedures that should be used by licensees to ensure that the cask will be loaded with fuel that is within the specifications of the approved contents. The administrative procedures should include an assembly measurement that confirms the reactor record assembly burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For an assembly reactor burnup record to be confirmed, the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by the combined uncertainties in the records and the measurement.

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Licensing, Design	8.4.5.6	**CRITICALITY EVALUATION** **Burnup Credit in the Criticality Analysis** Estimate of Additional Reactivity Margin	72.40, 72.124	***Recommendation*** : The applicant should provide design-specific analyses that estimate the additional reactivity margins available from fission product and actinide nuclides not included in the licensing safety basis (as described in 8.4.5.1). The analysis methods used for determining these estimated reactivity margins should be verified using available experimental data (e.g., isotopic assay data) and computational benchmarks that demonstrate the performance of the applicant's methods in comparison with independent methods and analyses. The Organization for Economic Cooperation and Development Nuclear Energy Agency's Working Group on Burnup Credit provides a source of computational benchmarks that may be considered. The design-specific margins should be evaluated over the full range of initial enrichments and burnups on the burnup credit loading curve(s). The resulting estimated margins should then be assessed against estimates of: (a) any uncertainties not directly evaluated in the modeling or validation processes for actinide-only burnup credit (e.g., k-effective validation uncertainties caused by a lack of critical experiment benchmarks with either actinide compositions that match those in spent fuel or material geometries that represent the most reactive ends of spent fuel in casks); and (b) any potential nonconservatisms in the models for calculating the licensing-basis actinide inventories (e.g., any outlier assemblies with higher-than-modeled reactivity caused by the use of control rod insertion during burnup).
Licensing, Design	9.4	**CONFINEMENT EVALUATION** **Acceptance Criteria**	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	This section identifies the acceptance criteria used for the confinement review. Two types of criteria are described. The first identifies the type of descriptive and analytical information and level of detail related to confinement evaluation that should be present in the Safety Analysis Report (SAR). The second identifies particular standards that are accepted by the NRC staff when conducting confinement analysis. Compliance with the first criteria provides the information that allows the NRC staff to develop a detailed understanding of the applicant's estimate of the effectiveness of the radionuclide confinement systems under a broad spectrum of conditions.

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Licensing, Design	9.4.1	**CONFINEMENT EVALUATION** **Confinement Description**	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	No text provided.
Licensing, Design	9.4.1.1	**CONFINEMENT EVALUATION** **Confinement Description** Confinement Casks or Systems	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	The application must describe the confinement system for spent fuel systems or high-level waste. The confinement design must be consistent with the regulatory requirements, as well as the applicant's "General Design Criteria" reviewed in Chapters 4 and 5 of this SRP. The NRC staff has accepted construction of the primary confinement barrier in conformance with Section III, Subsections NB or NC, of the Boiler and Pressure Vessel (B&PV) Code of the American Society of Mechanical Engineers (ASME). This code defines the standards for all aspects of construction including materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components. In such instances, the staff has relied upon Section III to define the minimum acceptable margin of safety; therefore, the applicant must fully document and completely justify any deviations from the specifications of Section III. In some cases after careful and deliberate consideration, the staff has made exceptions to this requirement.
Licensing, Design	9.4.1.2	**CONFINEMENT EVALUATION** **Confinement Description** Pool and Waste Management Facilities	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	A description of the confinement system for pool and waste management facilities must be presented in the SAR. Important design features associated with the control and confinement of radioactive materials include seals on closures and doors, negative pressure design, ventilation/filtration systems, charcoal beds, holdup volumes, etc. Instrumentation and control features may include detectors for monitoring releases, alarms, and control features to mitigate releases if abnormal conditions are detected.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.2	**CONFINEMENT EVALUATION** **Radionuclide Confinement Analysis**	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	Confinement analysis is concerned with the release of radioactive materials to the environment for normal operations and anticipated occurrences and for accident conditions including design basis accidents. The SAR must present a clear description of the proposed confinement system as either (1) a sealed system, as is the case in most spent fuel storage systems, or (2) a vented system with off-gas treatment systems, as is often the case in pools or waste management systems. The description must state how the confinement systems would respond during anticipated occurrences or accident conditions (both design basis and less than design basis). Estimates of releases should be based on the quantity of radioactive material such as vapor pressure, particle sizes, and adsorption kinetics and equilibrium. Data sources that are used to support the physical property estimates or release quantities should be identified in the SAR. Air and water effluents associated with normal operations must comply with average monthly concentration limits specified in Appendix B to 10 CFR 20 Sections 20.1001-20.2401. There are however, performance standards in 10 CFR Part 72 that place limits on the dose to individuals at or beyond the controlled area. Source terms developed as part of the confinement evaluation review are used for evaluating compliance with the performance standards. The assessment of performance under 10 CFR Part 72.104(a) is conducted in accordance with guidance in this Chapter and Chapter 11 (Radiation Protection Evaluation) of this FSRP. The assessment of performance relative to 72.106(b) is conducted in accordance with guidance in this Chapter and Chapter 15 (Accident Evaluation) of this FSRP.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.2.1	**CONFINEMENT EVALUATION** **Radionuclide Confinement Analysis** Confinement Casks or Systems	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	The application must identify the amount of radionuclides that would be released to the environment for normal operations, a spectrum of anticipated occurrences, and a spectrum of design basis accidents. In developing estimates of materials released to the environment, the staff uses guidance in the Cask Standard Review Plan (NUREG-1536)] and applicable ISGs, as estimates of material available for release following the failure of individual fuel pins when there are no additional forces that would move material out of the fuel pin structure. This information about material available for release must be used together with information about a specific release scenario to develop a release amount estimate. This guidance about fraction available for release is only for spent Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fuel with uranium in the form of UO_2. For purposes of estimating the source term at the time of retrieval operations, NRC has accepted the assumptions that casks will have experienced an off-normal condition (e.g., 10 percent rod failure). For storage casks having closure lids that are designed and tested to be “leak tight,” as defined in “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials,” ANSI N14.5-1997, confinement calculation of the doses under normal off-normal and accident conditions are unnecessary. For casks that are not tested to the leak tight standard of ANSI N14.5-1997, alternative justifications may be found acceptable to the staff.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.2.2	**CONFINEMENT EVALUATION** **Radionuclide Confinement Analysis** Pool and Waste Management Facilities (Part 1 of 2)	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	Systems that do not have sealed barriers to provide confinement (i.e., transfer pools or cells, waste management facilities) may have releases to the environment under normal, off-normal, and accident conditions. The SAR must present estimates of radionuclides released to the environment for normal conditions, anticipated occurrences, and design basis accidents. The estimates must be based on evaluation of the actual design and the physical process that will move radionuclides into the environment or retain them in the storage or holding systems. If the assumptions about material available for release that are used in the SAR analysis are different from those in NUREG-1536 and applicable ISGs, such assumptions must be justified. The SAR must include an estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal ISFSI or MRS operations. Because use of the pool facility may be intermittent, the estimated quantities of releases must be projected for the maximum usage year, typical years, and for standby (or shutdown) mode years. The estimated source term should also consider the possibility that radioactive gas from a failed sealed fuel container could be released from the facility under anticipated occurrences.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.2.2	**CONFINEMENT EVALUATION** **Radionuclide Confinement Analysis** Pool and Waste Management Facilities (Part 2 of 2)	72.24, 72.44, 72.104, 72.106, 72.122, 72.126, 72.128	The SAR must estimate pool and waste management facility emissions resulting from anticipated occurrences (off-normal conditions), including possible emissions of radioactive gases from sealed fuel containers that may fail. The SAR must also determine any pool and waste management facility emissions which may result from design basis accidents (accident level conditions). The NRC accepts that other sources on the site can be assumed to be at normal conditions during such accident conditions unless the same initiating event affects these other sources. Estimates of radionuclide quantities released after failure of fuel cladding can take credit for radionuclides being retained in the water in which fuel rods are immersed. The NRC accepts guidance included in Regulatory Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors,” for estimation of releases from fuel rods in pools.
Licensing, Design	9.4.3	**CONFINEMENT EVALUATION** **Confinement Monitoring**	72.24, 72.44, 72.122, 72.126, 72.128	Confinement monitoring for ISFSI and MRS has two aspects. The first is monitoring storage confinement closure seals or overall closure effectiveness. The second is providing a system to measure radionuclides released to the environment under normal and accident conditions. This second aspect includes all areas where there is the potential for significant releases to the environment and may include storage casks, pool facilities, and waste management facilities. The SAR must present a discussion of the extent of monitoring required consistent with 10 CFR Part 72 requirements for both of these aspects of confinement monitoring.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.3.1	**CONFINEMENT EVALUATION** **Confinement Monitoring** Dry Storage Cask Confinement Systems	72.24, 72.44, 72.122 (h)(4), 72.126, 72.128	The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. In instances involving welded closures, the staff has previously accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored. However, the lack of a closure monitoring system has typically been coupled with a periodic surveillance program that would enable the licensee to take timely and appropriate corrective actions to maintain safe storage conditions if closure degradation occurred. However, for storage casks having closure lids that are designed and tested to be “leak tight,” as defined in “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials,” ANSI N14.5-1997, monitoring capability and/or surveillance plans are unnecessary. To show compliance with 10 CFR Part 72.122(h)(4), cask vendors have proposed, and the staff has accepted, routine surveillance programs and active instrumentation to meet the continuous monitoring requirements. Some DCSS designs contain a component or feature whose continued performance over the licensing period has not been demonstrated to staff with a sufficient level of confidence. Therefore, the staff may determine that active monitoring instrumentation is required to provide for the detection of component degradation or failure. This particularly applies to components whose failure immediately affects or threatens public health and safety. To demonstrate compliance with 10 CFR Part 72.122(h)(4), the vendor or staff may propose a technical specification requiring such instrumentation as part of the initial use of a cask system. After initial use, and if warranted and approved by staff, such instrumentation may be discontinued or modified.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.3.2	**CONFINEMENT EVALUATION** **Confinement Monitoring** Effluents	72.24, 72.44, 72.122, 72.126, 72.128	The SAR must describe the monitoring system that provides measurement of releases under normal and accident conditions. The discussion must address all areas of the ISFSI that can release radionuclides into the environment. NRC accepts the following criteria and guidance for monitoring releases from ISFSI or MRS systems, to the extent applicable: • Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis” • Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants” • Regulatory Guide 4.1, “Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants” • NUREG-0800, “Standard Review Plan,” 11.5, “Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems”
Licensing, Design	9.4.4	**CONFINEMENT EVALUATION** **Protection of Stored Materials from Degradation**	72.24, 72.122	The materials that help confine the radionuclides in spent fuel and waste should be protected from degradation so that confinement effectiveness is not reduced. The SAR must identify these materials (i.e., fuel matrix and fuel cladding) and describe how these material are protected from degradation.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.4.1	**CONFINEMENT EVALUATION** **Protection of Stored Materials from Degradation** Confinement Casks or Systems	72.24, 72.122	The primary materials in spent fuel that must be protected from degradation are the fuel matrix and fuel cladding. The applicant's SAR must describe the actions proposed to protect these materials from degradation. The cask must provide a non-reactive environment to protect fuel assemblies against fuel cladding degradation, which might otherwise lead to gross rupture (Pacific Northwest Laboratories [PNL] 6365). Measures for providing a non-reactive environment within the confinement cask typically include drying, evacuating air and water vapor, and backfilling with a non-reactive cover gas (such as helium). For dry storage conditions, experimental data have not demonstrated an acceptably low oxidation rate for UO2 spent fuel, over the 20-year licensing period, to permit safe storage in an air atmosphere. Therefore, to reduce the potential for fuel oxidation and subsequent cladding failure, the NRC has accepted storage designs that have cask inventories of oxidizing gases less than 1.0 gram mole per cask and an inert atmosphere (e.g., helium cover gas) for storing UO2 spent fuel in a dry environment. Note that other fuel types, such as graphite fuels for the high-temperature gas-cooled reactors (HTGRs), may not exhibit the same oxidation reactions as UO_2 fuels, and therefore, may not require an inert atmosphere. Applicants proposing to use atmospheres other than inert gas should discuss how the fuel and cladding will be protected from oxidation.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	9.4.4.2	**CONFINEMENT EVALUATION** **Protection of Stored Materials from Degradation** Pool and Waste Management Facilities	72.24, 72.122	The SAR must also describe the pool and waste management facilities proposed by the applicant to prevent degradation of waste and fuel confinement materials. Pools must provide an environment that is compatible with stored materials and any elements important to safety. The SAR must give full consideration to maximum anticipated storage time for any projected corrosion. Permanent degradation of any pool confinement barrier should not occur for anticipated occurrences (off-normal events and conditions) when considering the cumulative corrosion effects over the proposed license period. The pool facility confinement barrier and liquid containment structures, systems, and components (SSCs) may experience some repairable degradation from accident-level conditions.
Licensing, Operation	10.4	**CONDUCT OF OPERATIONS EVALUATION** **Acceptance Criteria**	72.24, 72.28, 72.40, 72.180, 72.184, 72.190, 72.192, 72.194	Acceptance criteria for review of proposed conduct of operations include those pertaining to (a) the completeness of information submitted by the applicant describing the plan of operations, and (b) the adequacy of scope and content of the plan. The following sections provide criteria for each of these categories.
Licensing, Operation	10.4.1	**CONDUCT OF OPERATIONS EVALUATION** **Organizational Structure**	72.24, 72.28, 72.40	The application must describe the organizational structure and administrative control system that will be used for the proposed ISFSI or MRS (i.e., through construction, preoperational testing and initial operations, normal operations, and decommissioning).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.1.1	**CONDUCT OF OPERATIONS EVALUATION** **Organizational Structure** Corporate Organization	72.24, 72.28, 72.40	The application must describe the corporate organization responsible for the ISFSI or MRS installation, including organization charts and position descriptions. If the corporation is made up from two or more corporate identities, the relationship and responsibilities between each entity should be explained. The financial capabilities of the corporation to construct, operate, and decommission the installation must be demonstrated. The application must describe the corporate functions, responsibilities, and authorities related to each aspect of the installation (design, engineering, construction, quality assurance, testing, etc.). The in-house organization and technical staff (numbers of personnel, qualifications, educational and experience backgrounds, etc.) must be described. The relationship between the applicant's in-house organization and outside contractors and suppliers, including the extent of dependence on those sources for design, construction, quality assurance and other functions, must be described. The relationship between the corporate and onsite organizations must also be described. The nature of interaction between corporate management and the site related to health and safety, including any role in policy/procedure development, audits, inspections, and investigations, must be explained.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.1.2	**CONDUCT OF OPERATIONS EVALUATION** **Organizational Structure** Onsite Organization	72.24, 72.28, 72.40	The application must describe site organization, including organizational charts and position descriptions with emphasis on positions that perform functions important to safety. Such positions include, but are not limited to, those with responsibilities in health physics, nuclear criticality safety, training and certification, emergency planning and response, operations, maintenance, engineering, and quality assurance. The discussion of positions and responsibility must illustrate how these functions or aspects of these functions, including the degree of separation between the facility operations organization and other parts of the onsite organization that perform functions important to safety, are performed. The application must also identify alternates who are authorized to act in the absence of individuals assigned to key positions and identify which positions have shutdown or stopwork authority for health or safety reasons. The application must identify minimum staffing levels for major entities within the onsite organization. The application must identify whether the onsite organization includes a safety committee (or committees). The membership, duties, responsibilities, operating characteristics and reporting function of proposed safety committees must be described.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.1.3	**CONDUCT OF OPERATIONS EVALUATION** **Organizational Structure** Management and Administrative Controls (Part 1 of 3)	72.24, 72.28, 72.40	The application must describe the proposed management and administrative control system, including provisions for: • Administrative and general plant procedures • A program of surveillance, testing, and inspections of items and activities important to safety • Periodic independent audits • Change control • Employee training and certification programs • Records preparation and maintenance Administrative procedures address the process of planning, administrative controls, and document issuance; and provide rules and instructions on personnel conduct, preparation and retention of plant documents, and interfaces among plant organizations. General plant procedures are those that prescribe the actions required to achieve safe operation and provide necessary instruction for the operation and maintenance of plant systems and equipment. The application must describe the program for preparation, review, change, and approval of procedures. The identity of the onsite organizations that use procedures and the activities or operations that are covered by such procedures, must also be specified. (See Section 10.4.3.1 below for guidance on procedures for normal plant operation.)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.1.3	**CONDUCT OF OPERATIONS EVALUATION** **Organizational Structure** Management and Administrative Controls (Part 2 of 3)	72.24, 72.28, 72.40	The applicant must describe the program of surveillance, testing, and inspection to ensure satisfactory in-service performance of items and activities important to safety. The description must address the development and use of procedures that set forth the steps to be taken and identify the standards or criteria to be applied. The program must include provisions for: • Pre-operational testing (see Section 10.4.2.1) to demonstrate plant operability and identify conditions adverse to safety • Operational testing and surveillance to verify and record characteristics of plant equipment and components • Surveillance, testing, and inspection after modification or when corrective actions have been completed The management control system description must also include requirements for planned and scheduled internal and external audits to evaluate the application and effectiveness of management controls, plant procedures, and other activities affecting safety. The audit program must describe audit frequency, methods for documenting and communicating audit findings, resolution of issues, and implementation of corrective actions.
Licensing, Operation	10.4.1.3	**CONDUCT OF OPERATIONS EVALUATION** **Organizational Structure** Management and Administrative Controls (Part 3 of 3)	72.24, 72.28, 72.40	The system for change control, including how change control is integrated into the management control system, must also be described. The coordination of change between potentially affected organizations (engineering, operations, maintenance, training, etc.) must be described. The application must describe how operations are shut down to effect changes and how all plant equipment and procedural changes are completed. The training of staff before resumption of operations must also be addressed. The management system description must also include the system for maintaining records of facility operation (as addressed in Section 10.4.3.2.)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.2	**CONDUCT OF OPERATIONS EVALUATION** **Preoperational Testing and Startup Operations**	72.24, 72.40	The Safety Analysis Report (SAR) must describe the plans for preoperational testing and initial facility (startup) operations. Guidance on the SAR informational content related to preoperational testing and startup operations is provided by Regulatory Guide 3.48, Section 9.2. The following guidance summarizes and supplements that provided by Regulatory Guide 3.48. Classification of preoperational testing and operating startup are typically delineated by the receipt of radioactive material to be stored. In cases where an ISFSI or MRS is to be located at a site with a pre-existing pool, reoperational testing would occur before any withdrawal of the subject radioactive material from the pool for placement into storage. The administrative procedures used for conducting the testing and startup must be described. This description must include the system to be used for preparing, approving and executing the test procedures and for evaluating, documenting, and approving test results. Provisions must be made for incorporating changes to the system or individual procedures on the basis of inadequacies in test procedures or unexpected test results. The organizational responsibilities for administering the system must be identified, and the qualifications of involved personnel must be described.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.2.1	**CONDUCT OF OPERATIONS EVALUATION** **Preoperational Testing and Startup Operations** Preoperational Testing Plan	72.24, 72.40	The test program description must include an identification of testing objectives and the general methods to be used to meet those objectives. The SAR must identify each item (facility, component, piece of equipment, operation) to be tested. For each physical or operational item, the following information must be provided: • The type of test to be performed • The expected response • The acceptable margin of difference from the expected response • The method of validation (if applicable) • Appropriate corrective action for unexpected or unacceptable results If the proposed ISFSI or MRS contains any SSCs important to safety the functional adequacy or reliability of which has not been demonstrated by prior use or otherwise validated, the preoperational test plan must include a description and schedule showing how these safety questions will be resolved before the initial receipt of the radioactive materials to be stored. The applicant must commit to performing a dry run (cold test) of each operation involving the radioactive material to be stored and to use the results of these tests to make necessary changes to equipment and procedures. There must also be a commitment to conduct routine full load tests of any equipment that is to carry spent fuel or high-level waste containers.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.2.2	**CONDUCT OF OPERATIONS EVALUATION** **Preoperational Testing and Startup Operations** Operating Startup Plan	72.24, 72.40	The operating startup plan must identify those specific operations involving the initial handling of radioactive material to be placed into storage. Although plant procedures to be used for normal operations or during steady-state conditions are not necessarily included in the operating startup plan, the evaluation of the effectiveness of those procedures are elements of the operating startup plan. For as low as is reasonably achievable (ALARA) considerations, as many of the operating startup actions as feasible must be performed during preoperational testing (i.e., before sources of exposure are present). The operating startup plan must include the following elements: • Tests and confirmation of procedures and exposure times involving actual radioactive sources (e.g., radiation monitoring, in-pool operations) • Direct radiation monitoring of casks and shielding for radiation dose rates, streaming, and surface "hot-spots" • Verification of effectiveness of heat removal features • Documentation of results of tests and evaluations
Licensing, Operation	10.4.3	**CONDUCT OF OPERATIONS EVALUATION** **Normal Operations**	72.24, 72.28, 72.40	Regulatory Guide 3.48, Section 9.4, provides the primary guidance related to SAR information on procedures and recordkeeping in support of normal operations. The guidance in this section summarizes and supplements the guidance provided in Regulatory Guide 3.48.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.3.1	**CONDUCT OF OPERATIONS EVALUATION** **Normal Operations** Procedures	72.24, 72.28, 72.40	The SAR must describe the applicant's commitment to conduct all operations that are important to safety according to written procedures and to have proposed procedures and revisions reviewed and approved by the health, safety, and quality assurance organizations that are independent of the operating management function. The identification of proposed written procedures must include all routine and projected contingency operations. The applicant must also describe the review, change, and approval practices for all operating, maintenance, and testing procedures. This description may refer to the appropriate management controls addressed in Section 10.4.1.2. The listing of operations requiring written procedures must include, as applicable to the ISFSI or MRS: • All operations identified in the proposed technical specifications • Operating, maintenance, testing, and surveillance functions important to safety The procedures listed must clearly indicate, by title or subject, their purpose and applicability. The applicant should identify any standards used for the preparation of these procedures.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.3.2	**CONDUCT OF OPERATIONS EVALUATION** **Normal Operations** Records (Part 1 of 4)	72.24, 72.28, 72.30 (d)(3), 72.40, 72.156, 72.174	The management system for maintaining records must be described. This description may refer to the appropriate management controls addressed in Section 10.4.1.2. Although all records need not be maintained centrally, the management system must ensure that cognizance is being maintained of all records, the responsible staff, and locations. Records stored in electronic media will generally be acceptable if the capability is maintained to produce legible, accurate, and complete records over the required retention period. The record format must include all pertinent information, such as stamps, initials, and signatures. The retention period for each type of record, because it varies depending on applicable regulatory requirements, must be specified. The management system must also provide for adequate safeguards against tampering and loss of records over the retention period. The SAR must identify, by type, the records to be maintained. Records maintained must include: • Construction records, as specified in applicable construction codes (e.g., American Concrete Institute [ACI] 349) and including as-built drawings and specifications, material certifications and audit trail to the incorporating SSCs, inspection records, test reports, and certifications (per 10 CFR 72.30(d)(3), 72.156, 72.174)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.3.2	**CONDUCT OF OPERATIONS EVALUATION** **Normal Operations** Records (Part 2 of 4)	20.1003, 72.24, 72.28, 72.30 (d)(1), (d)(3), and (d)(4), 72.40, 72.44, 72.72 (a), (b), (c) and (d) 72.174	• As required by 10 CFR 72.30(d)(3), a list contained in a single document and updated no less than every 2 years of the following: — All areas designated and formerly designated as restricted areas as defined under 10 CFR 20.1003 — All areas outside of restricted areas that require documentation under 10 CFR 72.30(d)(1) (see next entry) • Records of spills or other abnormal occurrences involving the spread of radiation in and around the facility, equipment, or site (per 10 CFR 72.30(d)(1)) • Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning and records of the funding method used for ensuring funds, if either funding plan or certifications are used (per 10 CFR 72.30(d)(4)) (i.e., record copy of proposed decommissioning plan filed with license application, attached decommissioning funding plan, any modifications to these plans, and final decommissioning plan when prepared) • Receipt, inventory, disposal, acquisition, and transfer of all spent fuel and high-level radioactive waste in storage, as required by 10 CFR 72.72(a) (including provisions for duplicate records storage at different locations, per 10 CFR 72.72[d]) • Records of physical inventories and current material control and accounting procedures (per 10 CFR 72.72[b] and [c]) • Operating records, including principal maintenance, alternations, or additions made • Records of off-normal occurrences and events associated with radioactive releases • Records of employee certification (per 10 CFR 72.44) • Quality Assurance (QA) records (per 10 CFR 72.174)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.3.2	**CONDUCT OF OPERATIONS EVALUATION** **Normal Operations** Records (Part 3 of 4)	20 Subpart L, 72.24, 72.28, 72.40, 72.44(e), 72.92, 72.186, 73.21, 73.70	• Environmental survey records and environmental reports • Radiation monitor readings or records (e.g., stripcharts or electronic results) • Radiation protection program records (per 10 CFR 20, Subpart L), including those related to: — Program contents, audits, and reviews — Radiation surveys — Determination of prior occupational dose — Planned special exposures — Individual (worker) monitoring results — Dose to individual members of the public — Radioactive waste disposal — Tests of entry control devices for very high radiation areas • Records of changes to the physical protection plan (per 10 CFR 72.44(e) and 72.186), and other physical protection records (per 10 CFR 73.21 and 73.70) • Records of occurrence and severity of natural phenomena (10 CFR 72.92)
Licensing, Operation	10.4.3.2	**CONDUCT OF OPERATIONS EVALUATION** **Normal Operations** Records (Part 4 of 4)	20 Subpart L, 72.24, 72.28, 72.40, 72.70, 72.74, 72.76, 72.78, 72.82, 72.180, 73.71,	• Record copies of: — SAR, SAR updates, FSAR (10 CFR 72.70) — Reports of accidental criticality or loss of special nuclear material (10 CFR 72.74 and 10 CFR 73.71) — Material status reports (10 CFR 72.76) — Nuclear material transfer reports (10 CFR 72.78) — Physical security plan (per 10 CFR 72.180) — "Other" records and reports (per 10 CFR 72.82) — Report of preoperational test acceptance criteria and test results — Written procedures The radiation protection records required by 10 CFR 20 Subpart L must incorporate the units of curie, rad, and rem, as applicable, including multiples or subdivisions of those units (e.g., megacurie, millicurie, millirem, etc.). Where dose is part of a record, the dose quantity used on the record (e.g., total effective dose equivalent, committed effective dose equivalent, shallow dose equivalent, etc.) must be clearly indicated.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.4	**CONDUCT OF OPERATIONS EVALUATION** **Personnel Selection, Training, and Certification**	72.24, 72.28, 72.40, 72.190, 72.192, 72.194	The application must describe the organization responsible for personnel selection, training, and certification. The program that will be established and implemented to ensure that personnel whose responsibilities include functions that are important to safety will be appropriately qualified and trained must also be described. The process of selecting and training security guards must be described.
Licensing, Operation	10.4.4.1	**CONDUCT OF OPERATIONS EVALUATION** **Personnel Selection, Training, and Certification** Personnel Organization	72.24, 72.28, 72.40, 72.190, 72.192, 72.194	The description must include a discussion of the organization and management of the training component, and must identify the personnel responsible for development of training programs, conducting training and retraining of employees (including new employee orientations), and maintaining up-to-date records on the status of trained personnel.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.4.2	**CONDUCT OF OPERATIONS EVALUATION** **Personnel Selection, Training, and Certification** Selection and Training of Operating Personnel (Part 1 of 2)	72.24, 72.28, 72.40, 72.190, 72.192, 72.194	The applicant must identify the functions that are important to safety and describe the qualifications for personnel performing those functions. These personnel qualifications must include: • Minimum qualification requirements for operating, technical, and maintenance supervisory personnel • Qualifications, in resumé form, of persons who will be assigned to managerial and technical positions The program description must identify the scope of operational and safety training. Operational training must include topics such as installation design and operations, instrumentation and control, methods of dealing with operating functions, decontamination procedures, and emergency procedures. Radiation safety training must include topics such as nature and sources of radiation, methods of controlling exposure and contamination, radiation monitoring, shielding, dosimetry, biological effects, and criticality hazards control. The type and level of training to be provided for each job description (personnel classification), including specific training provided to specific job descriptions, must be listed. Alternatively, the basis used to identify the type and level of training by job description may be described. The requirements for certification of personnel who will operate equipment and controls that are important to safety must be clearly identified. The requirements must address the physical condition and general health of personnel to be certified in accordance with 10 CFR 72.194.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.4.2	**CONDUCT OF OPERATIONS EVALUATION** **Personnel Selection, Training, and Certification** Selection and Training of Operating Personnel (Part 2 of 2)	72.24, 72.28, 72.40, 72.190, 72.192, 72.194	The methods of testing to determine the effectiveness of the training program must be described. Effectiveness must be determined by evaluation against established objectives and criteria, and any standards used for development and implementation of the training program must be identified. The frequency of retraining, and the nature and duration of retention of training and testing records, must be described. Retraining must be periodic and not less than every 2 years. Training records must be kept up-to-date and retained for a minimum of 3 years. Implementation of the training program before conduct of operations involving radioactive material (i.e., preoperational training) must be described. The applicant must commit to substantial completion of staff training and certification before receipt of the radioactive material to be stored. The applicant should identify any standards used for selection, training, and certification of personnel.
Licensing, Operation	10.4.4.3	**CONDUCT OF OPERATIONS EVALUATION** **Personnel Selection, Training, and Certification** Selection and Training of Security Guards	72.24, 72.28, 73.55 (b)(4)(ii) and 73 Appendix B	The process by which security guards (including watchmen, armed response persons, etc.) will be selected and qualified must be described as required by 10 CFR 73.55(b)(4)(ii). This information may be submitted as part of the applicant's physical security plan, as addressed in Section 10.4.6. The criteria used must conform to the general criteria for security personnel contained in 10 CFR 73, Appendix B. Regulatory Guide 5.20, "Training, Equipping, and Qualifying of Guards and Watchmen," provides guidance in this area.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.5	**CONDUCT OF OPERATIONS EVALUATION** **Emergency Planning** (Part 1 of 2)	72.24, 72.32, 72.40	If the proposed installation is not on the site of an NRC-licensed reactor, the application must include an emergency plan that complies with 10 CFR 72.32(a) (for an ISFSI) or 10 CFR 72.32(b) (for an MRS or an ISFSI that may process or repackage spent fuel). The emergency plan may be incorporated by reference into the SAR. The SAR must include descriptive information on the applicant's plans for coping with emergencies, as required by the applicable paragraph of 10 CFR 72.32. Because it is possible that structures, systems and components important to safety, including the confinement boundary, could be damaged or fail by a means that has not been considered, the SAR must describe the applicant's ability to detect accident events or damage to SSCs caused by conditions not analyzed in the SAR.
Licensing, Operation	10.4.5	**CONDUCT OF OPERATIONS EVALUATION** **Emergency Planning** (Part 2 of 2)	72.24, 72.32, 72.40	Information provided in the SAR may be limited to the following: • A statement that the outline and content of the emergency plan submitted with the license application is in accordance with the requirements of 10 CFR 72.32(a) or (b), as applicable • Identification of types of radioactive material accidents provided for by the emergency plan (these must include or encompass all accident level events or conditions addressed in the SAR "Accident Analyses") • Identification of offsite response organizations provided opportunity to comment on the plan (under 10 CFR 72.32(a)(14) or (b)(14)) and summary of responses • Identification of organizations with whom arrangements have been made for offsite assistance (under 10 CFR 72.32(a)(15) or (b)(15)) Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," contains the principal guidance on preparation of emergency plans for ISFSI and MRS installations.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	10.4.6	**CONDUCT OF OPERATIONS EVALUATION** **Physical Security and Safeguards Contingency Plans** (Part 1 of 2)	72.24, 72.40, 72.180, 72.184 73	The application must contain a physical security plan as required by Section 72.180 and a safeguards contingency plan as required by Section 72.184. The security plan must describe how the applicant will comply with the applicable requirements of 10 CFR 73. The plan must provide for physical security of materials during transportation to and from the ISFSI or MRS, as well as during the storage period. The plan must establish a security organization and include: • Physical protection design features • Safeguard contingency plan • Guard training plan • Tests, inspections, audits, and other means to demonstrate compliance
Licensing, Operation	10.4.6	**CONDUCT OF OPERATIONS EVALUATION** **Physical Security and Safeguards Contingency Plans** (Part 2 of 2)	72 Subpart H, 72.24 (o), 73 Appendix C	If the application is from DOE, the SAR must include: (a) a description of the physical security plan for protection against radiological sabotage (as required by 10 CFR 72 Subpart H), and (b) a certification that it will provide safeguards at the ISFSI or MRS that meet the requirements for comparable surface DOE facilities (required by 10 CFR 72.24[o]). The safeguards contingency plan must comply with the format and content requirements of 10 CFR 73, Appendix C. An acceptable plan must contain: (a) a predetermined set of decisions and actions to satisfy stated objectives, (b) an identification of the data, criteria, procedures, and mechanisms necessary to efficiently implement the decisions, and (c) a stipulation of the individual, group, or organizational entity responsible for each decision and action. Regulatory Guide 5.55, “Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities,” provides guidance on safeguards contingency plans that are specifically applicable to ISFSI- and MRS-type facilities.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	11.4	**RADIATION PROTECTION EVALUATION** **Acceptance Criteria**	20.1101, 20.1201, 20.1301, 20.1302, 20.1406, 20.1501, 20.1701, 20.1702, 72.24, 72.104, 72.106, 72.126	This section describes the acceptance criteria used for review of radiation protection features and programs. These criteria are organized according to the areas specified in Section 11.2. The reviewer should note that some overlap exists between acceptance criteria for radiation protection and those related to shielding (Chapter/Section 7), confinement (Chapter/Section 9), and site-generated waste (Chapter/Section 14).
Licensing, Operation	11.4.1	**RADIATION PROTECTION EVALUATION** **As Low As Reasonably Achievable (ALARA) Considerations**	20.1101, 20.1406, 20.1501, 20.1702, 72.24, 72.104, 72.126	This element should include a description of the proposed program for maintaining exposures to workers and the public ALARA. An ALARA policy statement and the manner in which equipment design and layout and operational features contribute to the ALARA objective should be provided in the license application.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.1.1	**RADIATION PROTECTION EVALUATION** **As Low As Reasonably Achievable (ALARA) Considerations** ALARA Policy and Program (Part 1 of 2)	20.1101, 20.1406, 20.1501, 20.1702, 72.24, 72.104, 72.126	As a minimum, the policy, program, and activities for ensuring that radiation exposures will be ALARA should include the elements described below. Acceptable guidance on the development of an ALARA program is provided in Regulatory Guide 8.10, Revision 1R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable." ***Policy Statement :*** The application should include a written policy that states management's commitment to maintain exposures to workers and the public ALARA. The policy should include provisions that: • No practice involving radiation exposure will be undertaken unless its use produces a net benefit. • All exposures will be kept ALARA with technological, economic, and social factors considered. • Exposures to individuals will not exceed the limits recommended for the appropriate circumstances. • Supervisors will integrate appropriate radiation protection controls into all work activities. • Individuals will be appropriately instructed in the ALARA program. • There will be strict compliance with all regulatory requirements regarding procedures, radiation exposures, and releases of radioactive materials. • A comprehensive program will be maintained to ensure that both individual and collective doses are ALARA.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.1.1	**RADIATION PROTECTION EVALUATION** **As Low As Reasonably Achievable (ALARA) Considerations** ALARA Policy and Program (Part 2 of 2)	20.1101, 20.1406, 20.1501, 20.1702, 72.24, 72.104, 72.126	***ALARA program organizaton*** : This element should include the organizational structure of the ALARA program and the responsibilities and activities of ALARA personnel. ***ALARA program elements*** : Information should be provided to document how implementation of the program will ensure that ALARA objectives are achieved. ALARA program elements should include the use of: • Procedures and engineering controls to minimize dose (detailed in Section 11.4.1.2). • Tracking of individual doses to identify trends and causes and use of these data in the development of alternative procedures that can result in lower doses (detailed in Section 11.4.1.3). • Periodic training and exercises for management, radiation workers, and other site workers in radiation protection, operating procedures, and emergency response (see Section 11.4.1.3).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.1.2	**RADIATION PROTECTION EVALUATION** **As Low As Reasonably Achievable (ALARA) Considerations** Design Considerations	20.1101, 20.1406, 20.1501, 20.1702, 72.24, 72.104, 72.126	The applicant's discussion of facility design and layout should demonstrate consideration of ALARA principles. The design criteria (presented in Section 3 of the SAR) should include ALARA criteria, and the documentation should identify choices between otherwise comparable alternatives affected by ALARA considerations. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," should be used for ALARA design guidance, although specific alternative approaches may be used if clearly indicated. Examples of ALARA design considerations include the following: • Engineered design features that minimize the amount of time that maintenance, health physics, or inspection personnel must stay in restricted areas • Provisions for use of remotely operated or robotic equipment such as welders, wrenches, radiation monitors, etc. • Use of closed-circuit television to monitor for possible blockage of air cooling passages to perform inspections, etc. • Provisions for remote placement and use of temporary shielding. • Incorporation of materials and design features that minimize the potential for accumulation of radioactive materials or surface contamination and that facilitate decontamination and decommissioning. • Incorporation of proven ALARA design alternatives used at other ISFSIs, pool facilities, or waste management facilities. • Placement of occupiable areas (e.g., office, security, or laboratory facilities) away from radiation sources. • Provisions for ALARA and health physics training facilities and equipment.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.1.3	**RADIATION PROTECTION EVALUATION** **As Low As Reasonably Achievable (ALARA) Considerations** Operational Considerations	20.1101, 20.1406, 20.1501, 20.1702, 72.24, 72.104, 72.126	The description of proposed operations should reflect incorporation of ALARA principles in operational procedures. Detailed plans and procedures should be developed in accordance with Regulatory Guides 1.33, “Quality Assurance Program Requirements,” 8.8, and 8.10, and should consider, to the extent practical: • Tradeoffs between requirements for increased monitoring or maintenance activities (and the increased exposures that would result) and the potential hazards associated with reduced frequency of these activities. • Performance of cask preparation efforts (for loading) away from the pool or dry transfer facility. • Sequencing the placement of spent fuel in a manner that maximizes shielding by storage casks or structures. • Dry runs to develop proficiency in procedures involving radiation exposures, to determine exposures likely to be associated with specific procedures, and to consider alternative procedures to minimize exposures. • Inclusion of tested contingency procedures for potential off-normal occurrences. • Consideration of ALARA operational alternatives based on experience with other ISFSIs, pool facilities, or waste management facilities. • Operations research on procedures, types of tools and instruments, and personal protective equipment to minimize exposure times or the effects of exposure.
Licensing, Design	11.4.2	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features**	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	This section identifies elements that indicate whether adequate attention has been paid to radiation protection in the design of the ISFSI or MRS. There is considerable overlap between radiation protection and ALARA criteria in this regard. Applicable guidance on criteria and SAR contents for ISFSI or MRS radiation protection design features is provided in Regulatory Guide 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)” (Section 7.1.2), and NUREG-0800, Sections 12.3 - 12.4, “Radiation Protection Design Features.”

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.2.1	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Installation Design Features (Part 1 of 2)	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	Installation design features for radiation protection are listed separately below as they apply to minimizing offsite and onsite exposures. Features that specifically minimize offsite exposures include: • Siting considerations -- Site location away from population centers to the extent feasible, consistent with other factors. • Controlled area/perimeter distance -- Site the ISFSI or MRS-controlled area to maintain distance to the perimeter of the site and locations of public occupancy. • Transfer route -- Locate transfer routes for ISFSI or MRS containers to maintain distance from the site perimeter. • Effluent discharges and impacts -- Incorporate consideration of natural and manmade contours, existing or planned rerouting of natural surface water, and points at which surface water exits the site relative to residences and public use areas; use cutoffs, drains, well points, or other means to control water flow.
Licensing, Design	11.4.2.1	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Installation Design Features (Part 2 of 2)	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	Features that minimize onsite exposures include: • Transfer route -- Locate transfer routes for ISFSI or MRS containers to or from the storage area and the handling areas (intermodal transfer points, or wet or dry transfer facility) to minimize the route between the handling and storage facilities, provide for minimal other traffic on the route, remain within the single controlled area, and maintain distance from the site perimeter. • Multiple restricted areas -- Incorporate use of multiple restricted areas within the controlled area to provide control of access to areas with radiation levels that would pose unacceptable risks or exposures to workers within those areas. • Controlled area/perimeter distance -- Provide separation of radioactive material handling and storage functions from other functions on the site; provide distance between radioactive material and both the boundary of the controlled area and onsite work stations outside the restricted area.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.2.2	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Access Control (Part 1 of 2)	20.1101, 20.1406, 20.1601, 20.1602 20.1701, 20.1702, 72.21(b) 72.24, 72.104, 72.106, 72.126	Control of access to controlled and restricted areas is performed for reasons related to radiation protection and for safeguards and security purposes. This section addresses control of access for purposes of limiting exposure to external radiation and radiological contamination hazards. The description of the ISFSI or MRS installation design should include (with consideration of the “information to be protected” provisions of 10 CFR 73.21(b)) the following access control elements: • Site layout to scale showing ISFSI or MRS controlled area (per 10 CFR 72.106) and any traversing right(s)-of-way. • Description of barrier used to preclude ready access to the controlled area. • Location and summary description of gate and/or overlook stations. The criteria used to designate restricted areas (or zones within restricted areas) should be identified. Descriptions should be provided for all protective features designed to limit access to restricted areas, including physical barriers, locked entryways, and audible or visible alarm signals. Continuous direct or electronic surveillance used to prevent unauthorized entry should also be described. Restricted areas may require further designation as high or very high radiation areas according to 10 CFR 20.1601 and 1602, respectively. Guidance on access control features applicable to these areas is provided in Regulatory Guide 8.39, “Control of Access to High and Very High Radiation Areas of Nuclear Plants.”

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.2.2	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Access Control (Part 2 of 2)	20.1101, 20.1406, 20.1701, 20.1702, 20.2003, 72.21(b) 72.24, 72.104, 72.106, 72.126	Restricted areas may be further divided to identify areas where the potential for contamination exists. Criteria used to designate contamination control areas (including airborne radioactivity areas) should also be identified. Access control features applicable to contamination control areas may include: • Incorporation of access control facilities into the building design or provisions for temporary or mobile-type facility(ies) immediately adjacent to the confinement barrier of the potentially contaminated area. • Male and female change rooms, including lavatories and showers; provisions for protective equipment and garments; stations for monitoring hands, feet, and whole body; and threshold stations for removal of booties on leaving the area. • Shower and lavatory water collection, storage, and provisions for routing of potentially contaminated water. Drawings should document that appropriate measures are provided for collection of possibly contaminated wash water and that leakage of possibly contaminated liquid onto or into the ground is precluded. Wash water may include liquids temporarily stored pending sampling and sample analysis before release to the sanitary sewer (in accordance with 10 CFR 20.2003) or collection for handling and treatment as radioactive waste.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.2.3	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Radiation Shielding	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	Provisions for effective shielding must be incorporated into the ISFSI or MRS design as an integral part of the ALARA and radiation protection programs to protect the public and workers against direct radiation. Guidance for conducting detailed engineering evaluations aimed at determining the performance and effectiveness of the proposed shield design is beyond the scope of this section but is provided in Chapter 7, Shielding Evaluation. However, criteria that can be specifically evaluated to determine whether the proposed shielding and installation designs satisfy dose rate and ALARA requirements are addressed in Section 11.4.3. The radiation protection review also uses the dose rates from the shielding review in combination with radionuclide emission rate estimates (from confinement and site-generated waste reviews) to ensure that the combined dose rates (i.e., from all sources and pathways) meet the acceptance criteria, as described in Section 11.4.3.
Licensing, Design	11.4.2.4	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Confinement and Ventilation	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	Confinement refers to the ability of the ISFSI or MRS to prevent the release of radioactive materials from areas where these materials are normally contained to areas where they are not normally contained, and ultimately, to the surrounding environment. Confinement barrier systems may be sealed, as in the case of most ISFSIs, or vented with off-gas treatment systems, as in the case with storage pools or waste management systems. For the latter, intake and exhaust filters and dampers, as well as portions of ducts and stacks of the ventilation systems function as elements of the confinement system. Confinement and ventilation function to provide protection of personnel against radiation exposures associated with releases of radioactive materials under normal conditions, anticipated occurrences, and accidents. Detailed evaluation of confinement is beyond the scope of this section, but is provided in Chapter (Section) 9, Confinement Evaluation, of this FSRP. The radiation protection review uses the emission rate estimates of the confinement analysis with shielding and site-generated waste review findings to determine whether the combined dose rates meet regulatory criteria, as described in FSRP Section 11.4.3.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.2.5	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Area Radiation and Airborne Radioactivity Monitoring Instrumentation (Part 1 of 2)	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	The locations, types, capabilities, and parameters of fixed area radiation monitors and continuous airborne monitoring instrumentation (as required by Regulatory Guide 3.48, Section 7.3.4) must be detailed in the drawings and specifications defining the ISFSI or MRS design. The NRC accepts for an ISFSI or MRS the criteria for fixed area radiation monitors and continuous airborne monitoring instrumentation as provided in: • ANSI N13.1-1993, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," as it relates to principles for obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling. • ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Reactors," as it relates to the criteria for locating fixed continuous area gamma radiation monitors and for design features and ranges of measurement. • NUREG-0800, "Standard Review Plan," Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems." • Regulatory Guides 8.5, "Criticality and Other Interior Evacuation Signals," as it relates to use of interior evacuation alarm signals, and 8.25, "Air Sampling in the Workplace," as it relates to use of fixed and portable air samplers in the workplace.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.2.5	**RADIATION PROTECTION EVALUATION** **Radiation Protection Design Features** Area Radiation and Airborne Radioactivity Monitoring Instrumentation (Part 2 of 2)	20.1101, 20.1406, 20.1701, 20.1702, 72.24, 72.104, 72.126	Classification of auxiliary power for monitoring instrumentation as “emergency” (important to safety) or “standby” (not important to safety) should correspond to the classification of the instrumentation itself. Some discriminators for classifying instrumentation and auxiliary power as important to safety are shown below: • If data provided by the monitoring system can have an immediate and determining effect on personal actions and operations to maintain compliance with the basic safety criteria, including prevention of unacceptable worker doses, then monitoring instrumentation should be classified as emergency. • If any of the following exist, then the instrumentation and its auxiliary power are probably not important to safety: — Instrumentation data are not provided real-time to a central control room or, if provided, do not trigger an alarm that results in actions that should preclude or mitigate unacceptable consequences. — Instrumentation does not trigger an alarm necessary to avoid unacceptable worker exposures at its location when a threshold is reached. — Data are collected only periodically. — No normal, off-normal, or accident-level events or conditions can result in changes in the monitored phenomena that can jeopardize satisfaction of the basic safety criteria.
Licensing, Design	11.4.3	**RADIATION PROTECTION EVALUATION** **Dose Assessment**	20.1101, 20.1201, 20.1301, 20.1302, 72.104, 72.106	From the evaluations described in the shielding evaluation (Chapter/Section 7), estimated dose rates should be provided for representative points within the restricted areas as well as on and beyond the perimeter of the controlled area. Additionally, the confinement evaluation (Chapter/Section 9) and site-generated waste management evaluation (Chapter/Section 14) should have produced estimates of radionuclide concentrations in effluents. The radiation protection review includes a dose assessment that incorporates findings of each of those reviews, as applicable. The major elements of the dose assessment and the applicable acceptance criteria are described below.

Category	NUREG 1567 Section	NUREG 1567 <u>SECTION TITLE</u> Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.3.1	**<u>RADIATION PROTECTION EVALUATION</u>** **Dose Assessment** Onsite Doses (Part 1 of 2)	20.1101, 20.1201, 20.1301, 20.1302, 72.104, 72.106	Individual and collective dose rates should be calculated for all onsite areas at which workers will be exposed to elevated dose rates (e.g., greater than 2 mrem/hr) or airborne radioactivity concentrations. Radiation doses should be based on direct exposure and radionuclide inhalation and should be computed for workers performing specific ISFSI or MRS functions, including routine, contingency, maintenance, or repair procedures, or other activities that can occur in elevated dose-rate areas. Individual and collective doses should also be determined for onsite functions outside the ISFSI or MRS restricted area associated with transportation and intermodal transfer of the radioactive materials to be stored. The SAR should include estimated occupancy time for personnel involved in these functions, including the maximum expected total hours per year for any individual and total person-hours per year for all personnel. The annual collective doses associated with each major function and each radiation area should be estimated, and the bases, models, and assumptions used in arriving at these values should be identified. If the material to be put into dry storage is in holding or storage in a pool, the doses associated with preparation for the loading operations should be based on assumptions of maximum radioactive material in the pool and in the storage location (unless the SAR analysis acceptably demonstrates that alternative conditions are justified).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.3.1	**RADIATION PROTECTION EVALUATION** **Dose Assessment** Onsite Doses (Part 2 of 2)	20.1101, 20.1201, 20.1301, 20.1302, 72.104, 72.106	All individual doses to workers should be well below the dose limits specified in 10 CFR 20.1201. Collective doses should be consistent with the objectives contained in the applicant's ALARA program. The information provided by the applicant must allow for the determination of compliance with these criteria. In general, the following information will allow for such a determination: • Collective and individual doses associated with all operations involved with placing one full storage confinement cask in storage position are identified and listed according to associated function. • Annual collective and individual doses are estimated by multiplying the single-cask dose by the maximum annual rate for placing casks into storage. This estimation assumes that the same personnel will be involved in the same operations for each cask. If the doses exceed those allowed by 10 CFR 20.1201(a), the planned conduct of operations (Chapter/Section 10) should commit to conditions (staffing plan, monitoring, etc.) that would ensure that 10 CFR 20.1201(a) dose limits are not exceeded. • Estimates of annual doses for operation of the ISFSI for material in storage and material in wet holding or wet storage should be provided for comparison with maximum allowable doses given in 10 CFR 20.1201. • The SAR should include discussion of sensitivity of the doses to assumptions and uncertainties, including the use of conservative assumptions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.3.2	**RADIATION PROTECTION EVALUATION** **Dose Assessment** Offsite Doses (Part 1 of 2)	20.1101, 20.1201, 20.1301, 20.1302, 72.104, 72.106	The dose rate beyond the controlled area boundary should not exceed 2 mrem in any one hour from all licensed activities at the site (10 CFR 20.1301). The maximum projected annual dose at or beyond the controlled area boundary should not exceed 25 mrem to the whole body, 75 mrem to the thyroid, or 25 mrem to any other organ (10 CFR 72.104). Demonstration of compliance with 10 CFR 72.104 dose limits is an indication that the dose limits of U.S. Environmental Protection Agency regulations 40 CFR 190 and 191 also will be met. The collective dose should be determined as the sum of the products of individual doses in each of 16 compass sectors around the installation and the number of population members in each sector. Sectors should be centered between the arcs having radii of 1.5, 3, 5, 6.5, and 8 km (about 1, 2, 3, 4, and 5 miles). The dose should be based on all important exposure pathways (direct radiation, airborne releases, etc.) and modes of exposure (external exposure, inhalation, etc.) and should be specified as whole-body or effective. In addition, the organ receiving the highest dose should be identified. Dose calculations must consider direct radiation and discharges of radioactive material under both normal conditions and anticipated occurrences as well as contributions from other uranium fuel-cycle facilities within the region. The methodology applied must be acceptable to the NRC.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	11.4.3.2	**RADIATION PROTECTION EVALUATION** **Dose Assessment** Offsite Doses (Part 2 of 2)	20.1101, 20.1201, 20.1301, 20.1302, 72.104, 72.106	The following considerations also apply: • The direct radiation dose rate should be calculated on the basis of the maximum quantity of radioactive material permitted by the ISFSI or MRS license. • The dose assessment should assume that the radioactive material is distributed in such a manner as to produce the highest perimeter dose rate, unless such arrangements are specifically precluded by proposed operational considerations. • The effective dose to any member of the public resulting from airborne emissions of radioactive material should conform to the ALARA constraint level of 10 mrem/yr specified by 10 CFR 20.1101(d). • The applicant's environmental monitoring program should be designed to provide exposure and concentration data for those pathways that lead to the highest potential external and internal doses.
Licensing, Operation	11.4.4	**RADIATION PROTECTION EVALUATION** **Health Physics Program**	20.1101, 20.1302, 20.1406, 20.1051, 20.1702, 72.24, 72.126	No text provided.
Licensing, Operation	11.4.4.1	**RADIATION PROTECTION EVALUATION** **Health Physics Program** Organization	20.1101, 20.1302, 20.1406, 20.1051, 20.1702, 72.24, 72.126	Guidance on the organization and planning for health physics (radiation protection) activities at an ISFSI or MRS is contained in Regulatory Guides 8.8, 8.10, and others. The ISFSI or MRS management organization should identify an individual with clearly designated responsibility for health physics. To avoid the potential for conflict of interest, this individual's reporting line should not go through the manager responsible for operations. The position should be maintained for the life of the facility, including all decontamination and decommissioning (D&D) operations.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.4.2	**RADIATION PROTECTION EVALUATION** **Health Physics Program** Equipment, Instrumentation, and Facilities (Part 1 of 2)	20.1101, 20.1302, 20.1406, 20.1051, 20.1702, 72.24, 72.126	Health physics program equipment, instrumentation, and facilities should be described in the SAR. The need for specific health physics components depends on the nature of the installation, for example, whether it includes a pool facility or whether some laboratory functions are performed at offsite facilities. In any case, portable and laboratory equipment and instrumentation should include: • Personal monitoring devices for external dosimetry, including provisions for dosimeter processing by a National Voluntary Laboratory Accreditation Program-accredited dosimetry service. • Handheld/portable radiation meters and detectors for performing radiation and contamination surveys; an appropriate number of instruments should be available for each type of survey to be performed (e.g., Geiger-Mueller (G-M) survey instruments for contamination surveys and personnel “frisking,” ionization chambers for exposure rate surveys, neutron detectors for neutron flux or dose rate surveys, etc.). • Portable air sampling equipment. • Facilities for internal radiation monitoring, including whole-body counters, thyroid counters, bioassay sample analysis equipment, etc. • Personnel protective equipment (including respirators certified by National Institute for Occupational Safety and Health/Mine Safety and Health Administration). • Decontamination equipment and facilities, including spill control materials, shower, eyewash, changing facilities, etc.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.4.2	**RADIATION PROTECTION EVALUATION** **Health Physics Program** Equipment, Instrumentation, and Facilities (Part 2 of 2)	20.1101, 20.1302, 20.1406, 20.1051, 20.1702, 72.24, 72.126	Health physics facilities can be in permanent structures, temporary buildings, or trailers. Facilities should be located outside restricted areas and, if practicable, away from areas with elevated dose rates. Exceptions can include facilities for storing items that need to be readily available within restricted or elevated dose rate areas, as well as personnel decontamination, shower, and changing facilities. Health physics facilities should be identified on the plot drawing of the installation and should be described to an extent that acceptably demonstrates the applicant's understanding of the associated requirements and functions. The following regulatory guides and industry standards provide information, recommendations, and guidance on various aspects of health physics equipment, instrumentation, and facilities. The NRC considers these sources as acceptable in describing a basis for implementing activities to comply with applicable regulatory requirements: • ANSI/ANS N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities." • Regulatory Guides 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters"; 8.6, "Standard Test Procedures for Geiger-Mueller Counters"; 8.14, "Personnel Neutron Dosimeters"; 8.25, "Air Sampling in the Workplace"; and 8.28, "Audible Alarm Dosimeters." • NUREG-0800, Section 12.5, "Operations Radiation Protection Program."

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.4.3	**RADIATION PROTECTION EVALUATION** **Health Physics Program** Policies and Procedures (Part 1 of 2)	20.1101, 20.1302, 20.1406, 20.1051, 20.1702, 72.24, 72.126	10 CFR Part 20.1101 requires that licensees “develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities.” The application should describe the radiation protection program, including details on all health physics-related policies and procedures, to be implemented at the ISFSI or MRS. The applicant should commit to reviewing the program for content and implementation at least annually. A listing of major program elements, along with the parameters that should be described under each element, is given in Table 11.2. Applicable regulatory criteria and guidance documents are also listed.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	11.4.4.3	**RADIATION PROTECTION EVALUATION** **Health Physics Program** Policies and Procedures (Part 2 of 2)	20.1101, 20.1302, 20.1406, 20.1051, 20.1702, 72.24, 72.126	In addition to the regulatory guides identified in Table 11.2, applicable guidance and criteria for health physics procedures relevant to ISFSI or MRS operations are contained in the following: • ANSI/ANS N13.2, "Guide to Administrative Practices in Radiation Monitoring." • ANSI/ANS N13.6, "Practice for Occupational Radiation Exposure Record Systems." • ASTM E 1167, "Guide for Radiation Protection Program for Decommissioning Operations." • ASTM E 1168, "Guide for Radiation Protection Training for Nuclear Facility Workers." • HPS N 13.30-1996, "Performance Criteria for Radiobioassay" (An American National Standard). • HPS N 13.32-1995, "Performance Testing of Extremity Dosimeters" (An American National Standard). • HPS N 13.41-1997, "Criteria for Performing Multiple Dosimetry" (An American National Standard). • HPS N 13.42-1997, "Internal Dosimetry Program for Mixed Fission and Activation Products" (An American National Standard). • National Council on Radiation Protection (NCRP) 59, "Operational Radiation Safety Program." • NCRP 71, "Operational Radiation Safety Training." • National Safety Council (NSC), "Accident Prevention Manual for Industrial Operations." • NUREG-0800, Section 12.5.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4	**QUALITY ASSURANCE EVALUATION** **Acceptance Criteria** (Part 1 of 2)	72.24, 72.122, 72.140, 72.142, 72.144, 72.146, 72.148, 72.150, 72.152, 72.154, 72.156, 72.158, 72.160, 72.162, 72.164, 72.166, 72.168, 72.170, 72.172, 72.174, 72.176	The applicant's QA program must describe the program that is established, will be maintained, and will be executed for the design, fabrication, construction, testing, operation, modification, and decommissioning of the SSCs of the ISFSI or MRS that are important to safety. The QA program description must identify the items important to safety and include information about managerial and administrative controls to ensure safe operation of the ISFSI or MRS. The applicant's QA program must be structured to apply QA measures and controls to all activities and items in proportion to their importance to safety (graded approach). A graded application of QA requires applicant justification and reviewer acceptance. The graded approach for the application of QA must be described adequately. The QA program must identify the activities and items that are important to safety and the degree of their importance. The highly important-to-safety activities and items must have a high-level of control, while those less important may have a lower level of control. An applicant may choose to apply the highest level of QA and control to all activities and items. From that point on, the assignment of QA levels of control to be used within the QA program must be based on a graded application to the 18 criteria listed.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4	**QUALITY ASSURANCE EVALUATION** **Acceptance Criteria** (Part 2 of 2)	72.24, 72.122, 72.140, 72.142, 72.144, 72.146, 72.148, 72.150, 72.152, 72.154, 72.156, 72.158, 72.160, 72.162, 72.164, 72.166, 72.168, 72.170, 72.172, 72.174, 72.176	For each of the activities and items identified as important to safety, the applicant must identify and define the level of application for each of the following QA programmatic elements. The attributes listed for each topic must be applied collectively only in the most stringent application of the QA program. Less stringent application of requirements may be effected by modifying or eliminating some attributes from selected topics. The applicant's QA program and associated QA program controls and implementing procedures regarding activities performed must be in place before activities begin. Defining a process involves establishing authorities and assigning responsibilities. Implementing a process involves issuing instructions and procedures. The acceptance criteria are organized to reflect the 18 sections of 10 CFR 72, Subpart G, that correspond to the 18 basic requirements of NQA-1, 1983 Edition. Each acceptance criteria section shows the title of the relevant area of review in Section 12.2. Notes following each acceptance criterion identify the source(s) of substantive content of the criterion.
Licensing, Quality Assurance	12.4.1	**QUALITY ASSURANCE EVALUATION** **QA Organization**	72.24, 72.122, 72.140, 72.142	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.1.1	**QUALITY ASSURANCE EVALUATION** **QA Organization** Assignment of Functional Responsibilities	72.24, 72.122, 72.140, 72.142	The applicant must present organization charts describing functional responsibilities that: • Establish clear lines of authority and responsibility for activities important to safety to be performed by applicant organizational elements and by contractors, whether onsite or offsite (10 CFR 72.142; NQA-1/Part I/Sec II Basic 1, 1S-1 Par 2, 3.1, 3.2; NUREG-0800; NUREG-1536). • Cover activities to achieve quality objectives, i.e., performance in accordance with specified requirements of activities such as site design, purchasing, fabricating, constructing, handling, shipping, receiving, storing, cleaning, erecting, assembling, installing, inspecting, testing, operating, maintaining, repairing, modifying, and decommissioning (10 CFR 72.24(n), 72.140(a); NQA-1, Part I, Sec II Basic 1, 1S-1 Par 2.1(a)). • Cover QA functions, that is, ensuring that an appropriate QA program is established and effectively executed, and verifying, by procedures such as checking, auditing, and inspection, that work activities have been correctly performed and physical characteristics and quality of material and components adhere to predetermined requirements (10 CFR 72.140(a),(b), 72.142(a),(b); NQA-1/Part I/Sec II, 1S-1 Par 2.1(b)).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.1.2	**QUALITY ASSURANCE EVALUATION** **QA Organization** Responsibility for QA	72.24, 72.122, 72.140, 72.142	The applicant must: • Assign to elements of its own organization the responsibility for establishment and execution of its QA program (10 CFR 72.142, first paragraph; NQA-1/Part I/Sec II, 1S-1 Par 2.2; NUREG-0800; NUREG-1536) and oversight and evaluation of work to establish and execute its QA program that is delegated to others (10 CFR 72.142, first paragraph; NQA-1/Part I/Sec II, 1S-1 Par 2.2; NUREG-0800; NUREG-1536) • Assign to a high-level of applicant line management the responsibility for promulgating corporate or company QA policies, goals, and objectives; maintaining a continuing involvement in QA matters; and approving procedures for resolution of disputes between applicant organizational elements about QA activities (10 CFR 72.142, first paragraph; NQA-1/Part I/Sec II, 1S-1 Par 2.2; NUREG-0800; NUREG-1536) • Establish lines of communication among that line position, intermediate levels of line management, contractors involved in activities impacted by the QA program, and QA management positions (10 CFR 72.142, first paragraph; NQA-1/Part I/Sec II, 1S-1 Par 2.2; NUREG-0800; NUREG-1536)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.1.3	**QUALITY ASSURANCE EVALUATION** **QA Organization** Organization for Performing QA Function	72.24, 72.122, 72.140, 72.142	The applicant's organization for performing QA functions must: • Include a management position having authority and responsibility for developing the QA program (e.g., QA Manager, Director, or Vice President). This management position must (a) have appropriate qualification requirements (e.g., education, experience, technical competence) (10 CFR 72.142, last paragraph; NUREG-0800; NUREG-1536), (b) be at an organizational level at least as high as that of the highest level line manager having direct responsibility for cost and scheduling of work activities to attain quality objectives (10 CFR 72.142, last paragraph; NQA-1/Part I/Sec II Basic 1; NUREG-0800; NUREG-1536), (c) be independent of responsibility for cost and scheduling of work activities to attain quality objectives and have no other duties that would prevent full attention to QA matters (10 CFR 72.142, last paragraph; NQA-1/Part I/Sec II Basic 1; NUREG-0800; NUREG-1536), and (d) have an assigned staff (without responsibilities for work activities to impart quality) to develop and implement procedures for verifying conformance of work activities and results of work activities to specified requirements, or authority and a budget to contract for such services (10 CFR 72.142, last paragraph; NUREG-0800). • Include individuals assigned responsibility for ensuring effective execution of any portion of the QA program who have sufficient authority, access to work areas, and organizational freedom to (a) identify quality problems; (b) initiate, recommend, or provide solutions to quality problems through designated channels; (c) verify implementation of solutions; and (d) stop unsatisfactory work and ensure that furtherprocessing, delivery, installation, or use is controlled until proper disposition of a nonconformance, deficiency, or unsatisfactory condition (10 CFR 72.142, last paragraph; NQA-1/Part I/Sec II Basic 1; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.2	**QUALITY ASSURANCE EVALUATION** **QA Program**	72.24, 72.122, 72.140, 72.144	No text provided.

A-2. DETAILED REQUIREMENTS OF NUREG-1567

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.2.1	**QUALITY ASSURANCE EVALUATION** **QA Program** Scope of Program	72.24(n), 72.122, 72.140(a) and (b), 72.144(a)	The applicant must establish a QA program that: • Identifies SSCs and activities that are important to safety, major organizations participating in the program, and functions of those organizations (10 CFR 72.24(n), 72.140(a),(b), 72.144(a); NQA-1/Part I/Sec I Par 2, 3). • Applies to (a) siting, designing, purchasing, fabricating, constructing, handling, shipping, receiving, storing, cleaning, erecting, assembling, installing, inspecting, testing, operating, maintaining, repairing, modifying, and decommissioning of the identified SSCs; and (b) managerial and administrative controls to ensure safe operation of the facility, both before issuance of a license and throughout the life of licensed activity (10 CFR 72.24(n), 72.140(a),(b), 72.144(a); NQA-1/Part I/Sec I/Par 2; NUREG-0800; NUREG-1536). • Covers activities that provide confidence that an SSC will perform satisfactorily in service (QA), including activities that determine that physical characteristics and quality of materials or components adhere to redetermined requirements (quality control) (10 CFR 72.140(a); NQA-1/Part I/Sec II Basic 2).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.2.2	**QUALITY ASSURANCE EVALUATION** **QA Program** Implementation of Program (Part 1 of 2)	72.24(n), 72.122, 72.140(a) and (b), 72.142(a,) 72.144(a) and (b),	The applicant must have written statements of QA policies, goals, and objectives, and procedures and instructions that: • Are under controlled distribution (both original versions and subsequent changes) (10 CFR 72.144(a); NUREG-1536). • Inform responsible organizations and individuals of the applicant's QA policies, goals, and objectives, and make compliance with QA policies and procedures mandatory (10 CFR 72.142(a), 72.144(a); NUREG-0800; NUREG-1536). • Identify criteria of 10 CFR 72, Subpart G, and any other specific provisions that apply to applicant activities in progress (10 CFR 72.24(n), 72.140(a); 72.144(a); NQA-1/Part I/Basic 1). • Relate QA procedures to the criteria of 10 CFR 72, Subpart G, and other specific provisions that apply to applicant activities (10 CFR 72.142(a), 72.144(a); NUREG-1536). • Prescribe actions by persons and organizations engaged in activities affecting safety that will result in compliance with requirements of the applicable criteria of 10 CFR 72 to an extent that is commensurate with their importance to safety (NQA-1, Sec I Par 3; NQA-1/Part I/Sec II Basic 2) and ensure conformance to the approved design of each ISFSI or MRS (10 CFR 72.144(b)).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.2.2	**QUALITY ASSURANCE EVALUATION** **QA Program** Implementation of Program (Part 2 of 2)	72.24(n), 72.122, 72.140(a) and (b), 72.144(c)	• Prescribe suitably controlled conditions for performing activities affecting quality, including (a) satisfying prerequisites for the given activity, establishing suitable environmental conditions (e.g., adequate cleanliness), and using appropriate equipment; (b) providing special controls, processes, test equipment, tools, and skills needed to attain the required quality; and (c) verifying quality by inspection and test (NQA-1/Part I/Sec II Basic 2). • Provide a level of detail in requirements and procedures for performing work that is appropriate to the complexity and proposed use of the structures, systems, or components affected, including (a) the impact of malfunction or failure of the item on safety; (b) the design and fabrication complexity or uniqueness of the item; (c) the need for special controls and surveillance over processes and equipment; (d) the degree to which functional compliance can be demonstrated by inspection or test; and (e) the quality history and degree of standardization of the item (10 CFR 72.144(c); NQA-1/Part I/Sec II Basic 2). • Provide for indoctrination and training of personnel performing activities affecting quality including (a) instructing personnel responsible for performing activities affecting quality in the purpose, scope, and implementation of quality-related manuals, instructions, and procedures; (b) training and qualifying personnel performing activities affecting quality in the principles and techniques of the activity being performed; (c) certifying qualified personnel in accordance with relevant codes and standards; (d) maintaining proficiency of personnel performing activities affecting quality by retraining, reexamining, and recertifying; and (e) maintaining records of completed training and qualification (NQA-1/Part I/Sec II Basic 2; NUREG-1536)

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.2.3	**QUALITY ASSURANCE EVALUATION** **QA Program** Review of Program	72.24(n), 72.122, 72.140(a) and (b), 72.144(d)	The applicant must have assigned responsibilities and have provided, prior to implementation, written instructions and procedures for: • Review, by the applicant, of the status and adequacy of the QA program through frequent contact with the program through reports, meetings, and audits; and periodic formal assessments in which performance is documented and necessary corrective and follow up actions are identified (10 CFR 72.144(d); NQA-1/Part I/Sec II Basic 2; NUREG-0800; NUREG-1536). • Review by management of other organizations participating in the QA program of that part of the program they are executing (10 CFR 72.144(d); NQA-1/Part I/Sec II Basic 2; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance, Design	12.4.3	**QUALITY ASSURANCE EVALUATION** **Design Control**	72.24, 72.122, 72.140, 72.146	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for the design process, design interfaces, design verification, and design changes.
Licensing, Quality Assurance, Design	12.4.3.1	**QUALITY ASSURANCE EVALUATION** **Design Control** Design Process	72.24, 72.122, 72.140, 72.146(a)	Instructions and procedures must provide for: • Translation of applicable regulatory requirements and the design basis into specifications, drawings, procedures, and instructions (10 CFR 72.146(a); NQA-1/Part I/Sec II Basic 3, 3S-1 Par 2, 3; NUREG-1536). • Actions to ensure that appropriate quality standards are specified and included in design documents (10 CFR 72.146(a); NQA-1/Part I/Sec II 3S-1 Par 3; NUREG-1536) and deviations from standards are controlled (10 CFR 72.146(a); NQA-1/Part I/Sec II 3S-1 Par 2; NUREG-1536). • Actions to ensure selection and review for suitability of application of materials, parts, equipment (including commercial-grade items and computer systems), and processes that are essential to the functions of the SSCs important to safety (NQA-1/Part I/Sec II 3S-1 Par 3; NUREG-1536). • Application of the above actions to criticality physics, radiation, shielding, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for in-service inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections and tests (10 CFR 72.146(a)).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance, Design	12.4.3.2	**QUALITY ASSURANCE EVALUATION** **Design Control** Design Interfaces	72.24, 72.122, 72.140, 72.146(b)	Instructions and procedures must provide for: • Actions to identify and control design interfaces and to provide for coordination among participating design organizations (10 CFR 72.146(b); NQA-1/Part I/Sec II 3S-1 Par 6; NUREG-0800; NUREG-1536). • Actions to establish written procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces (10 CFR 72.146(b); NQA-1/Part I/Sec II 3S-1 Par 6).
Licensing, Quality Assurance, Design	12.4.3.3	**QUALITY ASSURANCE EVALUATION** **Design Control** Design Verification	72.24, 72.122, 72.140, 72.146(b)	Instructions and procedures must provide for: • Actions to verify and check the adequacy of design, by methods such as design reviews or alternate or simplified calculational methods, or by a suitable testing program (10 CFR 72.146(b); NQA-1/Part I/Sec II 3S-1 Par 4). • Verifying or checking of designs by individuals or groups other than those who were responsible for the original design (and normally other than the designer’s immediate supervisor) and who have a level of skill at least equal to that of the original designer to (a) confirm that the design of a structure, system, or component is suitable for its intended purpose by critical reviews of design inputs, assumptions, design methods, incorporation of inputs into the design, design outputs, and design interfaces; and (b) verify correctness of design calculations or analyses by calculations or analyses using alternate methods (10 CFR 72.146(b); NQA-1/Part I/Sec II 3S-1 Par 4; NUREG-0800; NUREG-1536). • Ensuring that a test program used to verify the adequacy of a specific design feature, in lieu of other verifying or checking processes, includes suitable qualification testing of a prototype or example unit under the most adverse design conditions (10 CFR 72.146(b); NQA-1/Part I/Sec II 3S-1 Par 4.2.3; NUREG-0800; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance, Design	12.4.3.4	**QUALITY ASSURANCE EVALUATION** **Design Control** Design Change	72.24, 72.122, 72.140, 72.146(c)	Instructions and procedures must provide for: • Subjecting design changes, including field changes, to design control measures commensurate with those applied to the original design (10 CFR 72.146(c); NQA-1/Part I/Sec II 3S-1 Par 5; NUREG-0800; NUREG-1536). • Obtaining NRC approval of changes in the conditions specified in the license (10 CFR 72.146(c)).
Licensing, Quality Assurance	12.4.4	**QUALITY ASSURANCE EVALUATION** **Procurement Document Control**	72.24, 72.122, 72.140, 72.148	No text provided.
Licensing, Quality Assurance	12.4.4.1	**QUALITY ASSURANCE EVALUATION** **Procurement Document Control** Control Process (Part 1 of 2)	72.24, 72.122, 72.140, 72.148	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for ensuring that documents for procurement of material, equipment, or services issued by the applicant or its contractors or subcontractors, and changes thereto: • Include a statement of work to be performed by the supplier (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.1). • Include or reference applicable regulatory requirements, design bases, and other requirements that are necessary to ensure adequate quality of purchased items or services. (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.2; NUREG-0800; NUREG-1536). • Specify technical requirements (may be done by reference to specific drawings, specifications, codes, standards, regulations, procedures, or instructions) (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1, Par 2.2; NUREG-0800; NUREG-1536). • Identify test, inspection, and acceptance requirements of the purchaser (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.2; NUREG-0800; NUREG-1536). • Provide for access to the supplier's facilities and records for inspection or audit by the purchaser or purchaser's representative (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.4; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.4.1	**QUALITY ASSURANCE EVALUATION** **Procurement Document Control** Control Process (Part 2 of 2)	72.24, 72.122, 72.140, 72.148	• Identify documentation to be submitted for information, review, or approval by the purchaser (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.5; NUREG-1536). • Prescribe retention times and disposition requirements for QA records that are to bemaintained by the supplier (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.5; NUREG-1536). • Include purchaser's requirements for reporting and approving disposition of nonconformances (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.6). • Require identification of appropriate spare and replacement parts or assemblies and delineation of the technical and QA-related data required for ordering these parts or assemblies (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.7). • Are reviewed prior to transmission to the supplier to ensure that they include appropriate provisions for assuring that items or services will meet specified requirements (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 3; NUREG-1536).
Licensing, Quality Assurance	12.4.4.2	**QUALITY ASSURANCE EVALUATION** **Procurement Document Control** Contractor QA Programs	72.24, 72.122, 72.140, 72.148	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for requiring, to the extent necessary, that contractors or subcontractors adhere to a QA program consistent with the applicable provisions of 10 CFR 72, Subpart G (10 CFR 72.148; NQA-1/Part I/Sec II 4S-1 Par 2.3; NUREG-0800).
Licensing, Quality Assurance	12.4.5	**QUALITY ASSURANCE EVALUATION** **Instructions, Procedures, and Drawings**	72.24, 72.122, 72.140, 72.150	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.5.1	**QUALITY ASSURANCE EVALUATION** **Instructions, Procedures, and Drawings** Development Process	72.24, 72.122, 72.140, 72.150	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for obtaining or producing instructions, procedures, and drawings that: • Prescribe how activities affecting quality are to be performed • Are appropriate to the circumstances under which the activities are performed. • Include acceptance criteria for determining that important activities have been accomplished satisfactorily, (e.g., quantitative criteria such as dimensions, tolerances, and operating limits, and/or qualitative criteria such as legible, “well-defined borders,” or “smooth to the touch”). • Are reviewed and concurred to by the applicant’s QA organization if the instructions or procedures prescribe performance of inspections, tests, calibration, or special processes (10 CFR 72.150; NQA-1/Part I/Sec II Basic 5; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.5.2	**QUALITY ASSURANCE EVALUATION** **Instructions, Procedures, and Drawings** Utilization	72.24, 72.122, 72.140, 72.150	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Distributing controlled copies of instructions, procedures, or drawings important to safety to those performing the activity (see Section 12.4.6 control process) • Enforcing adherence to the provisions of the instructions, procedures, and drawings (10 CFR 72.150; NQA-1/Part I/Sec II Basic 5).
Licensing, Quality Assurance	12.4.6	**QUALITY ASSURANCE EVALUATION** **Document Control**	72.24, 72.122, 72.140, 72.152	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.6.1	**QUALITY ASSURANCE EVALUATION** **Document Control** Scope of Control Process	72.24, 72.122, 72.140, 72.152	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for preparing, issuing, and changing documents that: • Specify quality requirements or • Prescribe activities affecting quality, including, at a minimum, design specifications; design and fabrication drawings; procurement documents; QA procedures; design criteria documents; instructions and procedures for fabrication, inspection, and tests; as-built documentation; QA procedures; and nonconformance reports (10 CFR 72.152; NQA-1/Part I/Sec II Basic 6; NUREG-1536).
Licensing, Quality Assurance	12.4.6.2	**QUALITY ASSURANCE EVALUATION** **Document Control** Control Process	72.24, 72.122, 72.140, 72.152	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying documents that prescribe activities affecting quality or specify quality requirements. • Preparing, reviewing, and approving such documents. • Preparing, reviewing, and approving changes to such documents, with review and approval by the same organizations that performed the original review and approval unless other organizations are specifically designated to do so. • Distributing such documents and changes to locations where they are to be used (10 CFR 72.152; NQA-1/Part I/Sec II 6S-1 Par 2, 3; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.7	**QUALITY ASSURANCE EVALUATION** **Control of Purchased Material, Equipment, and Services**	72.24, 72.122, 72.140, 72.154	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.7.1	QUALITY ASSURANCE EVALUATION **Control of Purchased Material, Equipment, and Services** Control Process (Part 1 of 3)	72.24, 72.122, 72.140, 72.154(a)	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for ensuring that purchased material, equipment, and services conform to relevant procurement documents by: • Evaluating potential suppliers' capability to provide acceptable products and services, prior to award of a procurement order or contract, by assigning qualified technical and QA personnel to make evaluations. The extent of investigation should be commensurate with importance to the safety of the item or service being procured and should document results of supplier evaluations for use in current and future procurement actions (10 CFR 72.154(a); NQA-1/Part I/Sec II Basic 7, 7S-1 Par 3; NUREG-1536). The investigation must be performed by either: (a) reviewing records of past performance of suppliers in providing items or services of the type being procured or items with similar technical and quality requirements or (b) surveying supplier facilities and QA programs to determine current capability to supply items or services meeting applicant's technical and quality requirements.

A-2. DETAILED REQUIREMENTS OF NUREG-1567

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.7.1	**QUALITY ASSURANCE EVALUATION** **Control of Purchased Material, Equipment, and Services** Control Process (Part 2 of 3)	72.24, 72.122, 72.140, 72.154(a)	• Conducting surveillance of supplier activities during fabrication, inspection, testing, and shipment of materials, components, assemblies, parts, and other products that are important to safety by: (a) identifying technical and quality requirements conformance of which to purchase orders or contracts cannot be confirmed satisfactorily by inspections or tests upon receipt; (b) identifying supplier processes or activities in which conformance to those requirements can be confirmed by inspecting, witnessing, or verifying results; (c) preparing plans specifying methods to be used in inspecting, witnessing, or verifying results of each supplier process or activity selected for surveillance; acceptance criteria for the process or activity; schedules for surveillance activities; and extent of documentation of the surveillance; (d) informing suppliers of the purpose and scheduling of surveillance activities and requirements for notification to the applicant when processes or activities are nearing hold or witness points (by reference to provisions of procurement documents or changes to them if original documents are not adequate); (e) assigning qualified technical and QA personnel to perform surveillance activities in accordance with the prepared plans and schedules; and (f) performing follow up surveillance activities when needed to verify correction of deficiencies found in scheduled surveillance activities (10 CFR 72.154(a); NQA-1/Part I/Sec II Basic 7, 7S-1, Par 8; NUREG-0800; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.7.1	**QUALITY ASSURANCE EVALUATION** **Control of Purchased Material, Equipment, and Services** Control Process (Part 3 of 3)	72.24, 72.122, 72.140, 72.154(a) and (b)	• Preparing or obtaining documentary evidence of the quality of products and services that are important to safety, including: (a) reports of results of surveillance activities; (b) supplier certificates of conformance and other documents identifying specific procurement requirements that have been met for products or services delivered; (c) supplier certificates of conformance and other documents identifying specific procurement requirements (e.g., codes, standards, specifications) that have been met for products or services delivered; and (d) applicant or supplier documents identifying specific procurement requirements that have not been met and describing the disposition of nonconforming items (e.g., accept as is, repair, rework, scrap, return to vendor) (10 CFR 72.154(b); NQA-1/Part I/Sec II Basic 7, 7S-1 Par 6, 8; NUREG-1536). • Retaining or having available the documentary evidence of quality of products and services that are important to safety for the life of ISFSI or MRS (10 CFR 72.154(b)). • Performing receiving inspections upon delivery of purchased items and materials to ensure that: (a) items or materials received are correctly identified and are the items or materials specified by relevant purchasing documents; (b) items or materials received conform to predetermined acceptance requirements (e.g., dimensions, weight, color, condition, accompanying documents) before they are released for use; and (c) items or materials that do not conform to predetermined acceptance requirements or require further inspections or tests before acceptance are identified and placed under appropriate controls (see Sections 12.4.14 and 12.4.15) (10 CFR 72.154(a); NQA-1/Part I/Sec II Basic 7, 7S-1 Par 8; NUREG-0800; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.7.2	**QUALITY ASSURANCE EVALUATION** **Control of Purchased Material, Equipment, and Services** Acceptance of Services Only	72.24, 72.122, 72.140, 72.154(a)	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for accepting services such as third-party inspection; engineering and consulting services; and installation, repair, overhaul, or maintenance work by: • Technical verification of data produced. • Surveillance or audit of the activity. • Review of certifications and reports submitted for evidence of conformance to procurement document requirements (10 CFR 72.154(a); NQA-1/Part I/Sec II 7S-1 Par 8.3).
Licensing, Quality Assurance	12.4.7.3	**QUALITY ASSURANCE EVALUATION** **Control of Purchased Material, Equipment, and Services** Commercial-Grade Items	72.24, 72.122, 72.140, 72.154(a) and (b)	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying commercial-grade items in approved design output documents. • Conducting supplier evaluation when warrented by complexity and importance to safety. • Identifying commercial-grade items in purchase documents by the manufacturer's published product description (e.g., catalog number). • Inspecting and testing items received to ensure conformance to the manufacturer's published requirements. • Documenting receipt and acceptability of the item (10 CFR 72.154(a),(b); NQA-1/Part I/Sec II 7S-1, Par 10).
Licensing, Quality Assurance	12.4.7.4	**QUALITY ASSURANCE EVALUATION** **Control of Purchased Material, Equipment, and Services** Assessments of Effectiveness	72.24, 72.122, 72.140, 72.154(c)	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for assessing the effectiveness of quality control by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services (10 CFR 72.154(c); NQA-1/Part I/Sec II 7S-1, Par 5; NUREG-0800; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.8	**QUALITY ASSURANCE EVALUATION** **Identification and Control of Materials, Parts, and Components**	72.24, 72.122, 72.140, 72.156	No text provided.
Licensing, Quality Assurance	12.4.8.1	**QUALITY ASSURANCE EVALUATION** **Identification and Control of Materials, Parts, and Components** Identification and Control Process	72.24, 72.122, 72.140, 72.156	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for ensuring that only correct and accepted materials, parts, and components are used or installed, by: • Identifying purchased materials, parts, or components upon receipt by tagging, marking, or labeling; physical separation; documents traceable to the items; or other means that provide adequate identification (10 CFR 72.156; NQA-1/Part I/Sec II Basic 8, 8S-1 Par 2; NUREG-1536). • Maintaining identification of purchased items during storage, subdivision for issue, and use or installation by means mentioned in the previous bullet or by procedural controls (10 CFR 72.156; NQA-1/Part I/Sec II Basic 8, 8S-1 Par 2; NUREG-1536). • Identifying materials produced, items fabricated onsite at the time that they are produced or fabricated, and items assembled onsite at the time assembly is begun, by means similar to that described above for purchased items in the first bullet above; and maintaining identification of such materials and items through use or installation by means similar to that described for purchased items (10 CFR 72.156; NQA-1/Part I/Sec II Basic 8, 8S-1 Par 2; NUREG-1536). • Providing for traceability of items (when required by codes, standards, or specifications) to: (a) applicable specification and grade of material; (b) heat, batch, lot, part, or serial number; and (c) specified inspection, test, or other records such as drawings, purchase orders, deviation reports, or reports of nonconformances and their disposition (10 CFR 72.156; NQA-1/Part I/Sec II Basic 8, 8S-1 Par 3.1; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.8.2	**QUALITY ASSURANCE EVALUATION** **Identification and Control of Materials, Parts, and Components** Stored Items	72.24, 72.122, 72.140, 72.156	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for maintaining identification of items in prolonged storage or storage under adverse conditions by: • Protecting markings and identification records of items in storage from deterioration from environmental exposure or adverse storage conditions. • Restoring or replacing markings or identification records that are damaged by aging or storage conditions (10 CFR 72.156; NQA-1/Part I/Sec II Basic 8, 8S-1 Par 3.3).
Licensing, Quality Assurance	12.4.8.3	**QUALITY ASSURANCE EVALUATION** **Identification and Control of Materials, Parts, and Components** Items With Limited Lifetimes	72.24, 72.122, 72.140, 72.156	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying items with limited calendar or operating life or cycles (e.g., certain batteries, chemical products, mechanical relays, and control switches). • Establishing and maintaining records of shelf or operating life or cycles remaining. • Preventing issue of items whose shelf life has expired. • Preventing further use of items that have reached the end of their operating life or cycles (10 CFR 72.156; NQA-1/Part I/Sec II Basic 8, 8S-1 Par 3.2).
Licensing, Quality Assurance	12.4.9	**QUALITY ASSURANCE EVALUATION** **Control of Special Processes**	72.24, 72.122, 72.140, 72.158	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.9.1	**QUALITY ASSURANCE EVALUATION** **Control of Special Processes** Control Process	72.24, 72.122, 72.140, 72.150, 72.158	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Performing all processes affecting quality of items or services in accordance with instructions, procedures, drawings, checklists, travelers, or other appropriate means that specify or reference applicable codes and standards and acceptance criteria for the process (10 CFR 72.150; NQA-1/Part I/Sec II Basic 9, 1S-1 Par 2, 3.2; also see NQA-1/Part II/Subpart 2.18, Subpart 2.21). • Identifying special processes (i.e., those meeting one or both of the following conditions): (a) process results that are highly dependent on control of the process or skill of the operators, or both; and (b) quality of results that cannot be readily determined by inspection or test of the product (e.g., welding, heat treating, and nondestructive testing) (10 CFR 72.158; NQA-1/Part I/Sec II Basic 9; NUREG-0800; NUREG-1536). • Preparing appropriate instructions for each special process that include or reference requirements for qualifying procedures, personnel, and equipment (10 CFR 72.158; NQA-1/Part I/Sec II Basic 9, 9S-1 Par 3; NUREG-0800; NUREG-1536). • Performing special processes in accordance with those instructions and applicable codes, standards, specifications, criteria, and other special requirements (10 CFR 72.158; NQA-1/Part I/Sec II Basic 9, 9S-1 Par 3; NUREG-1536).
Licensing, Quality Assurance	12.4.9.2	**QUALITY ASSURANCE EVALUATION** **Control of Special Processes** Qualification	72.24, 72.122, 72.140, 72.158	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Qualifying procedures, personnel, and equipment to be used in special processes in accordance with applicable codes, standard, and specifications. • Maintaining records of such qualifications (10 CFR 72.158; NQA-1/Part I/Sec II Basic, 9S-1 Par 3.1.1, 3.3; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.10	**QUALITY ASSURANCE EVALUATION** **Licensee Inspection**	72.24, 72.122, 72.140, 72.160	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.10.1	**QUALITY ASSURANCE EVALUATION** **Licensee Inspection** Scope of Inspection Program	72.24, 72.122, 72.140, 72.160	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for verifying conformance of items and activities to specified requirements by planning and conducting: • Receiving and pre-service inspections. • In-process inspections. • Final inspections. • In-service inspections, including inspections of modifications, repairs, and replacements (10 CFR 72.160; NQA-1/Part I/Sec II Basic 10, 10S-1 Par 6,7,8; also see NQA-1/Part II/Subpart 2.4).
Licensing, Quality Assurance	12.4.10.2	**QUALITY ASSURANCE EVALUATION** **Licensee Inspection** Implementation of Program (Part 1 of 2)	72.24, 72.122, 72.140, 72.160	The applicant must have responsibilities assigned and instructions and procedures issued for: • Preparing inspection instructions, procedures, and checklists that: (a) prescribe frequency of inspections or identify occasions on which testing is required, including inspections prescribed by documents establishing mandatory hold points, and modification, repair, or replacement of items; (b) specify sampling procedures if acceptability of batch materials or groups of items is to be based on inspection of a sample from the batch or group; (c) identify characteristics of items and activities to be inspected to verify conformance to original design and inspection requirements (or acceptable alternatives for modifications, repairs, and replacements); (d) describe means of inspection (e.g., examinations, measurements, tests) of each item and activity to be inspected; (e) describe methods of monitoring processing methods, equipment, and personnel if inspection of processed items is impossible or disadvantageous, or if a combination of inspection and monitoring is required, for an item or activity; (f) establish acceptance criteria for inspections and monitoring; and (g) provide for recording names of inspectors and data recorders used, and objective evidence of results of inspections and monitoring (10 CFR 72.160; NQA-1/Part I/Sec II Basic 10, 10S-1 Par 5; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.10.2	**QUALITY ASSURANCE EVALUATION** **Licensee Inspection** Implementation of Program (Part 2 of 2)	72.24, 72.122, 72.140, 72.160	• Performing inspections in accordance with the instructions, procedures, and checklists, and using inspection personnel who: (a) are qualified to perform the inspection task; (b) did not perform or supervise the work being inspected; and (c) do not report directly to supervisors who are responsible for the work (10 CFR 72.160; NQA-1/Part I/Sec II Basic 10, 10S-1 Par 3; NUREG-1536). • Qualifying and certifying personnel to be used in inspections or monitoring in accordance with requirements of applicable codes, standards, and specifications (10 CFR 72.160; NQA-1/Part I/Sec II Basic 10, 10S-1 Par 3.2; NUREG-1536).
Licensing, Quality Assurance	12.4.11	**QUALITY ASSURANCE EVALUATION** **Test Control**	72.24, 72.122, 72.140, 72.162	No text provided.
Licensing, Quality Assurance	12.4.11.1	**QUALITY ASSURANCE EVALUATION** **Test Control** Scope of Test Program	72.24, 72.122, 72.140, 72.162	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for verifying conformance of items (including safety-related computer programs) and activities to specified requirements by planning and conducting: prototype qualification tests, production tests, proof tests before installation, construction tests, pre-operational tests, and operational tests (10 CFR 72.162; NQA-1/Part I/Sec II Basic 11, 11S-1 Par 2; also see NQA-1/Part II/Subpart 2.4, Subpart 2.7).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.11.2	**QUALITY ASSURANCE EVALUATION** **Test Control** Test Control Process	72.24, 72.122, 72.140, 72.162	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying testing required to demonstrate that SSCs important to safety can perform satisfactorily in service. • Preparing test procedures that include or reference documents containing: (a) test objectives; (b) prerequisites for the test (e.g., condition, state, or configuration of the item or facility; required environmental conditions; instrumentation and personnel required); (c) applicable design, procurement document, and facility license requirements; (d) instructions and procedures for performing the test and recording results; and (e) instructions and procedures for evaluating test results (10 CFR 72.162; NQA-1/Part I/Sec II Basic 11, 11S-1 Par 3; also see NQA-1/Part II/Subpart 2.4, Subpart 2.7; NUREG-0800; NUREG-1536). • Performing testing and evaluating results in accordance with the instructions and procedures (10 CFR 72.162; NQA-1/Part I/Sec II Basic 11, 11S-1 Par 3, 4; 11S-2; also see NQA-1/Part II/Subpart 2.4, Subpart 2.7; NUREG-0800; NUREG-1536). • Preparing test records that identify, at a minimum, the item tested, date of test, names of test personnel or data recorder, type of observation, results and acceptability, action taken in connection with any deviations noted, and persons evaluating test results (10 CFR 72.162; NQA-1/Part I/Sec II Basic 11, 11S-1 Par 5 (see 11S-2 for computer programs); NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.12	**QUALITY ASSURANCE EVALUATION** **Control of Measuring and Test Equipment**	72.24, 72.122, 72.140, 72.164	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.12.1	**QUALITY ASSURANCE EVALUATION** **Control of Measuring and Test Equipment** Control Process (Part 1 of 2)	72.24, 72.122, 72.140, 72.164	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Selecting measuring and test equipment (MTE) for use in processes, inspections, and tests that: (a) is of the type appropriate for measuring specified physical characteristics of items being processed, inspected, or tested (e.g., appropriate instruments, tools, gauges, fixtures, reference and transfer standards, nondestructive inspection and test equipment, or other devices) (10 CFR 72.164; NUREG-0800; NUREG-1536); and (b) has range, accuracy, and tolerance sufficient to determine conformance of specified physical characteristics to specified requirements (10 CFR 72.164; NQA-1/Part I/Sec II 12S-1 Par 2). • Identifying each item of MTE in such a way that it can be: (a) distinguished from all others of similar type (e.g., by permanent markings such as serial numbers or by applied labels); and (b) traced to calibration data for it (e.g., by tagging, labeling, and/or documentation) (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12; NUREG-0800; NUREG-1536). • Prescribing methods and frequency (or occasions, such as immediately before use) for calibration and adjustment of MTE, on the basis of type of equipment, stability characteristics, required accuracy, intended use, and other conditions affecting measurements (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 3.2; NUREG-0800; NUREG-1536). • Performing calibration, adjustment, and maintenance of MTE on prescribed frequencies or occasions, or when its accuracy is suspect, against: (a) certified equipment having known, valid relationships to nationally recognized standards; or (b) documented bases for calibration if no nationally recognized standards exist (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 3.1; also see Part II/Subpart 2.16; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.12.1	**QUALITY ASSURANCE EVALUATION** **Control of Measuring and Test Equipment** Control Process (Part 2 of 2)	72.24, 72.122, 72.140, 72.164	• Indicating status (acceptable or not) of MTE "as found" and after each calibration, adjustment, or maintenance, by tagging or documentation traceable to the MTE (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 3.2; NUREG-1536). • Tagging or segregating out-of-calibration MTE to prevent its use until it has been re-calibrated (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 3.2; NUREG-1536). • Evaluating and documenting validity of previous inspection or test results, and acceptability of items previously inspected or tested when MTE is found to be out of calibration (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 3.2; NUREG-0800; NUREG-1536). • Repairing or replacing equipment found to be consistently out of adjustment (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 3.2). • Handling and storing such MTE so that its accuracy is maintained (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 4). • Maintaining records of calibration status that are traceable to the MTE (10 CFR 72.164; NQA-1/Part I/Sec II Basic 12, 12S-1 Par 5; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.12.2	**QUALITY ASSURANCE EVALUATION** **Control of Measuring and Test Equipment** Commercial Devices	72.24, 72.122, 72.140, 72.164	Commercial equipment such as rules, tape measures, and levels need not be subjected to the above control process if normal commercial equipment provides adequate accuracy. (10 CFR 72.164; NQA-1/Part I/Sec II 12S-1, Par 3.3).
Licensing, Quality Assurance	12.4.13	**QUALITY ASSURANCE EVALUATION** **Handling, Storage, and Shipping Control**	72.24, 72.122, 72.140, 72.166	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.13.1	**QUALITY ASSURANCE EVALUATION** **Handling, Storage, and Shipping Control** Control Process	72.24, 72.122, 72.140, 72.166	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Preparing and issuing work and inspection instructions, drawings, specifications, shipment instructions, or other documents or procedures prescribing activities to prevent damage or deterioration of items during handling, storage, and shipping (10 CFR 72.166; NQA-1/Part I/Sec II Basic 13, 13S-1 Par 2; NUREG-0800; NUREG-1536). • Identifying requirements for special equipment (e.g., containers, shock absorbers, accelerometers), special protective environments (e.g., inert gas, moisture content, temperatures), or special procedures for particular items (10 CFR 72.166; NQA-1/Part I/Sec II Basic 13, 13S-1 Par 3.1, 3.2; NUREG-0800; NUREG-1536). • Conducting cleaning, preserving, handling, storing, packing, and shipping activities in accordance with the prepared instructions, procedures, and special requirements, by using appropriately- trained and qualified personnel and appropriate tools and equipment (10 CFR 72.166; NQA-1/Part I/Sec II Basic 13, 13S-1 Par 3.4; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.13.2	**QUALITY ASSURANCE EVALUATION** **Handling, Storage, and Shipping Control** Tools and Equipment	72.24, 72.122, 72.140, 72.166	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying special handling tools and equipment that are required for safe handling of items. • Inspecting and testing such tools and equipment in accordance with specified procedures and at specified intervals to verify that they are adequately maintained (10 CFR 72.166; NQA-1/Part I/Sec II Basic 13, 13S-1 Par 3.3; NUREG-1536).
Licensing, Quality Assurance	12.4.13.3	**QUALITY ASSURANCE EVALUATION** **Handling, Storage, and Shipping Control** Markings	72.24, 72.122, 72.140, 72.166	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for marking or labeling items being handled, shipped, or stored to identify the items and any special environments or controls they require (10 CFR 72.166; NQA-1/Part I/Sec II Basic 13, 13S-1 Par 4; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.14	**QUALITY ASSURANCE EVALUATION** **Inspection, Test, and Operating Status**	72.24, 72.122, 72.140, 72.168	No text provided.
Licensing, Quality Assurance	12.4.14.1	**QUALITY ASSURANCE EVALUATION** **Inspection, Test, and Operating Status** Inspection and Test Status Process	72.24, 72.122, 72.140, 72.168(a)	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Indicating the status of inspections and tests of individual items of the facility by markings such as stamps, tags, labels, and routing cards, or in documents accompanying or traceable to the item. • Preventing installation, use, or further processing of an item that has not passed prerequisite inspections and tests, by tagging, labeling, or segregating, and by procedural controls. • Specifying authority and procedures for applying or removing inspection and test status stamps, tags, markings, and labels (10 CFR 72.168(a); NQA-1/Part I/Sec II Basic 14; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.14.2	**QUALITY ASSURANCE EVALUATION** **Inspection, Test, and Operating Status** Operating Status Process	72.24, 72.122, 72.140, 72.168(b)	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Preventing inadvertent use or operation of a structure, system, or component of the facility by indicating its operating status on tags or markings on control panels, switches, and other locations where its use or operation can be initiated. • Specifying authority and procedures for applying or removing operating status tags and markings (10 CFR 72.168(b); NQA-1/Part I/Sec II Basic 14; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.15	**QUALITY ASSURANCE EVALUATION** **Nonconforming Materials, Parts, or Components**	72.24, 72.122, 72.140, 72.170	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.15.1	**QUALITY ASSURANCE EVALUATION** **Nonconforming Materials, Parts, or Components** Control Process (Part 1 of 2)	72.24, 72.122, 72.140, 72.170	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying materials, parts, or components that do not conform to the applicant's requirements (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 2; NUREG-0800; NUREG-1536). • Segregating such items and materials in designated hold areas, or by other precautions if segregation is not practicable to prevent their inadvertent use or installation (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 3, 4.1; NUREG-0800; NUREG-1536). • Identifying individuals or groups with authority to perform evaluations and approve disposition of nonconforming items and materials (e.g., use-as-is, reject, repair, rework) (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 4.2). • Conducting evaluations in accordance with approved instructions and procedures using personnel with: (a) demonstrated competence in the area they are evaluating; (b) an adequate understanding of the requirements; and (c) access to pertinent background information (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 4.3). • Applying design control measures to items dispositioned "use-as-is" or "repair" commensurate with those applied in the original design (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 4.4; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.15.1	**QUALITY ASSURANCE EVALUATION** **Nonconforming Materials, Parts, or Components** Control Process (Part 2 of 2)	72.24, 72.122, 72.140, 72.170	• Documenting the nonconformance and disposition, with information in the documentation, including: (a) identification of the nonconforming item or material by heat, batch, lot, part, or serial number; (b) identification of the specification or requirement that was not met; (c) description of the nonconformance; (d) disposition of the nonconforming item or material; (e) technical justification for "use-as-is" or "repair" dispositions; and (f) individual or group authorizing the disposition (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 4.4; NUREG-0800; NUREG-1536). • Conducting reinspections and retests of items that are repaired or reworked against the original acceptance criteria unless the disposition of the nonconforming item establishedalternate acceptance criteria (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15, 15S-1 Par 4.5; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.15.2	**QUALITY ASSURANCE EVALUATION** **Nonconforming Materials, Parts, or Components** Notification	72.24, 72.122, 72.140, 72.170	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for notifying affected organizations of nonconforming items (10 CFR 72.170; NQA-1/Part I/Sec II Basic 15).
Licensing, Quality Assurance	12.4.16	**QUALITY ASSURANCE EVALUATION** **Corrective Action**	72.24, 72.122, 72.140, 72.172	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.16.1	**QUALITY ASSURANCE EVALUATION** **Corrective Action** Initiation Process	72.24, 72.122, 72.140, 72.172	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Identifying conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances. • Identifying a subset of such conditions that are significant conditions adverse to quality. • Reporting significant conditions adverse to quality to appropriate levels of management (10 CFR 72.172; NQA-1/Part I/Sec II Basic 16; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.16.2	**QUALITY ASSURANCE EVALUATION** **Corrective Action** Correction Process	72.24, 72.122, 72.140, 72.172	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Determining the cause of significant conditions adverse to quality. • Taking action to prevent their recurrence. • Documenting action taken and reporting it to appropriate levels of management for review and approval. • Following up to verify implementation of corrective actions (10 CFR 72.172; NQA-1/Part I/Sec II Basic 16; NUREG-0800; NUREG-1536).
Licensing, Quality Assurance	12.4.17	**QUALITY ASSURANCE EVALUATION** **QA Records**	72.24, 72.122, 72.140, 72.174	No text provided.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.17.1	**QUALITY ASSURANCE EVALUATION** **QA Records** Scope of Records Program	72.24, 72.122, 72.140, 72.174	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Generating, supplying, or maintaining, by or for the applicant, records of the quality of SSCs and activities important to safety (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 2.2; NUREG-1536). • Generating, supplying, or maintaining records including: (a) design records; (b) procurement records; (c) records of configuration of SSCs important to safety, "as built" and as changed; (d) records of conformance to operations requirements and constraints; (e) records of qualification of personnel, procedures, and equipment; (f) records of inspections, tests, surveillances, audits, and assessments; and (g) records of nonconformances (for material, items, processes and activities), disposition of nonconforming items and materials, and actions to correct nonconformances in process and activities (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 2.2; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.17.2	**QUALITY ASSURANCE EVALUATION** **QA Records** Implementation (Part 1 of 2)	72.24, 72.122, 72.140, 72.174	The applicant must have responsibilities assigned and, prior to implementation, instructions and procedures issued for: • Specifying in design specifications, procurement documents, operational procedures, or other documents the types of records to be generated, supplied, or maintained by or for the applicant (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 2.2; NUREG-1536). • Preparing records (individual records or groups of records assembled into record packages), including: (a) identifying the item, process, or activity to which the record applies; (b) assigning unique identification to each record; and (c) validating (e.g., stamping, initialing, signing and dating) or otherwise authenticating records (e.g., by a statement from the responsible individual or organization) (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 2.3, 2.6; NUREG-1536). • Distributing and handling records, including: (a) identifying the individual having custody of each record by a system of receipt control; and (b) specifying measures to be taken by the individual having custody to protect them from damage or loss (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 3; NUREG-1536). •Classifying records (lifetime or nonpermanent) and indexing records (at a minimum, by retention times and location within the records system) (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 2.4, 2.7; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.17.2	**QUALITY ASSURANCE EVALUATION** **QA Records** Implementation (Part 2 of 2)	72.24, 72.122, 72.140, 72.174	• Storing, preserving, and protecting records in storage, including: (a) providing storage facilities meeting requirements of applicable codes, standards, and regulations; and (b) specifying procedures for receiving, storing, preserving, and protecting records in such storage facilities (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 4; NUREG-0800; NUREG-1536). • Retrieving records, including identification of personnel who may have access to the applicant's record files, and procedures for retrieving records maintained by suppliers or contractors (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 5). • Disposition of records, including ensuring that disposition of records is governed by the most stringent regulatory requirements that apply to the records (this may be an agency other than the NRC) and ensuring that supplier's nonpermanent records are not disposed of until the following conditions are met: (a) items are released for shipment and a Code Data Report is signed, or a Code Symbol Stamp is affixed; (b) regulatory requirements are satisfied; (c) operational status permits; (d) warranty consideration is satisfied, and (e) purchaser's requirements are satisfied (10 CFR 72.174; NQA-1/Part I/Sec II Basic 17, 17S-1 Par 6).
Licensing, Quality Assurance	12.4.18	**QUALITY ASSURANCE EVALUATION** **Audits**	72.24, 72.122, 72.140, 72.176	No text provided.
Licensing, Quality Assurance	12.4.18.1	**QUALITY ASSURANCE EVALUATION** **Audits** Scope of Audit Program	72.24, 72.122, 72.140, 72.176	The applicant's audit program must address planning and performance of audits to: • Verify compliance with specifications and other requirements for activities affecting quality. • Determine the effectiveness of the QA program. • Verify that supplier's and contractor's QA programs comply with applicable requirements of 10 CFR 72 and meet acceptance criteria equivalent to those for such requirements in this Standard Review Plan (SRP) (10 CFR 72.176; NQA-1/Part I/Sec II Basic 18, 18S-1; NUREG-0800; NUREG-1536).

A-2. DETAILED REQUIREMENTS OF NUREG-1567

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.18.2	**QUALITY ASSURANCE EVALUATION** **Audits** Implementation (Part 1 of 2)	72.24, 72.122, 72.140, 72.176	The applicant must have responsibilities assigned and instructions and procedures issued for: • Initiating audits early enough to ensure effective QA of activities during the early stages of the installation life cycle (e.g., design, procurement) (10 CFR 72.176, 72.154; NQA-1/Part I/Sec II 18S-1 Par 4; NUREG-0800; NUREG-1536). • Scheduling audits at a frequency commensurate with the status and importance of the activity audited (10 CFR 72.176; NQA-1/Part I/Sec II 18S-1 Par 2; NUREG-0800; NUREG-1536). • Preparing written audit plans for each audit that identify the audit scope, requirements, audit personnel, activities to be audited, organizations to be notified, applicable documents, schedule, and written procedures or checklists (10 CFR 72.176; NQA-1/Part I/Sec II 18S-1 Par 3.1). • Selecting and assigning auditors who are members or under direction of the applicant QA organization and who are independent of any direct responsibilities for performance of the activities they audit (10 CFR 72.176; NQA-1/Part I/Sec II 18S-1 Par 3.2, 3.3; NUREG-1536).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Quality Assurance	12.4.18.2	**QUALITY ASSURANCE EVALUATION** **Audits** Implementation (Part 2 of 2)	72.24, 72.122, 72.140, 72.176	• Performing audits with an audit team led by a certified lead auditor in accordance with written procedures and checklists for the audit (10 CFR 72.176; NQA-1/Part I/Sec II 18S-1 Par 3.2, 3.3; NUREG-1536). • Preparing an audit report for each audit and requesting a response from the audited organization, additionally requiring that the aduit report be signed by the audit team leader and include: (a) a description of the audit scope; (b) identification of the auditors; (c) identification of persons contacted during audit activities; (d) a summary of audit results, including a statement about the effectiveness of the QA program elements that were audited; and (e) a description of each adverse audit finding in sufficient detail to enable the audited organization to take corrective action (10 CFR 72.176; NQA-1/Part I/Sec II Basic 18, 18S-1 Par 5; NUREG-1536). • Evaluating and concurring in corrective action planned by the audited organization and taking follow up action to verify that agreed-upon corrective action is performed as scheduled (10 CFR 72.176; NQA-1/Part I/Sec II Basic 18, 18S-1 Par 6, 7; NUREG-1536). • Maintaining audit plans, audit reports, written replies, and records of completion of corrective action as QA records (10 CFR 72.176; NQA-1/Part I/Sec II Basic 18, 18S-1 Par 8).
Licensing, Decommissioning	13.4	**DECOMMISSIONING EVALUATION** **Acceptance Criteria**	72.24, 72.30	The ISFSI must be decommissioned at the end of service life, and every effort must be made to terminate the license and release the ISFSI site for unrestricted use according to the requirements of 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination." The requirements related to eventual decommissioning the ISFSI or MRS applicable at the time of initial licensing are satisfied if the applicant adequately addresses the acceptance criteria for design features, operational features, and decommissioning plan.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Decommissioning	13.4.1	**DECOMMISSIONING EVALUATION** **Design Features**	72.24, 72.30	The application must identify the design features included in the design of the ISFSI or MRS that will facilitate decontamination and decommissioning. This information may be in the Safety Analysis Report (SAR) or in the decommissioning plan. Design features include surfaces that are less susceptible to contamination (or activation) and are readily decontaminated, as well as shielding to minimize any occupational exposure associated with decontamination. Design features also include equipment to facilitate the decontamination and removal of air circulation and filtration systems, and components of waste treatment and packaging systems.
Licensing, Decommissioning	13.4.2	**DECOMMISSIONING EVALUATION** **Operational Features**	72.24, 72.30	The application must identify the operational features that will facilitate eventual decontamination and decommissioning of the ISFSI or MRS. Such features include minimizing contamination buildup on components, maintaining accurate records of spills or other unusual occurrences involving the spread of contamination, and maintaining accurate as-built drawings or suitable substitutions. This information is in either the SAR or the decommissioning plan, and includes technical specifications or aspects of the proposed quality assurance (QA) program.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Decommissioning	13.4.3	**DECOMMISSIONING EVALUATION** **Decommissioning Plan** (Part 1 of 2)	72.24, 72.30	The application must include a proposed decommissioning plan as required by 10 CFR 72.30. The plan must describe the proposed practices and procedures for (a) the decontamination of the site and facilities, and (b) the disposal of residual radioactive materials after the stored spent fuel or high-level waste has been removed. Design features of the ISFSI or MRS that facilitate its decommissioning at the end of its useful life must be identified and discussed. The plan should provide reasonable assurance that the proposed decontamination and decommissioning of the ISFSI or MRS will adequatly protect public health and safety and will leave the site suitable for unrestricted use. A site is considered acceptable for unrestricted use if (a) residual radioactivity has been reduced to levels that are as low as is reasonably achievable (ALARA), and (b) compliance with other radiological criteria of 10 CFR 20.1402 can be demonstrated. The decommissioning plan submitted with the license application need not comply with the form and content requirements of Regulatory Guide 3.65, "Standard Format and Content of Decommissioning Plans for Licensees Under 10 CFR Parts 30, 40, and 70." Regulatory Guide 3.65 provides guidance on the content and format of final decommissioning plans submitted at the time of license termination. As part of the decommissioning plan, the application must contain a funding plan, which, in turn, includes a cost estimate for the decommissioning and a financial assurance mechanism that will ensure availability of funds in the amount of the cost estimate.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Decommissioning	13.4.3	**DECOMMISSIONING EVALUATION** **Decommissioning Plan** (Part 2 of 2)	72.24, 72.30	Guidance on the format and content of the financial assurance mechanism and the means for cost estimating are provided in Regulatory Guide 3.66, "Standard Format and Content of Financial Assurance Mechanisms Required for Decommissioning Under 10 CFR 30, 40, 70 and 72." A legal, executed copy of the financial assurance mechanism must be provided. Acceptance criteria are provided in NUREG-1337, Rev. 1, "Standard Review Plan (SRP) for the Review of Financial Assurance Mechanisms for Decommissioning Under 10 CFR Parts 30, 40, 70, and 72." The funding plan must be signed (i.e., certified) by an individual authorized to make financial commitments for the applicant.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Acceptance Criteria** (Part 1 of 2)	20.1101, 20..1301, 20,1302, 20.2001, 20.2003, 72.24, 72.40, 72.104, 72.122, 72.126, 72.128	The principal acceptance criteria that apply to confinement and management of site-generated waste are based on meeting the following regulations: • 10 CFR 72.104, as it relates to sufficient information being provided to demonstrate that the proposed ISFSI or MRS waste storage and management system has been designed and will be operated so that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area does not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ. • 10 CFR 20.1302, as it relates to maximum levels of radioactivity in effluents to unrestricted areas. • 10 CFR 20.1101, as it relates to constraints on air emissions of radioactive material to the environment such that no individual member of the public is likely to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions. Additional acceptance criteria apply to the descriptions provided in the SAR of waste sources and management systems, waste characteristics, operations, and monitoring. The SAR must describe the design bases for systems and equipment that maintain control over radioactive material in gaseous and liquid effluents, and identify the equipment and facilities important to safety. The SAR must also include the design objectives and the means to be employed to keep the levels of radioactivity in effluents as low as is reasonably achievable (ALARA) and to minimize the generation of waste. Waste operations, from generation and collection to final disposal offsite, must be described in the narrative descriptions and flowsheets.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Acceptance Criteria** (Part 2 of 2)	20.1101, 20..1301, 20,1302, 20.2001, 20.2003, 72.24, 72.40, 72.104, 72.122, 72.126, 72.128	Specific requirements are addressed in the following sections as they relate separately to waste sources, off-gas treatment and ventilation, liquid waste treatment and retention, and solid wastes. NUREG-0800, “Standard Review Plan,” can also be used to identify requirements that apply to acceptance criteria for these categories.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.1	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Waste Sources**	72.24, 72.104, 72.122, 72.128	Radioactive wastes that result from an ISFSI or MRS can be separated into two main categories: • Effluents -- radioactivity discharged to the environment in gaseous or liquid form. The activity content of these effluents must comply with regulatory limits and ALARA criteria. • Wastes -- radioactive materials that are of sufficient hazard or regulatory concern that they require special care before final disposal. The generation of such wastes must be ALARA. All actual and potential sources of site-generated radioactive waste must be identified in the SAR. Waste sources described must include activities that give rise to potentially radioactive wastes that would require treatment or special handling. The identification of sources must be comprehensive. Anticipated radioactive wastes must be described and classified with respect to source, chemical and radiological composition, method and design for treatment and handling, and storage mode before disposal. Sources of non-radioactive waste such as combustion products and chemical wastes must also be identified to the extent that the reviewer can ascertain whether site activities can result in radioactive materials being added to such sources. The total volume of liquid waste discharged to the environment must be estimated to provide the bases for determining concentrations of radionuclides in liquid effluents. Total sanitary sewer flow may be needed to determine concentrations of radionuclides in waste disposed to the sanitary sewer.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.2	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Off-Gas Treatment and Ventilation**	20.2001, 72.24, 72.104, 72.122, 72.126, 72.128	Off-gas treatment and ventilation systems are typically provided for removing radioactive and non-radioactive hazardous materials from the atmosphere within a confinement barrier before being released in the environment. The SAR must provide flowsheet and narrative descriptions of off-gas treatment and ventilation system operations. It must also identify design criteria and applicable regulatory limits. General design criteria must be based on site conditions and accident-level and off-normal analyses, design objectives, and projected volumes of gaseous (or airborne) waste. The SAR must also indicate those radioactive wastes that will be produced as a result of off-gas treatment. The applicant must show that system capacity is consistent with the confinement system requirements during normal and off-normal conditions. The descriptions must also address replacement and disposal of items such as filters and scrubber solutions, as well as any transfers of wastes to other waste treatment systems. The design must address the potential for personnel exposures and contamination that can result from handling operations. Continuous monitoring systems must be provided to detect effluent radioactivity and to alarm on high effluent activity. Monitoring systems are addressed in SRP Sections 9.4.2 and 11.4.2.
Licensing, Design, Operation	14.4.3	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention**	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	The SAR must provide flowsheet and narrative descriptions of the liquid waste treatment system and associated design criteria and regulatory limits.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.3.1	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Design Objectives	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	Basic liquid waste treatment concepts include volume reduction, immobilization of radioactive elements, change of composition, and removal of radioactive elements from the waste stream. The description of the facility liquid waste treatment and retention systems must identify the design objectives and demonstrate that the system can handle the expected volume of potentially radioactive and non-radioactive hazardous wastes generated during normal and off-normal operations. In general, engineered features should be emphasized over procedures to meet protective requirements.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.3.2	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Equipment and System Description (Part 1 of 2)	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	The SAR must describe the features, systems, and special handling techniques that are important to safety. Drawings must include location of equipment, flow paths, piping, valves, instrumentation, and other physical features. Seismic and quality group classification must conform to the guidelines of Regulatory Guide 1.1.43, “Design Guidance for Radioactive Waste Management Systems, Structures and Components in Light-Water-Cooled Nuclear Reactor Power Plants.” Where feasible, gravity flow must be used to reduce pressure and to avoid or minimize contamination of pumping and pressure system equipment. Measurement capability must be provided to determine the volume, concentration, and radioactivity of wastes fed into collection tanks. The SAR must identify the sources of all liquid wastes generated and their flow into and out of the liquid treatment systems. Measurement capability must be provided to determine the volume and radioactivity of wastes fed to the collection system. Individual lines must be used for each waste stream fed to the central collection system, where necessary, to prevent chemical reactions or introduction of contaminants such as complexing agents that can interfere with waste decontamination. Individual lines outside confinement (and liquid containment) barriers must be designed not to rupture in the event of frost heave, earth or structure settlement, or earth-structure motions during design basis earthquakes. A separate confinement barrier (e.g., drained outer pipe or drained tunnel) must be provided for these lines.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.3.2	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Equipment and System Description (Part 2 of 2)	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	Spills, overflows, or leakage from storage vessels must be collected or retained within a suitable secondary confinement structure (e.g., secondary vessel, elevated threshold, or dike). A capability must exist to transfer liquid from the secondary confinement to a suitable storage location. All transfer lines must have individual identification. The piping must be designed to minimize entrapment and buildup of solids in the system. Bypasses which route waste streams around collection tanks must be avoided. Provisions must be made for clean out or decontamination of liquid waste piping, as necessary, to clear potential blockages, perform maintenance or repair, or maintain occupational doses ALARA. Volume reduction or solidification methods may be used to process liquid wastes. Redundancy and other special features may be incorporated to safely confine the wastes. Adequate shielding must be provided for radioactive liquid waste system components, as necessary. The SAR must describe how influents to radioactive liquid waste systems are controlled (as necessary, depending on the sources) to prevent introduction of material that may adversely affect system performance. Such materials include, but are not limited to, oils, other organics, insoluble solids, solvents, and hazardous wastes.
Licensing, Design, Operation	14.4.3.3	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Operating Procedures	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	The flowsheets and narrative descriptions of operations must describe the design features and procedures that minimize generation of liquid waste and the possibility of spills, and they must provide for control and containment of spills. The SAR must state whether the procedures include performance tests, action levels, actions to be taken under normal and off-normal conditions, and methods for testing to ensure functional operation.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.3.4	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Characteristics, Concentrations, and Volumes of Solidified Wastes	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	The physical, chemical, and thermal characteristics of solidified (extracted or residue of liquid) wastes must be described. These wastes, or "characteristics," must be compatible with estimates of concentrations and volumes generated. They must also be compatible with the design ratings of the selected liquid waste treatment and retention systems.
Licensing, Design, Operation	14.4.3.5	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Packaging	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	The SAR must describe the packaging for solidified wastes. The package information must show the materials of construction, and include welding information. It must also show the maximum temperatures for waste and container at the highest design heat loads, the homogeneity of the waste contents, the corrosive interactions of the waste on materials of construction, the means for preventing over pressurization of the package, and the confinement provided by the package under off-normal conditions. If standard low-level waste containers (e.g., DOT-approved drums) are to be used for packaging, they must be identified. Otherwise, packaging details including vendor, make, model, and full manufacturer’s catalog information must be provided in the SAR or the supporting documentation. Suitability of packaging for holding and storage of wastes onsite at the designated location must be demonstrated. Aspects of the operating quality assurance program that specifically apply to solidified waste packaging must also be described.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.3.6	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Liquid Waste Treatment and Retention** Storage Facilities	20.2001, 20.2003, 72.24, 72.104, 72.122, 72.126, 72.128	The SAR must describe the storage facilities for site-generated liquid or solidified waste. Movement of containers into and out of storage, and monitoring must be described. Equipment, waste routing, and spare storage volume must be installed and available to transfer the contents of one tank to another. The minimum spare volume must exceed the maximum liquid content of any one tank. Provisions must be made so that liquids can be analyzed before transfer. Agitators must be included in storage vessels, when necessary, to promote mixing of the waste to ensure uniform decay heat distribution, minimize settling, or provide representative waste samples. If liquid wastes are to be held until site decommissioning or for radioactive decay, the SAR must demonstrate (by analyses or relevant experiential data) that the storage capability is appropriate for the duration of the ISFSI or MRS life and the chemistry of the contents.
Licensing, Design, Operation	14.4.4	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes**	72.24, 72.104, 72.122, 72.128	The SAR must describe the solid wastes produced during ISFSI or MRS operations. The wastes must be listed and characterized (see Section 14.4.4.4), and systems used to treat, package, and contain these wastes must be described in terms of radionuclide content, container size, and generation rate.
Licensing, Design, Operation	14.4.4.1	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes** Design Objectives	72.24, 72.104, 72.122, 72.128	The SAR must identify the design objectives and demonstrate that the system can handle the expected volume of potentially radioactive solid wastes generated during normal and off-normal operations. The design objectives must reflect waste minimization as well as safe management. If the design basis includes regulatory limits, these limits must be identified.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.4.2	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes** Equipment and System Description	72.24, 72.104, 72.122, 72.128	The SAR must describe the features, systems, and special handling techniques that are important to safety. Drawings must identify locations of equipment and associated features that will be used for volume reduction, confinement, packaging, storage, and disposal. The SAR must identify the source of all solid wastes generated and their flows into and out of the solid waste treatment systems. Fundamental solid waste treatment concepts include volume reduction, immobilization of radioactive material, change of composition, and removal of radioactive material from the waste stream. Solid waste management systems must include provisions for shielding, confinement, handling, and decontamination, as necessary, to ensure that occupational doses are maintained ALARA.
Licensing, Operation	14.4.4.3	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes** Operating Procedures	72.24, 72.104, 72.122, 72.128	The SAR must describe the procedures associated with solid waste system or equipment operations. The procedures must identify performance or functional testing, process limits, action levels, and actions to be taken under normal and off-normal conditions. The means for monitoring and controlling limits must also be described.
Licensing, Design, Operation	14.4.4.4	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes** Characteristics, Concentrations, and Volumes of Solid Wastes	72.24, 72.104, 72.122, 72.128	The SAR must describe the physical, chemical, and thermal characteristics of the solid wastes, and provide estimates of the waste volume and radionuclide concentrations. Those estimates must be consistent with design ratings of the solid waste treatment and retention systems.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	14.4.4.5	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes** Packaging	72.24, 72.104, 72.122, 72.128	The SAR must describe the packaging for solid wastes (as for solidified liquid waste, described in Section 14.4.3.5). Aspects of the operating quality assurance program that specifically apply to solid waste packaging must also be described. If a laundry is to be used (e.g., to minimize solid waste generation), the containers for transferring the used items must be described. If the laundry is offsite, it must be identified and must be licensed to possess radioactive material of the type and quantity to be generated at the ISFSI or MRS. (Note: An offsite laundry is not licensed under 10 CFR 72, but an on-site laundry capability to support the ISFSI or MRS can be included in the installation license.)
Licensing, Design, Operation	14.4.4.6	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Solid Wastes** Storage Facilities	72.24, 72.104, 72.122, 72.128	The SAR must describe the solid waste storage facilities. Movement of packages into and out of storage and monitoring to be performed must be described. Corrosive aspects of the wastes and monitoring of the confinement barrier must be addressed. Planned disposal of the wastes must be described. If solid wastes are to be held until site decommissioning or for radioactive decay, the SAR must demonstrate (by analyses or relevant operating experience) that the storage containers/confinement are appropriate for the duration of the ISFSI or MRS life or the projected decay holding time. The SAR must also show how the wastes will be handled at the time the installation is permanently decommissioned.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Operation	14.4.5	**WASTE CONFINEMENT AND MANAGEMENT EVALUATION** **Radiological Impact of Normal Operations**	20.1101, 20..1301, 20,1302, 72.40	Regulatory Guide 3.48 requires that the SAR provide a summary of the radiological impacts of wastes generated during normal site operations. Information must include: • A summary identifying each effluent and waste type • The amount of each waste type generated per metric ton of spent fuel or high-level waste handled and stored per unit of time (e.g., per year) • The quantity and concentration of each principal radionuclide in each waste stream • Identification of the locations beyond the restricted areas (as defined in 10 CFR 20.1003) and beyond the controlled area (as defined in 10 CFR 72.3) that are potentially affected by radioactive materials in effluents • The estimated concentrations of principal radionuclides at the locations identified and the collective (person-rem) dose to human occupants at these locations, including the contribution of each principal radionuclide to the dose • Sample calculations and a discussion of the reliability of the concentration and dose estimates • For each effluent, a summary of the constraints imposed on process systems and equipment to ensure safe operation. When combined with other site effluents (e.g., those emitted from the material being stored), the radionuclide concentrations in gaseous or liquid effluents must not exceed the concentration limits specified in Table 2 of Appendix B to Part 20, and the resultant doses must not exceed the applicable criteria of 10 CFR 72.104. Constraints on air emissions must ensure that no member of the public receives a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from those emissions.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	15.4	**ACCIDENT ANALYSIS** **Acceptance Criteria** (Part 1 of 2)	72.24, 72.90, 72.92, 72.94, 72.104, 72.106, 72.122, 72.124, 72.126, 72.128, 72.236	This section identifies the acceptance criteria used for each accident in the accident analysis review. The SAR must include complete information for each event, analysis of the safety performance of the system, and demonstration of compliance with all applicable regulations. Each evaluation should include (1) the cause of the event, (2) means of detecting the event, (3) a summary of the analysis of the event, including estimated consequences and comparison to regulatory limits, and (4) a corrective course of action. All events must meet the following acceptance criteria regarding criticality, confinement, retrievability, and instrumentation. Dose limit criteria are discussed in specific subsections. *Criticality* : 10 CFR 72.124(a) and 72.236(c) require that the spent fuel must be maintained in a subcritical condition (i.e., keff equal to or less that 0.95) under credible conditions. At least two unlikely, independent and concurrent or sequential changes must be postulated to occur in the conditions essential to nuclear criticality safety before a nuclear criticality accident is possible (double contingency).

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	15.4	**ACCIDENT ANALYSIS** **Acceptance Criteria** (Part 2 of 2)	72.24, 72.90, 72.92, 72.94, 72.104, 72.106, 72.122, 72.124, 72.126, 72.128, 72.236	*Confinement* : 10 CFR 72.128(a)(3) and 72.236(d) and (l) require that the systems important to safety must be evaluated, using appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably keep radioactive material confined under credible accident conditions. A breach of a confinement barrier is not acceptable for any accident event. A confinement system is defined in 10 CFR 72.3 as a system, including ventilation, that acts as a barrier between areas containing radioactive substances and the environment. *Retrievability* : 10 CFR 72.122(l) requires that ISFSI storage systems allow ready retrieval of the stored spent fuel or high-level waste for normal and off-normal design conditions. Retrievability is the capability of returning the stored radioactive material to a safe condition without endangering public health and safety or causing additional exposure to workers. Any potential release of radioactive materials during retrieval operations must not exceed the radioactive exposure limits in 10 CFR Part 20 or 10 CFR 72.122(h). *Instrumentation* : 10 CFR 72.122(h) through (j) and 72.128(a)(1) require that the SAR identify all instruments and control systems that must remain operational under accident conditions.

A-2. DETAILED REQUIREMENTS OF NUREG-1567

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	15.4.1	**ACCIDENT ANALYSIS** **Off-Normal Events**	72.24, 72.90, 72.92, 72.94, 72.104, 72.106, 72.122, 72.124, 72.126, 72.128, 72.236	In addition to the acceptance criteria stated in Section 15.4, 10 CFR 72.104 requires that the following criteria be met regarding dose limits for off-normal events. During an off-normal event, the annual dose equivalent to any individual located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to the following sources: • planned discharges to the general environment of radioactive materials (with the exception of radon and its decay products) • direct radiation from operations of the ISFSI • any other cumulative radiation from uranium fuel cycle operations (i.e., nuclear power plant) in the affected area All off-normal events listed in Section 15.2.1 must be considered.
Licensing, Design, Operation	15.4.2	**ACCIDENT ANALYSIS** **Accidents**	72.24, 72.90, 72.92, 72.94, 72.106, 72.122, 72.124, 72.126, 72.128, 72.236	In addition to the criteria stated in Section 15.4, 10 CFR 72.106(b) requires that any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. All accidents listed in Section 15.2.2 must be considered.
Licensing, Design, Operation	15.4.3	**ACCIDENT ANALYSIS** **Other Non-Specified Accidents**	72.24, 72.106, 72.122, 72.124, 72.126, 72.128, 72.236	In addition to all accidents given above, the applicant must list and evaluate other accident events that are specific to his design. If these other non-specified accidents have results that are enveloped by the accidents previously considered, the applicant must provide the basis for this evaluation, and no further consideration is required. It is expected that events such as human errors, operational errors, material aging, etc., may be enveloped by the required accidents listed in Sections 15.2.1 and 15.2.2.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design, Operation	16.4	**TECHNICAL SPECIFICATIONS** **Acceptance Criteria**	72.24, 72.26, 72.44	The technical specifications are required by 10 CFR 72.44(c) to include functional/operating limits, monitoring instruments, limiting control settings, limiting conditions, surveillance requirements, design features, and administrative controls. The applicant must identify proposed technical specifications necessary to maintain subcriticality, confinement, shielding, heat removal, and structural integrity under normal, off-normal, and accident conditions. The applicant must identify the basis for each of the proposed technical specifications by reference to the analysis in the SAR. Acceptance criteria for functional and operating limits, monitoring instruments, and limiting control settings include limits placed on fuel, waste handling, and storage conditions to protect the integrity of the fuel and container, to protect the employees against occupational exposures, and to guard against the uncontrolled release of radioactive materials. Acceptance criteria for limiting conditions are the lowest levels required for safe operation. Acceptance criteria for establishing surveillance requirements include the frequency and scope of surveillance requirements to verify performance and availability of SSCs important to safety, and the verification of the bases for the proposed limiting conditions. Acceptance criteria for administrative controls include organizational and management procedures, recordkeeping, review and audit systems, and reporting necessary to ensure that the ISFSI or MRS is managed in a safe and reliable manner. Administrative action that must be taken in the event of non-compliance with a limit or condition should be specified.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	16.4.1	**TECHNICAL SPECIFICATIONS** **Code Exceptions** (Part 1 of 4)	72.24, 72.26, 72.44	There is no existing American Society of Mechanical Engineers (ASME) Code for the design and fabrication of spent fuel dry storage casks. Therefore, the industry adopted, and NRC accepted,the use of ASME Code Section III as an acceptable standard for the design and fabrication of dry storage casks. However, since dry storage casks are not pressure vessels, ASME Code Section III cannot be implemented without allowing some exceptions to its requirements. Therefore, in the past, NRC has allowed specific exceptions to the code for those requirements that were not applicable or practical to implement for spent fuel dry cask storage systems. Early spent fuel dry cask storage licenses and certificates of compliance were issued without documenting which specific exceptions to ASME Code Section III were approved. Poor quality assurance practices during design and fabrication sometimes led to significant deviations from the Code without appropriate certificate holder design review or NRC review and approval. Therefore, the applicant should document commitments to ASME Code Section III, with proposed exceptions, in the application. Likewise the NRC should document these commitments in the 10 CFR Part 72 licenses, certificates of compliance, or technical specifications and its approval of the proposed exceptions in the SER. Also, the NRC should include a statement (in license, certificate of compliance, or technical specifications) which refers the reader to the SAR and applicable SERs for any exceptions to the codes. In addition, to ensure that similar problems do not exist in other areas, all other codes and standards applied to components important to safety should be identified in the SAR and should be included in the license, certificate of compliance, or technical specification.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	16.4.1	**TECHNICAL SPECIFICATIONS** **Code Exceptions** (Part 2 of 4)	72.24, 72.26, 72.44	Applicants should propose a condition to a license, certificate of compliance, or technical specification (Section 4, "Design Features) that describes commitments to specified codes. The condition or technical specification should also describe a process to address deviations from the applicable codes that may be necessary. In such cases, the licensee should request an exception to the requirements of the applicable code from the NRC. If the staff finds that the deviation does not adversely impact safety, it may authorize the requested exception in writing.

Category	NUREG 1567 Section	NUREG 1567 SECTION TITLE Subsection Title	10 CFR Section	Criteria Description
Licensing, Design	16.4.1	**TECHNICAL SPECIFICATIONS** **Code Exceptions** (Part 3 of 4)	72.24, 72.26, 72.44	The following is an example of a provision for allowing exceptions to applicable codes: *4.3 Codes and Standards* *The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994, is the governing Code for the storage system.* *4.3.1 Design Exceptions to Codes, Standards, and Criteria* *SAR Table XXX lists all approved exceptions for the design of the ISFSI.* *4.3.2 Construction/Fabrication Exceptions to Codes, Standards, and Criteria* *Proposed alternatives to ASME Code Section III, 1992 Edition with Addenda through 1994, including exceptions referenced in Section 4.3.1, may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee.* *The proposal to the NRC must demonstrate that the alternatives would provide an acceptable level of quality and safety, or that compliance with the specified requirements of ASME Code, Section III, 1992 Edition with Addenda through 1994, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.* *Request for exceptions should be submitted in accordance with 10 CFR 72.4.*

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Categories:
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Decommissioning
Design
General
Licensing
Operaton
Quality Assurance

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, General	1	**INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION**	1.4	Provide introductory information such as the purpose for and the general description of the installation. The information in this chapter should enable the reader to obtain a basic understanding of the installation and the protection afforded the public health and safety without having to refer to the subsequent chapters. Review of the detailed chapters that follow can then be accomplished with better perspective and with recognition of the relative safety importance of each individual item to the overall design of the installation.
Licensing, General	1.1	**INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION** **Introduction**	1.4	Briefly present the principal function and design features of the installation. Discuss the reason or need for an ISFSI (independent spent fuel storage installation). Include a brief description of the proposed location and the estimated time schedules for construction and operation.
Licensing, General	1.2	**INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION** **General Description of Installation**	1.4	Include a summary description of the principal characteristics of the site and a general description of the installation. The description should include a brief discussion of the principal design criteria; the nominal capacity of the installation; the type, form, quantities, and potential sources of the spent fuel or high-level radioactive wastes to be stored; and the waste products generated during ISFSI or MRS operations. The arrangement of major structures and equipment should be indicated on plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the installation. Any additional features likely to be of special interest because of their relationship to safety should be identified.
Licensing, General	1.3	**INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION** **General Systems Description**	1.4	A summary description of the storage mode and arrangement of the storage structure(s) to be used, including pertinent background information, should be presented. Include a brief description of the operating systems; fuel handling, decay heat removal, and other auxiliary systems; and the site-generated waste treatment system. Provide sufficient detail in the discussion and accompanying charts and tables to provide an understanding of the systems involved.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, General	1.4	**INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION** **Identification of Agents and Contractors**	1.4	Identify the prime agents or contractors for the design, construction, and operation of the installation. All principal consultants and outside service organizations, including those providing quality assurance services, should be identified. The division of responsibility among the designer, architect-engineer, constructor, and installation operator should be delineated.
Licensing, General	1.5	**INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION** **Material Incorporated by Reference**	1.4	This section should provide a tabulation of all topical reports incorporated that are by reference as part of the SAR. In this context, "topical reports" are defined as reports that have been prepared by architect-engineers or other organizations and filed separately with the NRC in support of this application or of other applications or of product lines. For each topical report, this tabulation should include the title, the report number, the date submitted to the NRC (or the Atomic Energy Commission (AEC)), and the sections of the SAR in which this report is referenced. For any topical reports that have been withheld from public disclosure pursuant to § 2.790(b) of 10 CFR Part 2 as proprietary documents, nonproprietary summary descriptions of the general con- tent of such reports should also be referenced. This section should include a tabulation of any documents submitted to the Commission in other applications that are incorporated in whole or in part in this application by reference. If any information submitted in connection with other applications is incorpo- rated by reference in this SAR, summaries of such information should be included in appropriate sections of this SAR.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, General	2	**SITE CHARACTERISTICS*** **Material Incorporated by Reference** *** Any material in this chapter that is covered in the applicant's Environmental Report (ER) may be covered by reference to the subject matter in the ER.**	2.4	Provide information on the location of the installation and a description of the geographical, demographical, meteorological, hydrological, seismological, and geological characteristics of the site and surrounding vicinity. The objective is to indicate what site characteristics influence installation design. An evaluation of the site characteristics from a safety viewpoint should be developed. Identify any assumptions that need to be applied in making the safety appraisal and that are further related by cross-reference both to the criteria developed in Chapter 4, "Installation Design," and to the design bases selected in subsequent chapters to meet these criteria. If it is planned to locate the proposed ISFSI or MRS at or in the vicinity of an existing licensed site such as a nuclear power plant, much of the required siting information may be available in previous submittals to the AEC or NRC. In such cases, it is particularly important that the applicant confer with the NRC staff prior to preparing the SAR to determine the applicability of such information.
Licensing, Design	2.1	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected**	2.4.1	Information concerning the site geography, population, access transportation routes, and land usage should be provided in support of the safety evaluation.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.1.1	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected** Site Location	2.4.1.1	The location of the site should be described with sufficient clarity to avoid any ambiguity about its location in relationship to features developed later in this chapter. The site location should be described by specifying the latitude and longitude to the nearest second and the Universal Transverse Mercator coordinates** to the nearest 100 meters. The State and county in which the site is located should be identified, as well as the location of the site relative to prominent natural and man-made features such as rivers, lakes, and the local road network. To facilitate the presentation of this information, maps and aerial photographs should be provided. The general location map should encompass at least an 8-kilometer (5-mile) radius. Additional maps should be provided to present detail near the site and site plots to establish orientation of buildings, roads, railroads, streams, ponds, transmission lines, and neighboring structures. Detail in this section may be referenced in subsequent chapters to minimize repetition. **As found on U.S. Geological Survey topographical maps.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.1.2	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected** Site Description	2.4.1.2	A map of the site should be included and should be of suitable scale clearly to define the boundary of the site and the distance from significant features of the installation to the site boundary. The area to be considered as the controlled area should be clearly delineated if its boundaries are not the same as the boundaries of the site. The application should include a description of the applicant's legal responsibilities with respect to the properties described (e.g., ownership, lease, easements). The topography of the site and vicinity should be described by suitable contour maps that indicate the character of surface drainage patterns. Vegetative cover and surface soil characteristics should be described sufficiently to indicate potential erosion and fire hazards. Traffic and transportation routes and onsite transmission lines should be identified.
Licensing, Design	2.1.2.1	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected** Site Description *Other Activities Within the Site Boundary*	2.4.1.2	For any activity con- ducted within the area controlled by the applicant but not related to the opera- tion of the ISFSI or MRS, identify the activities involved, the boundaries within which the applicant will control such activities, and any potential interaction of such activities and the operation of the ISFSI or MRS.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.1.2.2	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected** Site Description *Boundaries for Establishing Effluent Release Limits*	2.4.1.2	Identify the controlled area boundary and demarcate the area to whicch aess will be actively controlled for purposes of protection of individuals from exposure to radiation and radioactive materials. The degree of access control required is that which enables the licensee to comply with the requirements of § 72.104 of 10 CFR Part 72. The site map (discussed in Section 2.1.2) may be used to identify this area, or a separate map of the site may be used. Indicate the location of the boundary with respect to nearby rivers and lakes. The minimum distance from a proposed storage location, as well as from other possible effluent release points, to the controlled area boundary should be clearly presented.
Licensing, Design	2.1.3	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected** Population Distribution and Trends	2.4.1.3	Population information based on the most recent census data should be presented to show the population distribution as a function of distance and direction from the installation. On a map of suitable scale that identifies places of significant population grouping such as cities and towns within the 8-kilometer (5-mile) radius, concentric circles should be drawn, using the installation as the renter point, with radii of 1.5, 3, 5, 6.5, and 8 kilometers (approximately 1, 2, 3, 4, and 5 miles). The circles should be divided into 22½-degree segments with each segment centered on one of the 16 compass points (e.g., true north, north-northeast). Within each area thus formed by the concentric circles and radial lines, the current resident population, as well as projected future population changes, should be specified. The basis for the projection should be described. Significant transient or seasonal population variations should also be identified and discussed.
Licensing, Design	2.1.4	**SITE CHARACTERISTICS** **Geography and Demography of Site Selected** Uses of Nearby Land and Waters	2.4.1.4	Uses of nearby land and waters within at least an 8-kilometer (5-mile) radius should be described. Sufficient characterization of farming, dairy, industrial, residential, and recreational activities should be presented to permit estimates to be made of potential population radiation dose-commitments resulting from both airborne and liquid effluents. The localized population in facilities such as schools and institutions should be identified with respect to location and number of persons.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.2	**SITE CHARACTERISTICS** **Nearby Industrial, Transportation, and Military Facilities** (Part 1 of 2)	2.4.2	Provide the location and identification of other major nuclear facilities within an 8-kilometer (5-mile) radius. Identify nearby industrial, transportation, and military installations on a map that clearly shows their distance and relationship to the installation.* As appropriate for each, provide a description of products or materials produced, stored, or transported and the maximum quantities for each with detailed emphasis on those items that could present a hazard to the safe operation of the installation. *All activities within 8 kilometers (5 miles) of the site should be considered. Activities at greater distances should be described and evaluated as appropriate to their significance.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.2	**SITE CHARACTERISTICS** **Nearby Industrial, Transportation, and Military Facilities** (Part 2 of 2)	2.4.2	Summarize items that may present a hazard to the installation from nearby activities of the types identified above. The following are typical considerations to be evaluated: 1. The effects of explosion of chemicals, flammable gases, or munitions; 2. The effects of explosions of large natural gas pipelines that cross or pass close to the installation; 3. The effects of detonation of the maximum amount of explosives permitted to be stored at mines or stone quarries near the site; 4. The effects of a. Fires in adjacent oil and gasoline plants or storage facilities, b. Fires in adjacent industries, c. Fires from transportation accidents, and d. Brush and forest fires; 5. The effects of accidental releases of toxic gases from nearby industries and transportation accidents; 6. The effects of expected airborne pollutants on important features of the installation; and 7. The effects of aircraft impacts on the installation, taking into account aircraft size, velocity, weight, and fuel loading for sites in the vicinity of airports. If tall structures such as discharge stacks are used on site, evaluate the potential for damage to equipment or structures important to safety in the event that these structures collapse.
Licensing, Design	2.3	**SITE CHARACTERISTICS** **Meteorology**	2.4.3	This section should provide a meteorological description of the site and its surrounding area. Meteorological conditions that influence the design and operation of the installation should be identified. The bases for all meteorology parameters used as a design basis for any facility structure should be described. Sufficient information should be included to permit an independent evaluation by the NRC staff of atmospheric diffusion characteristics of the local area. The sources of the information and the data supplied should be stated.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.3.1	**SITE CHARACTERISTICS** **Meteorology** Regional Climatology	2.4.3.1	No text provided.
Licensing, Design	2.3.1.1	**SITE CHARACTERISTICS** **Meteorology** Regional Climatology *Data Sources*	2.4.3.1	Provide a discussion of the sources of the data that were used during the climatology analysis. Identify the subjects that are discussed in the various references.
Licensing, Design	2.3.1.2	**SITE CHARACTERISTICS** **Meteorology** Regional Climatology *General Climate*	2.4.3.1	Describe the climate of the region, pointing out characteristics attributable to the terrain. Indicate seasonal weather conditions, including temperature, precipitation, relative humidity, and prevalent wind direction.
Licensing, Design	2.3.1.3	**SITE CHARACTERISTICS** **Meteorology** Regional Climatology *Severe Weather*	2.4.3.1	Provide data on severe weather conditions that may occur within the region and could affect the design or operation of the ISFSI or MRS. If a condition is not considered severe (i.e., fog), it should be included in 2.3.1.2. The frequency, intensity, and duration of the following conditions should be provided: 1. Maximum and minimum temperatures, 2. Extreme winds, 3. Tornadoes, 4. Hurricanes and tropical storms, 5. Precipitation extremes, 6. Thunderstorms and lighting strikes, 7. Snow storms, 8. Hail and ice storms, and 9. Other conditions used in design consideration (i.e., blowing dust, stagnant air).
Licensing, Design	2.3.2	**SITE CHARACTERISTICS** **Meteorology** Local Meteorology	2.4.3.2	No text provided.
Licensing, Design	2.3.2.1	**SITE CHARACTERISTICS** **Meteorology** Local Meteorology *Data Sources*	2.4.3.2	Provide onsite data summaries and nearby weather summaries, identifying the methods and frequencies of collection and pointing out the data collection undertaken specifically for this SAR. Onsite data may not be necessary if data from nearby sources are shown to be adequate for the proposed installation.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.3.2.2	**SITE CHARACTERISTICS** **Meteorology** Local Meteorology *Topography*	2.4.3.2	Provide a map showing the detailed topographic features (as modifed by the facility) on a large scale within an 8-kilometer (5-mile) radius of the site. A smaller-scale map showing topography of the installation and a plot of maximum elevation vs. distance from the center of the installation in each of the sixteen 22h-degree compass point sectors (i.e., centered on true north, north northeast, northeast, etc.) radiating from the facility to a distance of 16 kilometers (10 miles) should also be provided.
Licensing, Design	2.3.3	**SITE CHARACTERISTICS** **Meteorology** Onsite Meteorological Measurement Program	2.4.3.3	Provide joint frequency distributions of wind speed, wind direction, and atmospheric stability, based on appropriate meteorological measurement heights and data-reporting periods. If an onsite meteorological measurement program exists, describe the program being conducted to develop local data and the programs to be used during operations to estimate offsite concentrations of airborne effluents. If an onsite meteorological measurement program does not exist, provide justification for using data from nearby sources. The information provided should include measurements made, locations and elevations of measurements, descriptions of the instruments used, instrument performance specifications, calibration and maintenance procedures, and data analysis procedures. The meteorological measurement program should be consistent with gaseous effluent release structures and systems design. (The effluent release structure and system design is assumed to be commensurate with the degree of risk to the health and safety of the public.)
Licensing, Design	2.3.4	**SITE CHARACTERISTICS** **Meteorology** Diffusion Estimates	2.4.3	No text provided.
Licensing, Design	2.3.4.1	**SITE CHARACTERISTICS** **Meteorology** Diffusion Estimates *Basis*	2.4.3	Provide estimates of atmospheric diffusion at and beyond the controlled area boundary for appropriate time periods for routine releases and after an accident. Consideration of any influence that local topography may have should be included. Beyond the controlled area boundary, show the decrease in relative concentration as a function of distance.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.3.4.2	**SITE CHARACTERISTICS** **Meteorology** Diffusion Estimates *Calculations*	2.4.3	Describe the diffusion equations and the parameters used in the diffusion estimates.
Licensing, Design	2.4	**SITE CHARACTERISTICS** **Surface Hydrology**	2.4.4	Sufficient information should be provided to allow an independent review of all hydrologically related design bases, performance requirements, and operating procedures important to safety. Provide a description characterizing the features relating to the hydrology of the region, area, and site, including additional topographic maps of the site and area as required to provide clarity. Identify the sources of the hydrologic information, the types of data collected, and the methods and frequency of collection.
Licensing, Design	2.4.1	**SITE CHARACTERISTICS** **Surface Hydrology** Hydrologic Description	2.4.4.1	Describe hydrologic features that influence the site or may influence the site or facilities under severe hydrometeorologic or geologic conditions. Include all streams, rivers, lakes, and shore regions adjacent to or running through the site. Identify population groups that use as a potable supply surface water subject to normal or accidental effluents from the plant, and provide the size, use rates, and location of the population groups.
Licensing, Design	2.4.1.1	**SITE CHARACTERISTICS** **Surface Hydrology** Hydrologic Description *Site and Structures*	2.4.4.1	Describe the site and all tant structures important to safety, exterior accesses hereto, and equipment and systems that are important to safety from the standpoint of hydrologic considerations. A topographic map of the site, indicating any proposed changes to natural drainage features, should be provided. Reference the topographic maps provided in Section 2.1.2, and identify the location of the installation and other engineered features.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.1.2	**SITE CHARACTERISTICS** **Surface Hydrology** Hydrologic Description *Hydrosphere*	2.4.4.1	A description should be provided of the location, size, shape, and other hydrologic characteristics of streams, rivers, lakes, shore regions, and ground-water environments influencing the site. Include a description of upstream and downstream river control structures, and explain the criteria governing their operation. Provide a regional topographic map showing the major hydrologic features. List the owner, location, and rate of use of surface-water users whose intakes could be adversely affected by accidental or normal releases of contaminants from the ISFSI or MRS. Refer to Section 2.5.1 for the tabulation of ground-water users.
Licensing, Design	2.4.2	**SITE CHARACTERISTICS** **Surface Hydrology** Floods	2.4.4.2	Provide evidence that the proposed site is a flood-dry site, as defined in ANSI/ANS-2.8-1981,* "Determining Design Basis Flooding at Power Reactor Sites." ANSI/ANS-2.8-1981 defines a flood-dry site as one where structures that are important to safety are so high above potential sources of flooding that safety is obvious or can be documented with minimum analysis. A descriptive statement of circumstances and relative elevations may be sufficient. Analogy may be drawn with comparable watersheds for which probable maximum flood (PMF) levels have been determined. Approximations of PMF levels may be used. Flood studies for dry sites should be carried only to the degree of detail required to prove that structures important to safety are safe from flooding. All methods and assumptions should be conservative. Procedures that can be used are described in ANSI/ANS-2.8-1981. If the proposed site is not clearly floodfree, a detailed analysis should be made in accordance with the procedures outlined in the following sections through Section 2.4.9. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," provides further guidance on specific analytical procedures that are pertinent to this analysis. *Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

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Licensing, Design	2.4.2.1	**SITE CHARACTERISTICS** **Surface Hydrology** Floods *Flood History*	2.4.4.2	Provide a synopsis of the flood** history (date, level, peak discharge, etc.) for the site. Provide frequency, intensity, and cause information for past flooding and other water inundation occurrences, such as tidal or windblown flood waters that may or may not be coincident with one another, with respect to the influence of such occurrences on the site. Include river or stream floods, surges, tsunami, dam failures, ice jams, and similar events. **A "flood" is defined as any abnormally high water stage or overflow from a stream, floodway, lake, or coastal area that results in significantly detrimental effects.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.2.2	**SITE CHARACTERISTICS** **Surface Hydrology** Floods *Flood Design Considerations*	2.4.4.2	Discuss the general capability of the storage structure to be used and of other structures, systems, and equipment that are important to safety to withstand floods, wave action, and flood-induced erosion. The design flood protection level for storage structures and other structures important to safety that are necessary to protect the installation from floods, erosion, and wave action should be based on the highest calculated floodwater-level elevations and flood-wave effects resulting from analysis of several different hypothetical floods. Possible flood conditions, up to and including the highest and most critical flood level, resulting from any of several different probable maximum events should be considered as the basis for the design protection level for storage structures and other structures of the installation that are important to safety. The probable maximum water level from a stream flood, surge, combination of surge and stream flood in estuarial areas, wave action, or tsunami (whichever is applicable and greatest) is that which Way cause the highest water level. Other possibilities are the flood level resulting from the most severe flood wave at the site caused by a landslide, dam failure, dam breaching resulting from a seismic or foundation disturbance, or inadequate design capability. The effects of coincident wind-generated wave action should be superimposed on the applicable flood level. The assumed hypothetical conditions should be evaluated both statically and dynamically to determine the design flood protection level and dynamically induced loadings. The topical information required is generally outlined in Sections 2.4.3 through 2.4.7, but the types of events considered and the controlling event should be summarized in this section.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.2.3	**SITE CHARACTERISTICS** **Surface Hydrology** Floods *Effects of Local Intense Precipitation* (Part 1 of 2)	2.4.4.2	Describe the effects of local probable maximum precipitation (PMP) (see Section 2.4.3.1) on adjacent drainage areas and site drainage systems, including drainage from the roofs of storage structures or other structures that are important to safety. Tabulate rainfall intensities for the selected and critically arranged time increments, provide characteristics and descriptions of runoff models, and estimate the resulting water levels. Summarize the design criteria for site drainage facilities, and provide analyses that demonstrate the capability of site drainage facilities to prevent flooding, due to local PMP, of storage structures or other facilities important to safety. Estimates of precipitation based on publications of the National Oceanic and Atmospheric Administration (NOAA) (formerly U.S. Weather Bureau) of the U.S. Department of Commerce with the time distribution based on critical distributions such as those employed by the Army Corps of Engineers usually provide acceptable bases. Sufficient detail should be provided to (1) allow an independent review of rainfall and runoff effects on storage structures or other facilities that are important to safety and (2) judge the adequacy of design criteria.
Licensing, Design	2.4.2.3	**SITE CHARACTERISTICS** **Surface Hydrology** Floods *Effects of Local Intense Precipitation* (Part 2 of 2)	2.4.4.2	Describe the design bases for snow and ice accumulation on the facilities that are important to safety such as storage structures, other roofs, and exposed equipment. Discuss any effects on the operational capabilities of the storage structures, other structures that are important to safety, and any exposed equipment that is important to safety. In addition, discuss the effect of snow and ice accumulation on site structures where such accumulation could coincide with local probable maximum (winter) precipitation and thus cause flooding or other damage to storage structures or other-structures that are important to safety. Finally, compare the above ice and snow design bases with historical maximum events in the region, and discuss the consequences of exceeding the design bases for storage structures or other structures that are important to safety (including available design margin).

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.3	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers	2.4.4.3	If the site is not clearly a flood-dry site, a detailed flood analysis must be performed. Indicate whether, and if so how, the guidance given in ANSI/ANS-2.8-1981 has been followed; if not followed, describe the specific alternative approaches used. Summarize the locations and associated water levels for which PMF determinations have been made.
Licensing, Design	2.4.3.1	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers *Probable Maximum Precipitation*	2.4.4.3	The PMP is the theoretical precipitation over the applicable drainage area that would produce flood flows that have virtually no risk of being exceeded. These estimates usually involve analyses of actual storms in the general region of the drainage basin under study. They also involve certain modifications and extrapolations of historical data to reflect more severe rainfall-runoff conditions than actually recorded, insofar as those conditions are deemed "reasonably possible" on the basis of hydrometeorological reasoning. Discuss considerations of storm configuration (orientation of areal distribution), maximized precipitation amounts (include a description of maximization procedures and/or studies available for the area such as reference to National Weather Service and Army Corps of Engineers determinations), time distributions, orographic effects, storm centering, seasonal effects, antecedent storm sequences, antecedent snowpack (depth, moisture content, areal distribution), and any snowmelt model. The selected maximized storm precipitation distribution (time and space) should be presented.
Licensing, Design	2.4.3.2	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers *Precipitation Losses*	2.4.4.3	Describe the absorption capability of the drainage basin, including consideration of initial losses, infiltration rates, and antecedent precipitation. Verification of those assumptions should be provided by reference to regional studies or by presenting detailed local storm-runoff studies.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.3.3	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers *Runoff Model*	2.4.4.3	Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), verification from historic floods or synthetic procedures, and the nonlinearity of the model at high rainfall rates. Provide a description of subbasin drainage areas (including a map), their sizes, and topographic features of watersheds. Include a tabulation of all drainage areas, runoff, and reservoir and channel-routing coefficients.
Licensing, Design	2.4.3.4	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers *Probable Maximum Flood Flow*	2.4.4.3	Present the PMF runoff hydrograph (as defined) that results from the PMP (and snowmelt, if pertinent), considering the hydrologic characteristics of the potential influence of existing and proposed upstream dams and river structures for regulating or increasing the water level. If such dams or structures are designed to withstand a PMF, their influence on the regulation of water flow and levels should be considered. However, if they are not designed or constructed to withstand the PMF (or inflow from an upstream dam failure), the maximum water flows and resulting static and dynamic effects from their failure by breaching should be included in the PMF estimate (see Section 2.4.4.2). Discuss the PMF stream course response model and its ability to compute floods of various magnitudes up to the severity of a PMF. Present any reservoir and channel-routing assumptions with appropriate discussions of initial conditions, outlet works (both uncontrolled and controlled), spillways (both uncontrolled and controlled), the ability of any dams to withstand coincident reservoir wind-wave action (including discussions of setup, the significant wave height, the maximum wave height, and runup), the wave protection afforded, and the reservoir design capacity (i.e., the capacity for PMF and coincident wind-wave action). Finally, provide the estimated PMF-discharge hydrograph at the site and, when available, provide a similar hydrograph without upstream reservoir effects to allow evaluation of reservoir effects and a regional comparison of the PMF estimate.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.3.5	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers *Water Level Determinations*	2.4.4.3	Describe the translation of the estimated peak PMF discharge to elevation, using (when applicable) crosssectional and profile data, reconstitution of historical floods (with consideration of high water marks and discharge estimates), standard step methods, roughness coefficients, bridge and other losses, verification, extrapolation of coefficients for the PMF, estimates of PMF water surface profiles, and flood outlines.
Licensing, Design	2.4.3.6	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Flood on Streams and Rivers *Coincident Wind-Wave Activity*	2.4.4.3	Discuss the runup, wave heights, and resultant static and dynamic effects of wave action on each facility that is important to safety from wind-generated activity that may occur coincidently with the peak PMF water level.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.4	**SITE CHARACTERISTICS** **Surface Hydrology** Potential Dam Failures (Seismically Induced)	2.4.4.4	Discuss the evaluation of the effects of potential seismically induced dam failures on the upper limit of flood capability for sites along streams and rivers. Consider the potential influence of upstream dams and river structures on regulating or increasing the water level. The maximum water flow and level resulting from failure of a dam or dams by seismically induced breaching under the most severe probable modes of failure should be taken into account, as well as the potential for subsequent downstream domino-type failures due to floodwaves, where such dams cannot be shown sufficient to withstand severe earthquakes. The simultaneous occurrence of the PMF and an earthquake capable of failing the upstream dams should not be considered since each of these events considered singly has a low probability of occurrence. The suggested worst conditions at the dam site may be evaluated by considering the following: a standard-project flood (as defined by the Army Corps of Engineers) or one-half the PMF, with full reservoirs, coincident with the maximum earthquake determined on the basis of historic seismicity; and a 25-year flood, with full reservoirs, coincident with the maximum earthquake determined on the basis of historic seismicity. Where downstream dams also regulate water supplies, their potential seismically induced failures should be discussed herein. The basis for the earthquake used in this evaluation should be presented.
Licensing, Design	2.4.4.1	**SITE CHARACTERISTICS** **Surface Hydrology** Potential Dam Failures (Seismically Induced) *Reservoir Description*	2.4.4.4	Include a description of the existing locations of or proposed dams (both upstream and downstream) that influence conditions at the site. Tabulate drainage areas above reservoirs, and provide descriptions of types of structures, all appurtenances, ownership, seismic design criteria, and spillway design criteria. Provide the elevation-storage relationships for pertinent reservoirs, and tabulate short- and long-term storage allocations.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.4.2	**SITE CHARACTERISTICS** **Surface Hydrology** Potential Dam Failures (Seismically Induced) *Dam Failure Permutations*	2.4.4.4	Discuss the locations of dams (both upstream and downstream), potential modes of failure, and results of seismically induced and other types of dam failures that could cause the most critical conditions (floods or low water) with respect to the site for such an event (see Section 2.4.3.4). Consideration should be given to possible landslides, antecedent reservoir levels, and river flow coincident with the flood peak (base flow). Present the determination of the peak flow rate at the site for the worst possible dam failure, and summarize an analysis to show that the presented condition is the worst permutation. Include a description of all coefficients and methods used.
Licensing, Design	2.4.4.3	**SITE CHARACTERISTICS** **Surface Hydrology** Potential Dam Failures (Seismically Induced) *Unsteady Flow Analysis of Potential Dam Failures*	2.4.4.4	In determining the effect of dam failures at the site (see Section 2.4.4.2), the analytical methods presented should be applicable to artificially large floods with appropriately acceptable coefficients and should also consider floodwaves through reservoirs downstream of failures. Domino-type failures due to flood- waves should be considered, where applicable. Discuss estimates of coincident flow and other assumptions used to attenuate the dam failure floodwave down- stream. Discuss static and dynamic effects of the attenuated wave at the site.
Licensing, Design	2.4.4.4	**SITE CHARACTERISTICS** **Surface Hydrology** Potential Dam Failures (Seismically Induced) *Water Level at Installation Site*	2.4.4.4	Describe the backwater, unsteady flow, or other computation leaFing to the water elevation estimate (see Section 2.4.4.2) for the most critical upstream dam failure, and discuss its reliability. Superimpose wind-wave conditions that may occur simultaneously in a manner similar to that described in Section 2.4.3.6.
Licensing, Design	2.4.5	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding	2.4.4.5	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.5.1	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Probable Maximum Wind and Associated Meteorological Parameters*	2.4.4.5	This mechanism is defined as a hypothetical hurricane or other cyclonic-type windstorm that might result from the most severe combinations of meteorological parameters that are considered "reasonably possible" in the region involved if the hurricane or other type windstorm should move along a critical path at optimum rate of movement. Present in detail the determination of probable maximum meteorological winds, which involves detailed analyses of actual historical storm events in the general region and certain modifications and extrapolations of data to reflect a more severe meteorological wind system than actually recorded, insofar as these events are deemed reasonably possible on the basis of meteorological reasoning. The probable maximum conditions are the most severe combinations of hydrometeorological parameters considered reasonably possible that would produce a surge or seiche that has virtually no risk of being exceeded (e.g., the meteorological characteristics of the probable maximum hurricane as reported by NOAA in their technical report NWS-23* for the East and Gulf Coasts, the most severe combination of meteorological parameters of moving squall lines for the Great Lakes, or the- most severe combination of meteorological parameters capable of producing high storm-induced tides for the West Coast). This hypothetical event is postulated along a critical path at an optimal rate of movement from correlations of storm parameters of record. Sufficient bases and information should be provided to ensure that the parameters presented are the most severe combination. *NOAA Technical Report NWS-23, "Meteorological Criteria for the Standard Project Hurricane and Probable Maximum Hurricane Windfields, Gulf and East Coasts of the United States," is available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.5.2	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Surge and Seiche History*	2.4.4.5	Discuss the proximity of the site to large bodies of water for which surge- or seiche-type flooding can reach the storage structures or other structures that are important to safety. The probable maximum water level (surges) for shore areas adjacent to large water bodies is the peak of the hypothetical surge or seiche-stage hydrograph (stillwater levels) and coincident wave effects. It should be based on relatively comprehensive hydrometeorological analyses and the application of probable maximum meteorological criteria (such as hurricanes, moving squall lines, or other cyclonic wind storms), in conjunction with the critical hydrologic characteristics, to estimate the probable maximum water level at a specific location. The effects of the probable maximum meteorological event should be superimposed on the coincident maximum annual astronomical and ambient tide levels, and associated wave action, to determine the effects of water level and wave action on structures. Provide a description of the surge and seiche history in the site region.
Licensing, Design	2.4.5.3	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Surge and Seiche Sources*	2.4.4.5	Discuss considerations of hurricanes, frontal-type (cyclonic) wind storms, moving squall lines, and surge mechanisms that are possible and applicable to the site. Include (1) the antecedent water level (with reference to the spring tide for coastal locations, the average monthly recorded high water for lakes, and a forerunner or ambient water level where applicable), (2) the determination of the controlling storm surge or seiche (consider the probable maximum meteorological parameters such as the storm track, wind fields, the fetch or direction of approach, bottom effects, and verification with historic events), (3) the method used, and (4) the results of the computation of the probable maximum surge hydrograph (graphical presentation).
Licensing, Design	2.4.5.4	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Wave Action*	2.4.4.5	Discuss the wind-generated activity that can occur coincidently with a surge or seiche, or independently thereof. Estimates of the wave period, the significant wave height and elevations, and the maximum wave height and elevations, with the coincident water level hydrograph, should be presented. Give specific data on the largest breaking wave height, setup, and runup that can reach each storage structure or other facility that is important to safety.

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Licensing, Design	2.4.5.5	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Resonance*	2.4.4.5	Discuss the possibility of oscillations of waves at natural periodicity, such as lake reflection and harbor resonance phenomena, and any resulting effects at the site.
Licensing, Design	2.4.5.6	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Runup*	2.4.4.5	Provide estimates of wave runup on Include the site structures. Include a discussion of the water levels on each affected structure and the protection to be provided against static effects, dynamic effects, and splash. Refer to Section 2.4.5.4 for breaking waves.
Licensing, Design	2.4.5.7	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Surge and Seiche Flooding *Protective Structures*	2.4.4.5	Discuss the location and design criteria for any special water-control structures for the protection of the storage structures or other structures that are important to safety against surges, seiches, wave reflection, and other wave action.
Licensing, Design	2.4.6	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding	2.4.4.6	For sites adjacent to coastal areas, discuss historical tsunami (either recorded or translated and inferred) that provide information for use in determining the probable maximum water levels and the geoseismic-generating mechanisms available, with appropriate references to Section 2.6.
Licensing, Design	2.4.6.1	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding *Probable Maximum Tsunami*	2.4.4.6	This event is defined as the most severe tsunami at the site that has virtually no risk of being exceeded. Consideration should be given to the most reasonably severe geoseismic activity possible in determining the limiting tsunami-producing mechanism (e.g., fractures, faults, landslide potential, and volcanism). Such considerations as the orientation of the site relative to the earthquake epicenter or generating mechanism, shape of the coastline, offshore land areas, hydrography, and stability of the coastal area should be presented in the analysis.

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Licensing, Design	2.4.6.2	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding *Historical Tsunami Record*	2.4.4.6	Provide local and regional historical tsunami information.
Licensing, Design	2.4.6.3	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding *Source Tsunami Wave Height*	2.4.4.6	Provide estimates of the maximum tsunami wave height possible at each major local generating source considered and the maximum offshore deep-water tsunami height from distant generators. Discuss the controlling generators for both locally and distantly generated tsunami.
Licensing, Design	2.4.6.4	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding *Tsunami Height Offshore*	2.4.4.6	For each major generator, provide estimates of the tsunami height in deep water adjacent to the site or before bottom effects appreciably alter wave configuration.
Licensing, Design	2.4.6.5	**SITE CHARACTERISTICS** **Surface Hydrology** Probable Maximum Tsunami Flooding *Hydrography and Harbor or Breakwater Influences on Tsunami*	2.4.4.6	Present the routing of the controlling tsunami. Include breaking wave formation, bore formation, and any resonance effects (natural frequencies and successive wave effects) that result in the estimate of the maximum tsunami runup on each storage structure or other structure that is important to safety. Also include a discussion of the analysis used to translate tsunami waves from offshore generator locations, or in deep water, to the site and a discussion of antecedent conditions. Provide, where possible, verification of the techniques and coefficients used by reconstituting tsunami of record.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.4.7	**SITE CHARACTERISTICS** **Surface Hydrology** Ice Flooding	2.4.4.7	Present design criteria for the protection of storage structures or other facilities that are important to safety from the most severe ice jam floods, wind-driven ice ridges, or ice-produced forces that are reasonably possible and could affect storage structures or other structures that are important to safety with respect to adjacent rivers, streams, or lakes. Include the location and proximity of such facilities to ice-generating mechanisms. Describe the regional ice and ice jam formation history.
Licensing, Design	2.4.8	**SITE CHARACTERISTICS** **Surface Hydrology** Flooding Protection Requirements	2.4.4.8	Describe the static and dynamic consequences of all types of flooding on each storage structure and component that is important to safety. Present the design bases required to ensure that the storage structures and components that are important to safety will be capable of surviving all design flood conditions. Reference appropriate discussions in other sections of the SAR where these design bases are implemented.
Licensing, Design	2.4.9	**SITE CHARACTERISTICS** **Surface Hydrology** Environmental Acceptance of Effluents	2.4.4.9	Describe the ability of the surface-water and ground-water environment to disperse, dilute, or concentrate normal and inadvertent or accidental liquid releases of radioactive effluents for the full range of anticipated operating conditions as such releases may relate to existing or potential future use of surface-water or ground-water resources. Describe any effects of normal or accidental releases of radionuclides on surface waters and ground waters, e.g., any potential for recirculation, sediment concentration, and hydraulic short-circuiting of cooling ponds, if applicable.
Licensing, Design	2.5	**SITE CHARACTERISTICS** **Subsurface Hydrology**	2.4.5	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.5.1	**SITE CHARACTERISTICS** **Subsurface Hydrology** Regional Characteristics	2.4.5	If local ground water is a major water resource, the ground-water system may be of importance beyond an ISFSI or MRS site. If so, describe the principal ground-water aquifers and associated hydrogeologic units and their recharge and discharge points in relationship to the site location. For each hydrogeologic unit identified, discuss the flow directions, hydraulic gradients, potential for reversibility of ground-water flow, and potential effects of future use on ground-water recharge areas within the influence of the installation. Provide a survey of ground-water users, including location, uses, static water levels, pumping rates, drawdown, and source aquifers.
Licensing, Design	2.5.2	**SITE CHARACTERISTICS** **Subsurface Hydrology** Site Characteristics	2.4.5	Provide data on ground-water potentiometric levels, hydraulic characteristics, including hydraulic conductivity, effective porosity, and storage coefficient, and hydraulic gradients at the site. The proposed ground-water sources and usage anticipated by the installation should also be given. Provide a water table contour map showing surface water bodies, recharge and discharge points, and the location of any monitoring wells used to evaluate possible leakage from storage structures. If monitoring wells are used, provide information on the elevations and the top of casings, the screened interval, and methods of installation. Identify any potential ground-water recharge areas within the influence of the installation, and discuss the effects of construction, including dewatering, on such areas. Provide information on the hydrochemistry of the water table to include presence major ions, pH-Eh values, and of radionuclides.
Licensing, Design	2.5.3	**SITE CHARACTERISTICS** **Subsurface Hydrology** Contaminant Transport Analysis	2.4.5	By use of the information collected to describe the regional and site characteristics, provide an analysis that indicates the bounds of potential contamination from the site operations to the ground-water system. Include in the analysis a graph of time versus concentration of the radionuclide migration at the location of the nearest existing or potential future user.
Licensing, Design	2.6	**SITE CHARACTERISTICS** **Geology and Seismology**	2.4.6	The geologic and seismic characteristics of the area and site, the nature of investigations performed, results of investigations, conclusions, and identi- fication of information sources should be provided. Supplement the written description with tables and legible graphics, as appropriate.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.1	**SITE CHARACTERISTICS** **Geology and Seismology** Basic Geologic and Seismic Information (Part 1 of 4)	2.4.6.1	The basic geologic and seismic information for the site should be presented. Information obtained from published reports, professional papers, dissertations, maps, private communications, or other sources should be referenced. Data from surveys, geophysical investigations, borings, trenches, or other investigations should be adequately documented by descriptions of techniques, graphic logs, photographs, laboratory results, identification of principal investigators, and other data. Areas of potential seismic or volcanic activity or unstable geologic Characteristics should be avoided, if possible, for the siting of an ISFSI or MRS. The methods used to determine that the site meets the design criteria of Part 72 should be presented. Material in this section may be included, as appropriate, in Section 2.6.3 and cross-referenced in this section. 1. Describe the site geomorphology. A site topographic map showing the locations of the principal facilities should be included. Describe the configuration of the land forms, and relate the history of geologic chdnges that have occurred. Areas in the site of actual or potential landsliding, surface or subsurface subsidence, uplift, or collapse resulting from natural features (such as tectonic depressions and cavernous or karst terrains) and from man's activities (such as withdrawal or addition of subsurface fluids or mineral extraction) should be evaluated.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.1	**SITE CHARACTERISTICS** **Geology and Seismology** Basic Geologic and Seismic Information (Part 2 of 4)	2.4.6.1	2. Discuss the geologic history of the site and the surrounding region. Describe the lithologic, stratigraphic, and structural geologic conditions of the site and of the surrounding region. A stratigraphic column should be included. Describe the thicknesses, physical characteristics, mineral composition, origin, and degree of consolidation of each lithologic unit. Furnish summary logs of borings and excavations such as trenches used in the geologic evaluation. 3. Identify specific structural features of significance to the site, e.g., folds, faults, joints, synclines, anticlines, domes, and basins. Provide a large-scale structural geologic map of the site showing bedrock surface contours (surface contour maps) and the location of structures. 4. Furnish a large-scale geologic map of the site area that shows surface geology and includes the locations of major structures of the installation. Areas of direct observations of bedrock outcrop should be distinguished from areas that are covered and about which geologic interpretation has been extrapolated (i.e., outcrop map). When the interpretation differs substantially from the published geologic literature on the area, the differences should be noted and documentation for the differing conclusions presented. 5. Furnish a plot plan showing the locations of major structures of the installation and the locations of all borings, trenches, and excavations. Also include a description, logs, and maps of the borings, trenches, and excavations, as necessary, to indicate the results. 6. Provide geologic profiles that show the relationship of major foundations to subsurface materials, including ground water. Describe the significant engineering characteristics of the subsurface materials. 7. Provide plan and profile drawings showing the extent of excavations and backfill planned at the site. Describe compaction criteria for all engineered backfill.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.1	**SITE CHARACTERISTICS** **Geology and Seismology** Basic Geologic and Seismic Information (Part 3 of 4)	2.4.6.1	8. Include an evaluation from an engineering-geology standpoint of the local geologic features that could affect ISFSI or MRS structures. a. Describe available physical evidence concerning the behavior during previous earthquakes of the surface geologic materials and the substrata underlying the site. This determination may require lithologic, stratigraphic, and structural geologic studies. b. Identify and evaluate deformation zones, such as shears, joints, fractures, faults, and folds, or combinations of these features, relative to structural foundations. c. Describe and evaluate zones of alteration or irregular weathering profiles and zones of structural weakness composed of crushed or disturbed materials. d. Describe all rocks or soils that might be unstable because of their mineral composition, lack of consolidation, water content, or potentially undesirable response to seismic or other events. Seismic response characteristics to be considered include liquefaction, thixotropy, differential consolidation, cratering, and fissuring. 9. Define site ground-water conditions and their relationship to regional ground-water conditions. Include the properties of aquifer materials and any fine-grained materials associated with the uppermost unconfined or semiconfined aquifer. 10. Provide profiles, maps, and tables showing the results of any geophysical surveys (e.g.; seismic refraction, seismic reflection, acoustic, and aeromagnetic) conducted to evaluate the stratigraphic structure and bedrock and showing subsurface material characteristics of the site. Results of compressional and shear wave velocity surveys and crosshole and uphole velocity surveys, where performed, should be provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.1	**SITE CHARACTERISTICS** **Geology and Seismology** Basic Geologic and Seismic Information (Part 4 of 4)	2.4.6.1	11. Furnish in detail static and dynamic engineering soil and rock properties of the materials underlying the site, including grain-size classification, Atterberg limits, water content, unit weight, shear strength, relative density, shear modulus, Poisson's ratio, bulk modulus, damping, consolidation characteristics, seismic wave velocities, density, porosity, strength characteristics, and strength under cyclic loading. These data should be substantiated with appropriate representative laboratory test records. The results should be interpreted and integrated to provide a comprehensive understanding of the surface and subsurface conditions. 12. Discuss the analysis techniques used and the factors of safety for foundation materials for evaluating the stability of foundations for all structures and for all embankments under normal operating and extreme environmental conditions.
Licensing, Design	2.6.2	**SITE CHARACTERISTICS** **Geology and Seismology** Vibratory Ground Motion	2.4.6.2	Information should be presented to describe how the data were selected for determining the design basis for vibratory ground motion. The following specific information and determinations should also be included to the extent necessary to clearly establish the design basis for vibratory ground motion. Information presented in other sections may be cross-referenced and need not be repeated.
Licensing, Design	2.6.2.1	**SITE CHARACTERISTICS** **Geology and Seismology** Vibratory Ground Motion *Engineering Properties of Materials for Seismic Wave Propagation and Soil-Structure Interaction Analyses*	2.4.6.2	Describe the static and dynamic engineering properties of the materials underlying the site. Included should be properties needed to determine the behavior of the underlying material during earthquakes and the characteristics of the underlying material in transmitting earthquake-induced motions to the foundations of the structures, e.g., seismic wave velocities, density, water content, porosity, and strength.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.2.2	**SITE CHARACTERISTICS** **Geology and Seismology** Vibratory Ground Motion *Engineering Properties of Materials for Seismic Wave Propagation and Soil-Structure Interaction Analyses*	2.4.6.2	List all historically reported earthquakes that have affected or could be reasonably expected to have affected the site. The listing should include the date of occurrence, the magnitude or highest intensity, and a plot of the epicenter or region of highest intensity. Include all historically reported earthquakes that could have caused a maximum ground acceleration of at least one-tenth the acceleration of gravity (0.1g) at ground surface in the free field. Since earthquakes have been reported in terms of various parameters such as magnitude, intensity at a given location, and effect on ground, structures, and people at a specific location, some of these data may have to be estimated by use of appropriate empirical relationships. Where appropriate, the comparative characteristics of (1) the material underlying the epicentral location or region of highest intensity and (2) the material underlying the site in transmitting earthquake vibratory motion should be considered.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.2.3	**SITE CHARACTERISTICS** **Geology and Seismology** Vibratory Ground Motion *Procedures to Determine the Destan Earthquake* (Part 1 of 2)	2.4.6.2	The design earthquake for the ISFSI or MRS structures that are important to safety should be defined by response spectra corresponding to the maximum horizontal ground motion accelerations. Refer to § 72.102, "Geological and Seismological Characteristics," for licensing requirements. An ISFSI or MRS located in an area of low potential seismic activity or surface offset potential as defined in § 72.102 should be designed for a standardized 0.25g or alternatively for a site-specific g value no less than 0.10g and response spectra as determined by the following procedure: 1. Identification of Capable Faults. For faults, any part of which are within 161 kilometers (100 miles) of the site and which may be of significance in establishing the design criteria for earthquake protection, determine whether these faults should be considered as capable faults.* 2. Description of Capable Faults. For faults, any part of which are within 161 kilometers (100 miles) of the site and which may be of significance in establishing the earthquake criteria and may be considered as capable faults, the following should be determined: the length of the fault; the relationship of the fault to regional tectonic structures; and the nature, amount, and geologic history of the maximum Quaternary displacement related to any one earthquake along the fault. 3. Maximum Earthquake. Determine the historic earthquakes of greatest magnitude or intensity that have been correlated with tectonic structures. For capable faults, the earthquake of greatest magnitude related to the faults should be determined, taking into account geologic evidence. The vibratory ground motion at the site should be determined, assuming the epicenters of the earthquakes are situated on the structures closest to the site. *Capable faults are defined in Appendix A to 10 CFR Part 100, "Reactor Site Criteria."

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Licensing, Design	2.6.2.3	**SITE CHARACTERISTICS** **Geology and Seismology** Vibratory Ground Motion *Procedures to Determine the Destan Earthquake* (Part 2 of 2)	2.4.6.2	Where epicenters or regions of highest intensity of historically reported earthquakes cannot be related to tectonic structures but are identified with tectonic provinces in which the site is located, determine the accelerations at the site assuming that these earthquakes occur adjacent to the site. If the epicenters or regions of highest intensity of historically reported earthquakes are identified with adjacent or nearby tectonic provinces, determine the accelerations at the site assuming that the epicenters or regionsof highest intensity of these earthquakes are located at the closest point to the site on the boundary of the tectonic province.
Licensing, Design	2.6.3	**SITE CHARACTERISTICS** **Geology and Seismology** Surface Faulting	2.4.6.3	Information that describes surface faulting at the site should be presented. The following specific information and determinations should also be included. Information presented in Section 2.6.1 may be cross-referenced and need not be repeated.
Licensing, Design	2.6.3.1	**SITE CHARACTERISTICS** **Geology and Seismology** Surface Faulting *Evidence of Fault Offset.*	2.4.6.3	Determine the fault geologic evidence of offset at or near the ground surface at or near the site.
Licensing, Design	2.6.3.2	**SITE CHARACTERISTICS** **Geology and Seismology** Surface Faulting *Identification of Capable Faults*	2.4.6.3	For faults greater than 300 meters (1,000 feet) any part of which are within 8 kilometers (5 miles) of the site, determine whether these faults should be considered as capable faults.

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Licensing, Design	2.6.4	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials	2.4.6.4	Information should be presented concerning the stability of rock (defined as having a shear wave velocity of 1,166 m/sec (3,500 ft/sec)) and soil underneath the structure foundations during the vibratory motion associated with earthquake design criteria. Evaluate the following geologic features that could affect the foundations. Information presented in other sections may be cross-referenced and need not be repeated.
Licensing, Design	2.6.4.1	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Geologic Features*	2.4.6.4	Describe the following geologic features: 1. Areas of actual or potential surface or subsurface subsidence, uplift, or collapse resulting from; a. Natural features such as tectonic depressions and cavernous or karst terrains, particularly those underlaid by calcareous or other soluble deposits, b. Man's activities such as withdrawal or addition of subsurface fluids or mineral extraction, or c. Regional warping. 2. Deformation zones such as shears, joints, fractures, faults, and folds or combinations of these features; 3. Zones of alteration or irregular weathering profiles and zones of structural weakness composed of crushed or disturbed materials; 4. Stresses in bedrock; and 5. Rocks or soils that might be unstable because of their mineral com- position, lack of consolidation, water content, or potentially undesirable response to seismic or other events. Seismic response characteristics to be considered include liquefaction, differential consolidation, cratering, and fissuring.
Licensing, Design	2.6.4.2	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Properties of Underlying Materials*	2.4.6.4	Describe in detail the static and dynamic engineering properties of themtafer-ials underlying the site. Furnish the physical properties of foundation materials such as grain-size classification, consolidation characteristics, water content, Atterberg limits, unit weight, shear strength, relative density, shear modulus, damping, Poisson's ratio, bulk modulus, strength under cyclic loading, seismic wave velocities, density, porosity, and strength characteristics. These data should be substantiated with appropriate representative laboratory test records.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	2.6.4.3	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Plot Plan*	2.4.6.4	Provide a plot plan (or plans) showing the locations of all borings, trenches, seismic lines, piezometers, geologic profiles, and excavations, and superimpose the locations of all installation structures. Furnish profiles showing the relationship of the foundations of structures to subsurface materials, including ground water and significant engineering characteristics of the subsurface materials.
Licensing, Design	2.6.4.4	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Soil and Rock Characteristics*	2.4.6.4	Provide the results by means of tables and profiles of compressional and shear wave velocity surveys performed to evaluate the characteristics of the foundation soils and rocks. Provide graphic core boring logs and the logs of trenches or other excavations.
Licensing, Design	2.6.4.5	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Excavations and Backfill*	2.4.6.4	Furnish plan and profile drawings showing the extent of excavations and backfill planned at the site and compaction criteria for all engineered backfill. The criteria should be substantiated with representative laboratory or field test records. Where possible, these plans and profiles may be combined with profiles in Sections 2.6.4.3 or 2.6.4.4.
Licensing, Design	2.6.4.6	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Ground-Water Conditions*	2.4.6.4	Provide a history of ground-water fluctuations beneath the site and a discussion of anticipated ground-water conditions during construction of the installation and during its expected life.

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Licensing, Design	2.6.4.7	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Response of Soil and Rock to Dynamic Loading*	2.4.6.4	Furnish analyses of the response of soil and rock to dynamic loading.
Licensing, Design	2.6.4.8	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Liquefaction Potential*	2.4.6.4	Provide a discussion of the liquefaction potential of material beneath the site. Either demonstrate that there are no liquefaction-susceptible soils beneath the site, or provide the following information regarding soil zones where the possibility for liquefaction exists: relative density, void ratio, ratio of shear stress to initial effective stress, number of load cycles, grain-size distribution, degrees of cementation and cohesion, and ground-water elevation fluctuations.
Licensing, Design	2.6.4.9	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Liquefaction Potential*	2.4.6.4	The analysis for soil stability should be based on the design earthquake and response spectra used.
Licensing, Design	2.6.4.10	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Static Analyses*	2.4.6.4	Discuss the static analyses, such as settlement analyses (with appropriate representative laboratory data), and lateral pressures (with backup data).

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Licensing, Design	2.6.4.11	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Techniques To Improve Subsurface Conditions*	2.4.6.4	Discuss and provide specifications for required techniques to improve subsurface conditions such as grouting, vibraflotation, rock bolting, and anchors.
Licensing, Design	2.6.4.12	**SITE CHARACTERISTICS** **Geology and Seismology** Stability of Subsurface Materials *Criteria and Design Methods*	2.4.6.4	List and furnish a brief discussion of the criteria, references, or methods of design employed (or to be employed) and factors of safety (documented by test data).
Licensing, Design	2.6.5	**SITE CHARACTERISTICS** **Geology and Seismology** Slope Stability	2.4.6.5	Information and appropriate substantiation should be presented concerning the stability of all slopes, both natural and man-made (both cut and fill), the failure of which could adversely affect the site.
Licensing, Design	2.6.5.1	**SITE CHARACTERISTICS** **Geology and Seismology** Slope Stability *Slope Characteristics*	2.4.6.5	Cross sections of the slopes should be provided along with a summary of the static and dynamic properties of embankment and foundation soil and rock underlying the slope. Substantiate with representative laboratory test data.
Licensing, Design	2.6.5.2	**SITE CHARACTERISTICS** **Geology and Seismology** Slope Stability *Design Criteria and Analyses*	2.4.6.5	The design criteria and analyses used to determine slope stability should be described. Include factors important to safety, along with the adverse conditions considered in the analyses, such as sudden drawdown and earthquake.

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Licensing, Design	2.6.5.3	**SITE CHARACTERISTICS** **Geology and Seismology** Slope Stability *Logs of Core Borings*	2.4.6.5	Furnish logs of core borings to test pits taken in proposed borrow areas.
Licensing, Design	2.6.5.4	**SITE CHARACTERISTICS** **Geology and Seismology** Slope Stability *Compaction Specifications*	2.4.6.5	Provide compaction specifications along with representative lab data on-which they are based.
Licensing, Design	2.7	**SITE CHARACTERISTICS** **Summary of Site Conditions Affecting Construction and Operating Requirements**	None	Summarize all factors developed in this chapter that are deemed significant to the selection of design bases for the installation.
Licensing, Design	3	**PRINCIPAL DESIGN CRITERIA**	4.4.3.1	Principal design criteria are established by the applicant in the SAR. The NRC staff analyzes these design criteria for adequacy before the application is approved. Changes in the criteria are not anticipated after that approval is granted. Therefore, the criteria selected should encompass all considerations for alternatives that the applicant may choose.
Licensing, Design	3.1	**PRINCIPAL DESIGN CRITERIA** **Purposes of Installation**	4.4.3.1	Describe in general terms the mode of storage; the installation; its functions, operation, and storage capacity; and the types of fuel or high-level radioactive waste to be stored.

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Licensing, Design	3.1.1	**PRINCIPAL DESIGN CRITERIA** **Purposes of Installation** Materials To Be Stored	4.4.1	A detailed description of the physical, thermal, and radiological characteristics of the spent fuels or high-level radioactive wastes to be stored should be provided. Include spent fuel characteristics such as specific power, burnup, decay time, and heat generation rates and high-level radioactive waste characteristics such as curie content, waste form, and generation rates.
Licensing, Operation, Design	3.1.2	**PRINCIPAL DESIGN CRITERIA** **Purposes of Installation** General Operating Functions	3.4.1	Provide information related to the overall functioning of the installation as a storage operation. Information should be included on onsite-generated waste processing, waste packaging and storage areas, transportation, and utilities.
Licensing, Design	3.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria**	4.4.3.2	Based on the site selected, identify and quantify the environmental and geologic features that are used as design criteria for identified structures, systems, and components that are important to safety.
Licensing, Design	3.2.1	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Tornado and Wind Loadings	4.4.3.2	No text provided.

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Licensing, Design	3.2.1.1	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Tornado and Wind Loadings *Applicable Design Parameters*	4.4.3.2	The design parameters applicable to the design tornado such as translational velocity, rotational velocity, and the design pressure differential and its associated time interval should be specified.
Licensing, Design	3.2.1.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Tornado and Wind Loadings *Determination of Forces on Structures*	4.4.3.2	Describe the methods used to convert the tornado and wind loadings into forces on the structures, including the distribution across the structures and the combination of applied loads. If factored loads are used, the basis for selection of the load factor used for tornado loading should be furnished.
Licensing, Design	3.2.1.3	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Tornado and Wind Loadings *Ability of Structures To Perform Despite Not Failure of Structures Designed for Tornado Loads*	4.4.3.2	Information to show that the failure of any structure not being designed for tornado loads will not affect the ability of other structures or systems important to safety to perform their intended design functions should be presented.

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Licensing, Design	3.2.1.4	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Tornado and Wind Loadings *Tornado Missiles*	4.4.3.2	The dimensions, mass, energy, velocity, and .other parameters should be selected for a potential tornado-driven missile.* An analysis should be presented to show the potential effect of such a missile on structures, systems, and components important to safety. *Section 3.5.1.4 (paragraph 4 of Section III) of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, contains information that may be helpful in developing these data. A copy of Section 3.5.1.4 is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW., Washington, DC.
Licensing, Design	3.2.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Water Level (Flood) Design	4.4.3.2	If the facility is not to be located on a flood-dry site, discuss the design loads from forces developed by the PMF, including water height and dynamic phenomena such as velocity. By reference, relate the design criteria to data developed in Section 2.4.
Licensing, Design	3.2.2.1	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Water Level (Flood) Design *Flood Elevations*	4.4.3.2	The flood elevations that will be used in the design of each structure for buoyancy and static water force effects should be provided.

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Licensing, Design	3.2.2.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Water Level (Flood) Design *Phenomena Considered in Design Load Calculations*	4.4.3.2	The phenomena (e.g., flood current, wind wave, hurricane, or tsunami) that are being considered if dynamic water force is a design load for any structure should be identified and discussed.
Licensing, Design	3.2.2.3	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Water Level (Flood) Design *Flood Force Application*	4.4.3.2	Describe the manner in which the forces and other effects resulting from flood loadings are applied.
Licensing, Design	3.2.2.4	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Water Level (Flood) Design *Flood Protection*	4.4.3.2	Describe the flood protection measures for storage structures and other systems located below grade or below flood level that are important to safety.

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Licensing, Design	3.2.3	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design	4.4.3.2	From data developed in Chapter 2, "Site Characteristics," present the design criteria to be used in construction of the installation and its associated equipment. Sufficient detail should be presented to allow an independent evaluation of the criteria selected. For clarity, cross-reference appropriate information presented in Section 2.6.
Licensing, Design	3.2.3.1	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design *Input Criteria* (Part 1 of 3)	4.4.3.2	This section should discuss the input criteria for seismic design of the installation, including the following specific information: 1. Design Response Spectra. Design response spectra should be provided for the design earthquake. A discussion of effects of the following parameters should also be included: a. Earthquake duration, b. Earthquake distance and depths between the seismic disturbances and the site, and c. Existing earthquake records and the associated amplification response range where the amplification factor is greater than one. 2. Design Response Spectra Derivation. If response spectral shapes other than those ign Response Spectra for Seismic Design of Nuclear Power Plants," are proposed for design of the storage structures or other structures that are important to safety or for the determination of liquefaction potential, these should be justified and the earthquake time functions or other data from which these were derived should be presented. For all the damping values that are used in the design, submit a comparison of the response spectra derived from the time history and the design response spectra. The system period intervals at which the spectra values were calculated should be identified. The response spectra applied at the finished grade in the free field or at the various foundation locations of structures that are important to safety should be provided.

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Licensing, Design	3.2.3.1	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design *Input Criteria* (Part 2 of 3)	4.4.3.2	3. Desion Time History. For any time-history analyses, the response spectra derived from the actual or synthetic earthquake time-motion records should be provided. A comparison of the response spectra obtained in the free field at the finished grade level and the foundation level (obtained from an appropriate time history at the base of the soil-structure interaction system) with the design response spectra should be submitted for each of the damping values to be used in the design of structures, systems, and components. Alternatively, if the design response spectra are applied at the foundation levels of the storage structures or other structures that are important to safety in the free field, a comparison of the free-field response spectra at the foundation level (derived from an actual or synthetic time history) with the design response spectra should be provided for each of the damping values to be used in the design. The period intervals at which the spectra values were calculated should be identified. 4. Use of Equivalent Static Loads. The basis for load factors used on the seismic design of storage structures or other structures, systems, and components that are important to safety in lieu of the use of a seismic-system multimass dynamic analysis method should be identified. For example, dynamic soil pressures can be adequately estimated by using modifications to the Mononobe-Okabe theory. 5. Critical Damping Values. The specific percentage of critical damping values used for identified structures, systems, components, and soil should be provided. For example, damping values for the type of construction or fabrication and the applicable allowable design stress levels for these installation features should be submitted.

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Licensing, Design	3.2.3.1	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design *Input Criteria* (Part 3 of 3)	4.4.3.2	6. Bases for Site-Dependent Analysis. The bases for a site-dependent analysis, if used to develop the shape of the design response spectra from bedrock time history or response spectra input, should be provided. Specifically, the bases for use of in situ soil measurements, soil layer location, and bedrock earthquake records should be provided. If the analytical approach used to determine the shape of the design response spectra neglects vertical amplification and possible slanted soil layers, these assumptions as well asthe influence of the effect of possible predominant thin soil layers on the analytical results should be discussed. 7. Soil-Supported Structures. A list of all soil-supported storage structures or other structures that are important to safety should be provided. This list should include the depth of soil over bedrock for each structure listed. 8. Soil-Structure Interaction. For nonbedrock sites, soil-structure interaction is to be treated inth e same manner as for the Safe Shutdown Earthquake (SSE) at nuclear power plants. Describe any soil-structure interaction techniques used in the analyses of the structures. Nonlinear, or equivalent linear, finite element techniques should be used as the analytical tools for soil-structure interaction analyses for all structures where the foundations are deeply embedded in soil. For shallowly embedded structures on deep, uniform soil strata, the soil spring model based on the elastic half-space theory is adequate. For shallowly embedded structures with shallow soil overburden over rock or layered soil with varying soil properties, the finite element approach or multiple shear beam spring approach should be used.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.2.3.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design *Seismic-System Analyses* (Part 1 of 3)	4.4.3.2	This section should discuss the seismic-system analyses applicable to structures, systems, and components that are important to safety. The following specific information should be included: 1. Seismic Analysis Methods. For all storage structures or other structures, systems, and components identified in Section 3.2 that are important to safety, the applicable methods of seismic analysis (e.g., modal analysis response spectra, modal analysis time history, equivalent static load) should be identified in the SAR. Applicable stress or deformation criteria and descriptions (sketches) of typical mathematical models used to determine the response should be specified. All seismic methods of analyses used should also be described in the SAR. 2. Natural Frequencies and Response Loads. A summary of natural frequencies and response loads (e.g., in the form of critical mode shapes and modal responses) determined by the seismic-system analysis should be provided. The ISFSI and the MRS design earthquake is considered a faulted condition as is the SSE for nuclear power plants. Dynamic or equivalent static loads are to be treated as outlined in Regulatory Guide 3.60, "Design of an Independent Spent Fuel Storage Installation (Dry Storage)." 3. Procedure Used to Lump Masses. Provide a description of the procedure used to lump masses for the seismic-system analyses (ratio of system mass and compliance to component mass and compliance and the ratio of floor mass and compliance to supported equipment mass and compliance).

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.2.3.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design *Seismic-System Analyses* (Part 2 of 3)	4.4.3.2	4. Rocking and Translational Response Summary. If a fixed base in the mathematical models for the dynamic system analyses is assumed, a summary of the rocking and translational responses should be provided. A brief description should be included of the method, mathematical model, and-damping values (rocking, vertical, translation, and torsion) that have been used to consider the soil-structure interaction. 5. Methods Used to Couple Soil with Seismic-System Structures. Describe the methods and procedures used to couple the soil and the seismic-system structures and components in the event that a finite element analysis for the layered site is used. 6. Method Used to Account for Torsional Effects. The method used to consider the torsional modes of vibration in the seismic analysis of the structures should be described. The use of static factors to account for torsional accelerations in the seismic design structures or, in lieu of the use of a combined vertical, horizontal, and torsional multimass system, dynamic analysis should be indicated. 7. Methods for Seismic Analysis of Dams. A description of the analytical methods and procedures used for the seismic-system analysis of dams that impound bodies of water important to safety should be provided. 8. Methods to Determine Overturning Moments. A description of the dynamic methods and procedures used to determine structure overturning moments, including a description of the procedures used to account for soil reactions and vertical earthquake effects, should be provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.2.3.2	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Seismic Design *Seismic-System Analyses* (Part 3 of 3)	4.4.3.2	9. Analysis Procedure for Damping. The analysis procedure followed to account for the damping in different elements of a coupled system model, including the criteria used to account for composite damping in a coupled system with different elements, should be described. 10. Seismic Analysis of Overhead Cranes. The provisions taken to ensure that all overhead cranes and transfer machines that are important to safety will not be dislodged from their rails in the event of the design earthquake should be described. 11. Seismic Analysis of Specific Safety Features. The integrity of specific design features (e.g., sealed surface storage casks [SSSCs] containing spent fuel or high-level radioactive waste) in the event of an earthquake should be provided.
Licensing, Design	3.2.4	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Snow and Ice Loadings	4.4.3.2	Describe design and operating load criteria used to ensure that maximum snow and ice loads can be accommodated.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.2.5	**PRINCIPAL DESIGN CRITERIA** **Structural and Mechanical Safety Criteria** Combined Load Criteria	4.4.3.2	Describe, for combined loads, the criteria selected to provide mechanical and structural integrity. The loads and loading combinations to which the facility is subjected, including the load factors selected for each load component where a factored load approach is used, should be defined. The design approachused with the loading combination and any load factors should be specified. Describe the loads acting on the structures such as earth dead loads, live loads, and pressure loads, as well as the design basis accident loads and loads resulting from natural phenomena such as earthquakes, floods, tornadoes, hurri- canes, and missile effects unique for the site. The design loading combinations used to examine the effects on localized areas such as penetrations, structural discontinuities, prestressing tendon anchor zones, crane girder brackets, and local areas of high thermal gradients should be provided together with time- dependent loading such as the thermal effects, effects of creep and shrinkage, and other related effects. Explanation should be provided of the use of an ultimate strength approach with a load factor of 1.0.
Licensing, Design	3.3	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems**	4.4.3.1	No text provided.
Licensing, Design	3.3.1	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** General	4.4.3.1	Identify items requiring special consideration in design because of site selection, operating conditions, or other requirements. Since the spent fuel or high-level radioactive waste may be stored in SSSCs, drywells, or vaults, the long-term safety and secure confinement of these systems must be ensured. In addition, if the ISFSI or MRS includes systems for unloading shipping casks, transferring fuel or high-level radioactive waste to an SSSC or drywell, placing fuel or high-level radioactive waste in sealed containers, or other similar operations with fuel or high-level radioactive waste, each such operation should be considered in view of its operating hazards.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.2	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Protection by Multiple Confinement Barriers and Systems	11.4.2.4	No text provided.
Licensing, Design	3.3.2.1	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Protection by Multiple Confinement Barriers and Systems *Confinement Barriers and Systems*	11.4.2.4	Discuss each method of confinement that will be used to ensure that there will be no uncontrolled release of radioactivity to the environment. Include for each: 1. Criteria for protection against any postulated internal accident or external natural phenomena, 2. Design criteria selected for vessels, piping, effluent systems, and backup confinement, and 3. Delineation for each case of the extent to which the design is based on achieving the lowest practical level of releases from the operation of the installation. Where the release limits selected are consistent with proven practice, a referenced statement to that effect will suffice; where the limits extend beyond present practice, an evaluation and an explanation based on developmental work and/or analysis should be provided. Those criteria may be expressed as explicit numbers or as general conditions.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.2.2	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Protection by Multiple Confinement Barriers and Systems *Ventilation--Offgas*	11.4.2.4	Describe the criteria selected for providing suitable ventilation for fuel handling and storage structures by showing capacity standards for normal and off-normal conditions, zone interface flow velocity and differential pressure standards, the flow pattern, and control instrumentation. Establish the criteria for the design of the ventilation and offgas systems, including (1) airflow patterns and velocity with respect to contamination control, (2) minimum negative pressures at key points in the system to maintain proper flow control, (3) interaction of offgas systems with ventilation systems, (4) minimum filter performance with respect to particulate removal efficiency and maximum pressure drop, (5) minimum performance of other radioactivity removal equipment, and (6) minimum performance of dampers and instrumented controls.
Licensing, Design	3.3.3	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Protection by Equipment and Instrumentation Selection	4.4.3.1	No text provided.
Licensing, Design	3.3.3.1	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Protection by Equipment and Instrumentation Selection *Equipment*	4.4.3.1	Itemize design criteria for key equipment items that have been specifically selected to provide protection.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.3.2	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Protection by Equipment and Instrumentation Selection *Instrumentation*	4.4.3.1	Discuss the design criteria for instrumentation selected with particular emphasis on features to provide testability and contingency for safety purposes.
Licensing, Design	3.3.4	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Nuclear Criticality Safety	4.4.3.5	Supply all pertinent criteria relating to the appropriate safety margins provided to ensure that a subcritical situation exists at all times, both for passive storage and for handling operations.
Licensing, Design	3.3.4.1	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Nuclear Criticality Safety *Control Methods for Prevention of Criticality*	4.4.3.5	Present the methods to be used to ensure that subcritical situations are maintained in operations and storage under the worst credible conditions.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.4.2	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Nuclear Criticality Safety *Error Contingency Criteria*	4.4.3.5	To support the above information, define the error contingency criteria selected.
Licensing, Design	3.3.4.3	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Nuclear Criticality Safety *Verification Analyses*	4.4.3.5	Present the criteria for establishing verification of models or programs used in the analysis.
Licensing, Design	3.3.5	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Radiological Protection	11.4.2	A portion of the radiological protection design criteria was discussed in Section 3.3.2. Present any additional radiological protection design criteria.
Licensing, Design	3.3.5.1	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Radiological Protection *Access Control*	11.4.2.2	Describe the methods and procedures to be designed into the installation for limiting access, as necessary, to minimize exposure of people to radiation and radioactive materials.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.5.2	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Radiological Protection *Shielding*	11.4.2.3	Provide an estimate of collective doses (in person-rem) per year in each area and for various operations. When special provisions such as time and distance are to be included, determine the design dose rate in occupancy areas. Show that further reduction of collective doses is not practicable.
Licensing, Design	3.3.5.3	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Radiological Protection *Radiological Alarm Systems*	11.4.2.5	Describe the criteria used for action levels from radiological alarm systems. Describe the systems that will be used to ensure personnel and environmental protection from radiation and airborne radioactivity.
Licensing, Design	3.3.6	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Fire and Explosion Protection	6.4.5	Provide the design criteria selected to ensure that all safety functions will successfully withstand credible fire and explosion conditions.
Licensing, Design	3.3.7	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Materials Handling and Storage	4.4.3.1	No text provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.7.1	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Materials Handling and Storage *Spent Fuel or Hiqh-Level Radioactive Waste Handling and Storage*	4.4.3.1	Describe the design criteria for spent fuel or high-level radioactive waste handling and storage systems. Specifically cover cooling requirements, onsite movement criticality, and contamination control. Discuss criteria for handling damaged fuel elements or waste containers.
Licensing, Design	3.3.7.2	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Materials Handling and Storage *Radioactive Waste Treatment*	14.4.1	Describe the facilities to be used for the treatment and storage of radioactive wastes that are generated as a result of ISFSI or MRS operations, including (1) reduction in volume, (2) control of releases of radioactive materials during treatment, (3) conversion to solid forms, (4) suitability of product containers for storage or shipment to a disposal or storage site, (5) safe confinement during onsite storage, (6) monitoring during onsite storage to demonstrate safe confinement, and (7) final decontamination, retrieval, and disposal of stored wastes during decommissioning.
Licensing, Design	3.3.7.3	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Materials Handling and Storage *Waste Storage Facilities*	14.4.1	Describe the facilities associated with the storage of waste generated onsite as a result of the ISFSI or MRS operations.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.3.8	**PRINCIPAL DESIGN CRITERIA** **Safety Protection Systems** Industrial and Chemical Safety	4.4.3.1	Any specific design criteria that are important to personnel and plant safety should be described. Effects of various industrial accidents (e.g., fire and explosion) and potentially hazardous chemical reactions (e.g., spontaneous ignition of ion exchange resins) should be presented.
Licensing, Design	3.4	**PRINCIPAL DESIGN CRITERIA** **Classification of Structures, Components, and Systems**	4.4.2	Provide a classification of the structures, components, and systems selected in the design according to their importance as to the safety function they perform, the seismic design considerations, and the relationship of the quality requirements of an item with respect to its function and performance. As appropriate, this classification presentation should relate to details in Chapter 4, "Installation Design"; Chapter 5, "Operation Systems"; and Chapter 11, "Quality Assurance." Define the criteria for selecting the categories used for the classifications with regard to safety, seismic considerations, and quality assurance.
Licensing, Design, Decommissioning	3.5	**PRINCIPAL DESIGN CRITERIA** **Decommissioning Considerations**	4.4.3.6	The applicant should discuss the consideration given in the design of the facility and its auxiliary systems, including the storage structures, to facilitating eventual decommissioning. Examples of subjects to be covered are: (1) the provisions made for the decontamination and removal of potentially contaminated components of an air circulating and filtration system and (2) the components of waste treatment and packaging systems.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	3.6	**PRINCIPAL DESIGN CRITERIA** **Summary of Design Criteria**	None	Provide a summary of the design criteria for all structures, systems, and components that are important to safety. This summary may be presented in tabular form. It should include, as a minimum, the following items: 1. Maximum load capacity of cranes and other handling equipment, 2. Maximum dimensions of loads that can be handled, 3. Criticality factor, 4. Maximum dose rates (i.e., incoming casks, storage cask surfaces), 5. Ambient temperature, 6. Ambient humidity, 7. Tornado wind velocities (rotational and translational), 8. Tornado pressure drop, 9. Maximum winds, 10. Design earthquake peak acceleration, 11. Explosion peak overpressure, and 12. Flood elevations.
Licensing, Design, Quality Assurance	4	**INSTALLATION DESIGN**	5.4	Provide descriptive information on the buildings and other installed features of the installation and their locations on the site. Use drawings and maps as appropriate. Describe and evaluate each part of the installation with emphasis on those features that serve functions that are important to safety. Describe and evaluate special design features employed to withstand environmental forces and accident forces. Relate the design bases and use of industrial codes to the design criteria presented in Chapter 3, "Principal Design Criteria." Identify those features that are covered by the quality assurance program.
Licensing, Design	4.1	**INSTALLATION DESIGN** **Summary Description**	5.4	No text provided.
Licensing, Design	4.1.1	**INSTALLATION DESIGN** **Summary Description** Location and Layout of Installation	2.4.1.1	Identify the location of the storage structures, storage areas, and other installed facilities on a map or drawing to scale. Also include roadways, railroad lines, and utility and water service locations.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.1.2	**INSTALLATION DESIGN** **Summary Description** Principal Features	2.4.1.2	No text provided.
Licensing, Design	4.1.2.1	**INSTALLATION DESIGN** **Summary Description** Principal Features *Site Boundary*	2.4.1.2	Show the boundary that encompasses the area owned or controlled by the applicant.
Licensing, Design	4.1.2.2	**INSTALLATION DESIGN** **Summary Description** Principal Features *Controlled Area*	2.4.1.2	Show the controlled area established by the criteria in § 72.106 of 10 CFR Part 72.
Licensing, Design	4.1.2.3	**INSTALLATION DESIGN** **Summary Description** Principal Features *Site Utility Supplies and Systems*	2.4.1.2	Identify the utility supplies and systems, including the source(s) of water. Include the location of test wells and coolers.
Licensing, Design	4.1.2.4	**INSTALLATION DESIGN** **Summary Description** Principal Features *Storage Facilities*	2.4.1.2	Show the location of holding ponds, chemical and gas storage vessels, or other open-air tankage on or near the site that may or may not be associated with ISFSI or MRS operations.
Licensing, Design	4.1.2.5	**INSTALLATION DESIGN** **Summary Description** Principal Features *Stacks*	2.4.1.2	Show the location of any stacks in relationship to the other facilities.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.2	**INSTALLATION DESIGN** **Storage Structures**	4.4.3.1	Provide the design bases for storage structures such as vaults, SSSCs, or drywells, including (1) analysis and design procedures for tornado, earthquake, fire, explosion, and differential subsidence effects, (2) the general analysis and design procedures for normal, off-normal, and special loadings and load combinations, (3) allowable foundation loads and deflections and deformation stresses for structures, (4) provisions and methods for making connections between the proposed structures and future modifications and additions, and (5) consideration given to combination stress loadings.
Licensing, Design, Quality Assurance	4.2.1	**INSTALLATION DESIGN** **Storage Structures** Structural Specifications	4.4.3.2	Describe the bases and engineering design specifications of the storage structures. Discuss applicable nationally recognized codes and standards, the materials of construction, and the fabrication and inspection to be used, and itemize in tabular form activities that will be covered by the quality assurance program discussed in Chapter 11, "Quality Assurance."
Licensing, Design	4.2.2	**INSTALLATION DESIGN** **Storage Structures** Installation Layout	5.4	No text provided.
Licensing, Design	4.2.2.1	**INSTALLATION DESIGN** **Storage Structures** Installation Layout *Building Plans*	5.4	Provide engineering drawings, plans, and elevations showing the layout of the functional features of the storage structures. Show sufficient detail to identify all features to be discussed in this chapter. Include spatial and equipment identification data directly on the layouts with suitable designations in tabular listings. Provide engineering drawings, plans, and elevations showing the total array of the SSSCs, the drywells, or vault storage cells, as applicable, and their auxiliaries.
Licensing, Design	4.2.2.2	**INSTALLATION DESIGN** **Storage Structures** Installation Layout *Building Sections*	5.4	Include sectional drawings to relate all features to be discussed in this chapter.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.2.2.3	**INSTALLATION DESIGN** **Storage Structures** Installation Layout *Confinement Features*	5.4.1.1	Identify and discuss general layout criteria for the installation that have been included in the design to ensure confinement of radioactivity. This should be a general discussion with details to be presented in the appropriate part of this chapter. Include in the discussion ventilation, piping, and other physical means such as barriers, encasements, liners, and protective coatings. Identify the interfaces between the systems, and discuss the safety aspects of the interfaces. Details on ventilation systems should be presented in Chapter 7, "Radiation Protection."
Licensing, Design	4.2.3	**INSTALLATION DESIGN** **Storage Structures** Individual Unit Description	5.4	List the operational areas associated with SSSC placement (if the SSSCs are not of the permanently located type) and monitoring locations while in storage. Show the location of each by using engineering drawings.
Licensing, Design	4.2.3.1	**INSTALLATION DESIGN** **Storage Structures** Individual Unit Description *Function*	5.4	Describe the function of the individual operations, and discuss the performance objectives.
Licensing, Design	4.2.3.2	**INSTALLATION DESIGN** **Storage Structures** Individual Unit Description *Components*	5.4	Discuss the components used for the operation. Use individual equipment sketches, layouts of equipment location to identify aspects of the components that must be relied on, and limits imposed on the design to achieve safety.
Licensing, Design	4.2.3.3	**INSTALLATION DESIGN** **Storage Structures** Individual Unit Description *Design Bases and Safety Assurance*	5.4.1.2	Present the design codes used and additional specifications necessary to provide a sufficient margin of safety under normal and accident conditions to ensure that a single failure will not result in the release of significant radioactive material. Detail on backup provisions and interfaces with other areas should be included. Include a discussion of the features used to ensure that operating personnel are protected from radiation and that criticality will not occur.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3	**INSTALLATION DESIGN** **Auxiliary Systems**	3.4.3	Provide information on auxiliary systems that are important to safety for the installation. Emphasis should be placed on provisions for coping with unscheduled occurrences in a manner that will preclude an unsafe condition. Define the design bases, codes, specifications, and standards that will provide a safety margin to ensure that a single failure within a support system will not result in releases of radioactive materials. For certain auxiliary systems involving building ventilation, electric power, air, and water, three categories of loads are possible: 1. Loads determined by normal operations, 2. Load situations resulting from primary failure and/or accident conditions, and 3. Emergency load defined as the minimum requirement for the total safety of a shutdown operation, including its surveillance requirements. Minimum loads are further defined as the design characteristics for the confinement systems that are required for such systems to prevent the release of radioactive materials under design basis accident conditions. Describe the location of the various auxiliary systems in relationship to their functional objectives. This section should refer to drawings presented in Sections 4.2.2 and 4.7.2 and should present additional details to identify the detailed physical arrangement. For each auxiliary system, as appropriate, provide single line drawings and a narrative description of its operating characteristics and safety considerations.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.1	**INSTALLATION DESIGN** **Auxiliary Systems** Ventilation and Offgas Systems	3.4.3	Describe in detail the design, operating features, and limitations for performance of the ventilation-filtration systems to show that there will be sufficient backup, excess capacity, repair and replacement capability, and structural integrity to ensure controlled airflow in all credible circumstances to minimize release of radioactive particulates. Supplement the discussion with appropriate drawings to show the flow distribution, pressure differentials, flow quantity, velocity, and filter and fan housing arrangements. Identify each of the areas serviced and the interfaces among areas in the following sections:
Licensing, Design	4.3.1.1	**INSTALLATION DESIGN** **Auxiliary Systems** Ventilation and Offgas Systems *Major Components and Operating Characteristic* s	3.4.3	Present the design bases selected for the building and unit ventilation systems. Present detailed discussions justifying these bases, the system designs, and operating characteristics. Describe the components making up each system and the relationship of the various systems to one another. Describe each system in terms of air supply, their collection and distribution systems, modes of gas conditioning, jetting, sequence of filtration, filter protection, the exhaust fans, and the stack. For clarity, provide and reference in the discussion appropriate engineering drawings and sketches.
Licensing, Design	4.3.1.2	**INSTALLATION DESIGN** **Auxiliary Systems** Ventilation and Offgas Systems *Safety Considerations and Controls*	3.4.3	Emphasize that the design features ensure confinement of radioactive articulates under conditions of power failure, adverse natural phenomena, breakdown of equipment, fire and explosion, improper flow of air, contaminated spills, and loss of fiTter integrity.
Licensing, Design	4.3.2	**INSTALLATION DESIGN** **Auxiliary Systems** Electrical Systems	3.4.3	No text provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.2.1	**INSTALLATION DESIGN** **Auxiliary Systems** Electrical Systems *Major Components and Operating Characteristics*	3.4.3	Discuss the source and characteristics of the primary electrical system providing normal power to the installation. Provide a description of the source of the secondary system, if applicable. Describe the design providing for the emergency power source(s) and the means for ensuring an uninterruptible service to those items requiring it. For each of these latter items, list the equipment and systems serviced, locations, required kilowatts, and type of startup system for each.
Licensing, Design	4.3.2.2	**INSTALLATION DESIGN** **Auxiliary Systems** Electrical Systems *Safety Considerations and Controls*	3.4.3	Itemize and discuss the mechanisms and sequence and timing of events that will occur in the event of a partial loss of normal power and in the event of a total loss of normal power to ensure safe storage conditions and shutdown of handling operations. Present the design features pertinent to the use of emergency power. Also describe the procedure for subsequent reestablishment of normal load service.
Licensing, Design	4.3.3	**INSTALLATION DESIGN** **Auxiliary Systems** Air Supply Systems	3.4.3	No text provided.
Licensing, Design	4.3.3.1	**INSTALLATION DESIGN** **Auxiliary Systems** Air Supply Systems *Compressed Air*	3.4.3	Present the design for supplying the compressed air needs of the plant, the components, and their location and operating characteristics. Include a description of the compressors, receivers and dryers, and distribution systems.
Licensing, Design	4.3.3.2	**INSTALLATION DESIGN** **Auxiliary Systems** Air Supply Systems *Breathing Air*	3.4.3	Present the design for supplying the breathing air needs of the facility. Include a description of the compressors, receivers and dryers, alarms and safety systems, and distribution systems. Discuss in detail the backup provisions for the breathing air system and its ability to function during emergency situations.
Licensing, Design	4.3.4	**INSTALLATION DESIGN** **Auxiliary Systems** Steam Supply and Distribution System	3.4.3	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.4.1	**INSTALLATION DESIGN** **Auxiliary Systems** Steam Supply and Distribution System *Major Components and Operating Characteristics*	3.4.3	Present the design for supplying steam to the plant, including a discussion of the fuel supply and boiler type.
Licensing, Design	4.3.4.2	**INSTALLATION DESIGN** **Auxiliary Systems** Steam Supply and Distribution System *Safety Considerations and Controls*	3.4.3	Discuss features of the steam supply system with respect to continuity of operations that are important to safety.
Licensing, Design	4.3.5	**INSTALLATION DESIGN** **Auxiliary Systems** Water Supply System	3.4.3	No text provided.
Licensing, Design	4.3.5.1	**INSTALLATION DESIGN** **Auxiliary Systems** Water Supply System *Major Components and Operating Characteristics*	3.4.3	For the water supply, discuss the primary source, alternative sources, storage facilities, and supply system. Itemize design considerations to demonstrate the continuity of the water supply. Also itemize by service (potable, operations such as cask washdown, and fire) the quantities of water-used under normal and off-normal conditions.
Licensing, Design	4.3.5.2	**INSTALLATION DESIGN** **Auxiliary Systems** Water Supply System *Safety Considerations and Controls*	3.4.3	Discuss the effects of loss of water supply source, failure of main supply pumps or supply lines, and power failure. Also discuss the means for coping with drought and flood conditions.
Licensing, Design	4.3.6	**INSTALLATION DESIGN** **Auxiliary Systems** Sewage Treatment System	3.4.3	No text provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.6.1	**INSTALLATION DESIGN** **Auxiliary Systems** Sewage Treatment System *Sanitary Sewage*	3.4.3	Describe the sanitary sewage handling system to show that no radioactive material can be discharged in this effluent.
Licensing, Design	4.3.6.2	**INSTALLATION DESIGN** **Auxiliary Systems** Sewage Treatment System *Chemical Sewage*	3.4.3	Describe any system that may be used for handling and treatment of other nonradioactive liquid effluents.
Licensing, Design	4.3.7	**INSTALLATION DESIGN** **Auxiliary Systems** Communications and Alarm Systems	3.4.3	No text provided.
Licensing, Design	4.3.7.1	**INSTALLATION DESIGN** **Auxiliary Systems** Communications and Alarm Systems *Major Components and Operating Characteristics*	3.4.3	Discuss the systems for external and internal communications with particular emphasis on the facilities to be used under emergency conditions.
Licensing, Design	4.3.7.2	**INSTALLATION DESIGN** **Auxiliary Systems** Communications and Alarm Systems *Safety Considerations and Controls*	3.4.3	Describe the functioning of the communication systems and alarms in response to normal and off-normal operations and under accident conditions.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.8	**INSTALLATION DESIGN** **Auxiliary Systems** Fire Protection System	3.4.3	No text provided.
Licensing, Design	4.3.8.1	**INSTALLATION DESIGN** **Auxiliary Systems** Fire Protection System *Design Bases*	3.4.3	1. Identify the fires that could indirectly or directly affect structures, systems, and components that are important to safety. Describe and discuss those fires that provide the bases for the design of the fire protection system, i.e., fires considered to be the maximum fire that may develop in local areas assuming that no manual, automatic, or other firefighting measures have been started and the fire has passed flashover and is reaching its peak burning rate before firefighting can start. Consider fire intensity, location, and (depending on the effectiveness of fire protection) the duration and effect on adjacent areas. 2. Discuss fire characteristics such as maximum fire intensity, flame spreading, smoke generation, production of toxic contaminants, and the contribution of fuel to the fire for all individual installation areas that have combustible materials and are associated with storage structures or other structures, systems, and components that are important to safety. Include in the discussion the use and effect of noncombustible and heat-resistant materials. Provide a list of the dangerous and hazardous combustibles and the maximum amounts estimated to be present. State where these will be located in the installation in relationship to storage and safety systems. 3. Discuss and list the features of building and installation arrangements and the structural design features that provide for fire prevention, fire extinguishing, fire control, and control of hazards created List by fire. and describe in the discussion the egress, fire barriers, fire walls, and the isolation and confinement features provided for flame, heat, hot gases, smoke, and other contaminants. 4. List the codes and standards considered and used for the design of the fire protection systems, including published standards of the National Fire Protection Association.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.8.2	**INSTALLATION DESIGN** **Auxiliary Systems** Fire Protection System *System Description*	3.4.3	1. Provide a general description of the fire protection system, including drawings showing the physical characteristics of the installation location and outlining the fire prevention and fire suppression systems to be provided for all areas associated with physical security storage structures and other structures, systems, and components that are important to safety. 2. Discuss the protection and suppression systems provided in the control room and other operating areas containing security equipment and other equipment important to safety. 3. Describe the design features of detection systems, alarm systems, automatic fire suppression systems, and manual, chemical, and gas systems for fire detection, confinement, control, and extinguishing. Discuss the relationship of the fire protection system to the onsite ac and dc power sources. 4. Discuss smoke, heat, and flame control; combustible and explosive gas control; and toxic contaminant control, including the operating functions of the ventilating and exhaust systems during the period of fire extinguishing and control. Discuss the fire annunciator warning system, the appraisal and trend evaluation systems provided with the alarm detection system in the pro- posed fire protection systems, and the backup or public fire protection if this is to be provided in the installation. Include drawings and a list of equipment and devices that adequately define the principal and auxiliary fire protection systems. 5. Describe electrical cable fire protection and detection and the fire confinement, control, and extinguishing systems provided. Define the integrity of the essential electric circuitry needed during the fire for safe shutdown of operations and for firefighting. Describe the provisions made for protecting this essential electrical circuitry from the effects of fire-suppressing agents.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.3.8.3	**INSTALLATION DESIGN** **Auxiliary Systems** Fire Protection System *System Evaluation*	3.4.3	Provide an evaluation for those fires identified in Section 4.3.8.1. This evaluation should consider the quantities of combustible materials present, the installation design, and the fire protection systems provided. Describe the estimated severity, intensity, and duration of the fires and the hazards created by the fires. Indicate for each of the postulated events the total time involved and the time for each step from the first alert of the fire hazard until safe control or extinguishment is accomplished. Provide a failure mode and effects analysis to demonstrate that operation of the fire protection system in areas containing security and operational safety features would not produce an unsafe condition or preclude safe shutdown of operations. An evaluation of the effects of failure of any portion of the fire protection system not designed to seismic requirements should be provided with regard to the possibility of damaging other equipment. Include an analysis of the fire detection and protection system with regard to design features to withstand the effects of single failures.
Licensing, Design	4.3.8.4	**INSTALLATION DESIGN** **Auxiliary Systems** Fire Protection System *Inspection and Testing Requirements*	3.4.3	List and discuss the installation, testing, and inspection planned during construction of the fire protection systems to demonstrate the integrity of the systems as installed. Describe the operational checks, inspection, and servicing required to maintain this integrity. Discuss the routine testing necessary to maintain a highly reliable alarm detection system.
Licensing, Design	4.3.8.5	**INSTALLATION DESIGN** **Auxiliary Systems** Fire Protection System *Personnel Qualification and Training*	3.4.3	State the qualification requirements for the fire protection engineer or consultant who will assist in the design and selection of equipment, inspect and test the, completed physical aspects of the system, develop the fire protection program, and assist in thefirefighting training for the operating installation. Discuss the initial training and the updating provisions such as fire drills provided for maintaining the competence of the station firefighting and operating crew, including personnel responsible for maintaining and inspecting the fire protection equipment.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design, Operation	4.3.9	**INSTALLATION DESIGN** **Auxiliary Systems** Maintenance Systems	3.4.3	No text provided.
Licensing, Design, Operation	4.3.9.1	**INSTALLATION DESIGN** **Auxiliary Systems** Maintenance Systems *Major Components and Operating Characteristics*	3.4.3	Provide the design bases, locations, and modes of operation related to the maintenance programs for the installation. Emphasis should be placed on provisions for maintenance of remotely operated equipment and ventilation system components; hot cell components; decontamination and disposal of contaminated equipment, piping, and valves; quality control; and testing.
Licensing, Design, Operation	4.3.9.2	**INSTALLATION DESIGN** **Auxiliary Systems** Maintenance Systems *Safety Considerations and Controls*	3.4.3	Discuss the means for conducting required maintenance with a minimum of personnel radiation exposure or injury as a result of designing for accessibility for maintenance and ensuring the confinement of contaminated materials and radioactive wastes as necessary.
Licensing, Design, Operation	4.3.10	**INSTALLATION DESIGN** **Auxiliary Systems** Cold Chemical Systems	3.4.3	Describe the major components and operating characteristics of facilities that will be used in association with cold chemical operations. If hazardous chemicals or materials are involved, discuss the provisions for mitigating accidents. Itemize the chemicals and materials to be used and their quantities, indicate where they will be used, codify them with respect to hazard, and discuss the potential impact of an accident involving their use on the storage system.
Licensing, Design, Operation	4.3.11	**INSTALLATION DESIGN** **Auxiliary Systems** Air Sampling Systems	3.4.3	Discuss the various types of air sampling systems; include design and operating features for each system. Include, in detail, limitations for performance of the air sampling systems to show there will be sufficient vacuum and backup capability to ensure that proper sampling will be conducted in all credible circumstances. Supplement the discussion to show with appropriate drawings flow quantity, fixed-head and constant air monitor placements, and vacuum pump and exhaust arrangements. Identify each of the areas serviced, and show how each area is interconnected.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design, Operation	4.3.11.1	**INSTALLATION DESIGN** **Auxiliary Systems** Air Sampling Systems *Major Components and Operating Characteristics*	3.4.3	Present the design selected for the room and area air sampling systems. Present detailed discussions justifying the system design and operating characteristics. Describe the components of each system and the relationship of the various systems to each other. Describe each system in terms of vacuum supply, collec- tion system, and exhaust points. For clarity, provide and reference in the discussion the appropriate engineering drawings.
Licensing, Design, Operation	4.3.11.2	**INSTALLATION DESIGN** **Auxiliary Systems** Air Sampling Systems *Safety Considerations and Controls*	3.4.3	Discuss features of the air sampling systems with respect to continuity of operations to ensure that sampling is conducted during off-normal conditions.
Licensing, Design, Operation	4.4	**INSTALLATION DESIGN** **Decontamination Systems**	**3.4.3**	No text provided.
Licensing, Design, Operation	4.4.1	**INSTALLATION DESIGN** **Decontamination Systems** Equipment Decontamination	3.4.3	Describe the design and operating features of the equipment decontamination system. Discuss the various decontamination techniques that will be available as part of this system and the limitations of each technique.
Licensing, Design, Operation	4.4.1.1	**INSTALLATION DESIGN** **Decontamination Systems** Equipment Decontamination *Major Components and Operating Characteristics*	3.4.3	Present the design selected for the equipment decontamination system. Present detailed discussions justifying the system design and operating characteristics. Describe the components of this system and how this system interacts with the other service and utility systems. Discuss the ventilation requirements for this system. For clarity, provide and reference in the discussion the appropriate engineering drawings.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design, Operation	4.4.1.2	**INSTALLATION DESIGN** **Decontamination Systems** Equipment Decontamination *Safety Considerations and Controls*	3.4.3	Emphasize the design features that ensure confinement of radioactive waste generated by this system. Discuss the design features to ensure that radiation exposure received by workers during the decontamination operations will be as low as is reasonably achievable.
Licensing, Design, Operation	4.4.2	**INSTALLATION DESIGN** **Decontamination Systems** Personnel Decontamination	3.4.3	Describe the design and operating features of the personnel decontamination system. Discuss the type of decontamination that will be available and the limitations of this system. Describe actions that will be taken if decontamination requirements exceed the limitations of this system.
Licensing, Design, Operation	4.5	**INSTALLATION DESIGN** **Shipping Cask Repair and Maintenance**	3.4.7	Indicate the location of the shipping cask repair and maintenance facility or area on a plot plan of the ISFSI or MRS. Provide an engineering drawing of the shop layout with major items of equipment identified. This activity may be incorporated into other maintenance areas or facilities. Describe planned modes of operation with emphasis on contamination control and occupational exposure reduction.
Licensing, Design, Operation	4.6	**INSTALLATION DESIGN** **Cathodic Protection**	3.4.3	Describe the design and operating characteristics of the cathodic protection system provided for the drywells or any other affected structures. Reference to cathodic protection is not meant to preclude nonelectric means of corrosion protection, which, if used, should be described with respect to their design and operating characteristics.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design, Operation	4.7	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems**	3.4.2	Spent fuel or high-level radioactive waste handling facilities will be needed at the facility site for some or all of the following functions: receiving and inspection of loaded shipping casks, cask unloading, spent fuel or highlevel radioactive waste transfer and examination, fuel assembly/disassembly, placement of spent fuel in a container, container sealing and testing, spent fuel or high-level radioactive waste container short-term storage, shipping cask decontamination, SSSC and drywell loading and preparation for storage, SSSC transfer to storage, fuel or high-level radioactive waste container removal from storage site to shipping cask, and damaged fuel element containerization. The functions and design bases for systems and structures to perform these operations should be described, including (1) analysis and design procedures for tornado, earthquake, fire, explosion, and differential subsidence effects, (2) the general analysis and design procedures for normal, off-normal, and special loadings and load combinations, (3) allowable foundation loads and deflections and deformation stresses for structures, (4) provision and methods for making connections between planned structures and future modifications and additions, and (5) considerations given to combination of stress loadings.
Licensing, Design, Quality Assurance	4.7.1	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Structural Specifications	3.4.2	Establish the bases and engineering design required to maintain the structural integrity of the spent fuel or high-level radioactive waste handling operation systems. Where applicable, identify nationally recognized codes and standards, the materials of construction, and the fabrication and inspection to be used, and itemize in tabular form features that will be covered by the quality assurance program discussed in Chapter 11, "Quality Assurance." Identify the specifications and design details covering the information discussed in Section 4.3.
Licensing, Design	4.7.2	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Installation Layout	3.4.2	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	4.7.2.1	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Installation Layout *Building Plans*	3.4.2	Provide engineering drawings, plans, and elevations showing the layout of the functional features of buildings. Show sufficient detail to identify all features to be discussed in this chapter. Include spatial and equipment identification data directly on the layouts with suitable designations in tabular listings.
Licensing, Design	4.7.2.2	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Installation Layout *Building Sections*	3.4.2	Include sectional drawings features to relate all to be discussed in this chapter.
Licensing, Design	4.7.2.3	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Installation Layout *Confinement Features*	3.4.2	Identify and discuss general layout criteria for the installation that have been included in the design to ensure confinement of radioactivity. This should be a general discussion with details to be presented in the appropriate part of this chapter. Include in the discussion ventilation, piping, and other physical means such as barriers, encasements, liners, and protective coatings. Identify the interfaces between the systems, and discuss the safety aspects of the interfaces. Details on ventilation systems should be presented in Chapter 7, "Radiation Protection."
Licensing, Design, Operation	4.7.3	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Individual Unit Description	3.4.2	List each operational unit sequentially from the receipt of spent fuel or high-level radioactive waste through the various operations. The following are typical items: shipping cask receiving and inspecting, cask unloading, spent fuel or high-level radioactive waste canister transfer, spent fuel or high-level radioactive waste storage, hot-cell operations, and control loca- tions. Show the location of each by use of engineering drawings.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design, Operation	4.7.3.1	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Individual Unit Description *Function*	3.4.2	Describe the function of the individual operational areas, and discuss the performance objectives.
Licensing, Design, Operation	4.7.3.2	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Individual Unit Description *Components*	3.4.2	Discuss the components in the area under discussion. Use individual equipment sketches, layouts of equipment location to identify aspects of the components that must be relied on, and limits imposed on the design to achieve safety objectives.
Licensing, Design, Operation	4.7.3.3	**INSTALLATION DESIGN** **Spent Fuel and High-Level Radioactive Waste Handling Operation Systems** Individual Unit Description *Design Bases and Safety Assurance*	3.4.2	Present the design codes used and additional specifications necessary to provide a sufficient margin of safety under normal and accident conditions to ensure that a single failure will not result in the release of significant radioactive material. Detail on backup provisions and interfaces with other areas should be included. Also include a discussion of the features used to ensure that operating personnel are protected from radiation and contamination and that criticality will not occur.
Licensing, Operation	5	**OPERATION SYSTEMS**	3.4	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	5.1	**OPERATION SYSTEMS** **Operation Description**	3.4.1	In this chapter, provide a detailed description of all operations, including systems, equipment, and instrumentation and their operating characteristics. Identify potentially hazardous operation systems. Provisions made for operation safety features to ensure against a hazard should be so designated in the details presented. The latter information should include, but not be limited to, listing systems necessary for curtailing operations under normal and off-normal conditions, maintaining the installation in a safe condition, secondary confinement, and backup or standby features. In addition to describing the operations, reference the items that will require continuing attention with respect to the quality assurance program after installation startup. For each system, describe the considerations used to achieve as low as is reasonably achievable (ALARA) levels of radioactive material in the installation effluents and to ensure safe nuclear conditions at all times. The SAR should show a definition of limits and parameters for developing the Technical License Conditions (Technical Specifications).
Licensing, Operation	5.1.1	**OPERATION SYSTEMS** **Operation Description** Narrative Description	3.4.1	Describe the proposed spent fuel and high-level radioactive waste handling and passive storage operations, and relate them to the equipment and associated controls. Include in this discussion ancillary activities as pertinent, i.e., preparation of reactants, offgas handling, volume reduction of wastes, and decontamination. In the description, identify the interfaces between systems, and discuss the safety aspects of the interfaces. Describe the means that will be routinely used during storage to evaluate the condition of the SSSCs and drywells and their associated containment systems, e.g., monitoring of external radiation, interior and/or external temperatures, and for leakage. Conduct periodic examinations for structural deterioration, foundation soundness, and security of contents.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design Operation	5.1.2	**OPERATION SYSTEMS** **Operation Description** Flowsheets	3.4.1	In support of the description above, supply flowsheets showing the sequence of operations and their controls. Provide identification of each step in sufficient detail so that an independent review can be made to ensure a safe operation. Provide the flow input and output characteristics for effluent control equipment for effluent streams to show the efficiencies obtained. Sufficient detail should be given to provide source terms for radiation exposure determinations to be developed in Chapter 7, "Radiation Protection." Include equipment descriptions with dimensions, design and operating characteristics, materials of construction, special design features, and operating limitations. Appropriate engineering and operating instrumentation details should be provided.
Licensing, Design Operation	5.1.3	**OPERATION SYSTEMS** **Operation Description** Identification of Subjects for Safety Analysis	3.4.1	Identify subjects for safety analysis. Reference this part of the as chapter, applicable, in subsequent discussions of design and operating features.
Licensing, Design Operation	5.1.3.1	**OPERATION SYSTEMS** **Operation Description** Identification of Subjects for Safety Analysis *Criticality Prevention*	3.4.1	Provide a summary description of the principal design features, procedures, and special techniques used to preclude criticality in all portions of the installation.
Licensing, Operation	5.1.3.2	**OPERATION SYSTEMS** **Operation Description** Identification of Subjects for Safety Analysis *Chemical Safety*	3.4.1	Provide a summary description of any chemical hazards and the approaches used to preclude associated accidents.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	5.1.3.3	**OPERATION SYSTEMS** **Operation Description** Identification of Subjects for Safety Analysis *Operation Shutdown Modes*	3.4.1	Describe the general conditions and surveillance needs in various shutdown modes (extended, short-term, emergency). Indicate the time required to shut down and start up for each mode.
Llicensing,Operation, Design	5.1.3.4	**OPERATION SYSTEMS** **Operation Description** Identification of Subjects for Safety Analysis *Instrumentation*	3.4.1	Provide a summary description of the instruments used to detect operating conditions and the systems used to control operations. The description should include testability, redundancy, and failure conditions. Also describe effluent and process monitors and data loggers.
Licensing, Operation	5.1.3.5	**OPERATION SYSTEMS** **Operation Description** Identification of Subjects for Safety Analysis *Maintenance Techniques*	3.4.1	Discuss the rationale for and outline the techniques to be used for major maintenance tasks. This discussion should include a statement of areas where specific techniques apply. Include system and component spares.
Licensing, Operation, Design	5.2	**OPERATION SYSTEMS** **Spent Fuel or High-Level Radioactive Waste Handling Systems**	3.4.2	Each of the following sections is intended to provide an understanding of the functions, design bases, and pertinent design features of the operating system as they relate to installation or environmental safety. To the extent pertinent, sketches should be used to describe unique equipment or design features.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.2.1	**OPERATION SYSTEMS** **Spent Fuel or High-Level Radioactive Waste Handling Systems** Spent Fuel or High-Level Radioactive Waste Receipt, Handling, and Transfer	3.4.2	Describe the systems associated with spent fuel or high-level radioactive waste receipt, transfer, and removal from the storage structure for shipment. From the design criteria, present the provisions for cooling and maintaining fuel assemblies and waste containers in subcritical arrays and the provisions for shielding.
Licensing, Operation, Design	5.2.1.1	**OPERATION SYSTEMS** **Spent Fuel or High-Level Radioactive Waste Handling Systems** Spent Fuel or High-Level Radioactive Waste Receipt, Handling, and Transfer *Functional Description*	3.4.2	Present a flow diagram and functional description of the spent fuel or high-level radioactive waste receiving, storage, and retrieval systems, including provisions for handling defective fuel assemblies or damaged high-level radioactive waste containers. Include drawings or references to drawings as needed.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.2.1.2	**OPERATION SYSTEMS** **Spent Fuel or High-Level Radioactive Waste Handling Systems** Spent Fuel or High-Level Radioactive Waste Receipt, Handling, and Transfer *Safety Features*	3.4.2	Describe all features, systems, or special handling techniques included in the system that provide for the safety of the operation under both normal and off-normal conditions. Include the limit(s) selected for a commitment to action.
Licensing, Operation	5.2.2	**OPERATION SYSTEMS** **Spent Fuel or High-Level Radioactive Waste Handling Systems** Spent Fuel or High-Level Radioactive Waste Storage	3.4.2	Describe the operations used for transfer of spent fuel assemblies or high-level radioactive waste containers to the storage position, the storage surveillance program, and the operations used for removal from the storage position.
Licensing, Operation	5.2.2.1	**OPERATION SYSTEMS** **Spent Fuel or High-Level Radioactive Waste Handling Systems** Spent Fuel or High-Level Radioactive Waste Storage *Safety Features*	3.4.2	Describe all features, systems, and special techniques included in the system that provide for the safety of the operation under both normal and off-normal conditions. Include the limit(s) selected for a commitment to action.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.3	**OPERATION SYSTEMS** **Other Operating Systems**	3.4.3	Each operating system should be related to the process description and appropriate flowsheets. Where appropriate, identify the system as a source of effluents and onsite-generated wastes, discussed in Chapter 6, "Site-Generated Waste Confinement and Management," and Chapter 7, "Radiation Protection." Reference the physical layout presentations discussed in Chapter 4, "Installation Design." Use subsections to present the information on each operating system.
Licensing, Operation, Design	5.3.1	**OPERATION SYSTEMS** **Other Operating Systems** Operating System	3.4.3	Name the actual operating system described in this section. Continue additional systems sequentially (e.g., 5.3.1-1, 5.3.1-2 ...).
Licensing, Operation, Design	5.3.1.1	**OPERATION SYSTEMS** **Other Operating Systems** Operating System *Functional Description*	3.4.3	Describe the portion of the operating system to be discussed, its function, and how the function will be accomplished.
Licensing, Operation, Design	5.3.1.2	**OPERATION SYSTEMS** **Other Operating Systems** Operating System *Major Components*	3.4.3	If more than one component is included in a particular system, explain the interrelationship of the individual components and the means by which these are combined within the system.
Licensing, Operation, Design	5.3.1.3	**OPERATION SYSTEMS** **Other Operating Systems** Operating System *Design Description*	3.4.3	Discuss the design bases; design capacity, including materials of construction; pressure and temperature limits; corrosion allowances; and standards or codes used. Itemize material and fabrication specifications pertaining to the system in sufficient detail to relate, as appropriate, to Chapter 9, "Conduct of Operations," and Chapter 11, "Quality Assurance." Describe the layout of equipment from the standpoint of minimizing personnel exposures to radiation during operations and maintenance. With suitable cross-reference, it will not be necessary to duplicate this information in Chapter 9 or in Chapter 11.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.3.1.4	**OPERATION SYSTEMS** **Other Operating Systems** Operating System *Safety Criteria and Assurance*	3.4.3	From the parameters discussed in the preceding sections, summarize the criteria for the means of ensuring a safe system as constructed, operated, and maintained. Summarize those limit(s) selected for commitment to action. Identify those items that can be characterized as being operation safety features that are considered necessary beyond normal operation and control. Emphasis should be placed on personnel exposure considerations.
Licensing, Operation, Design	5.3.1.5	**OPERATION SYSTEMS** **Other Operating Systems** Operating System *Operating Limits*	3.4.3	Identify limits, conditions, and performance requirements in sufficient detail to make possible an evaluation as to whether a Technical License Condition may be necessary. The relationship systems to other should be clearly described.
Licensing, Operation, Design	5.3.2	**OPERATION SYSTEMS** **Other Operating Systems** Component/Equip-ment Spares	3.4.3	Describe in detail design features that include installation of spare or alternative equipment to provide continuity of safety under normal and off-normal conditions. Particular emphasis is needed on design provisions to minimize exposure to radiation for maintenance operations. Describe the bases for inspection, preventive maintenance, and testing programs to ensure continued safe functioning.
Licensing, Operation, Design	5.4	**OPERATION SYSTEMS** **Operation Support Systems**	3.4.4	Although effluent handling systems may be considered operation support, these systems should be discussed in Chapter 6, "Site-Generated Waste Confinement and Management." Describe any chemical systems used to monitor or control the operations described in Chapter 4, "Installation Design." Principal auxiliary backup equipment should also be discussed in Chapter 4.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.4.1	**OPERATION SYSTEMS** **Operation Support Systems** Instrumentation and Control Systems	3.4.4	By means of instrumentation engineering flowsheet(s) of the operations, discuss the instrumentation and control features associated with operation control, monitors and alarms, and the relationship of one to the other. Identify those aspects relied on to establish that adequate reliability is provided and that provisions have been included in the design to ensure continued safe operation or safe curtailment of operations under accident conditions. Relate these to the design criteria presented in Chapter 3, "Principal Design Criteria." Discuss how instrumentation and control systems monitor safety-related variables and operating systems over anticipated ranges for normal operation, off-normal operation, accident conditions, and safe shutdown. Describe the redundancy of safety features necessary to ensure adequate safety of spent fuel or high-level radioactive waste storage operations. The variables and systems that are important to safety and may need constant surveillance and control include (1) atmospheric conditions such as precipitation, winds, and air temperature, (2) water and air radioactivity levels, and (3) confinement leakage indications. Storage area radiation and airborne radioactivity levels also may require constant monitoring. Discuss the provisions for in situ testability of the instrumentation and control systems, particularly for sumps, sump pumps, sump liquid level monitors, and other hard-to-get-at equipment. Describe how instrumentation and control systems are designed to be fail-safe or to assume a state demonstrated to be acceptable if conditions such as disconnection, loss of energy or motive power, or adverse environments are experienced. For each, provide the following information:

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.4.1.1	**OPERATION SYSTEMS** **Operation Support Systems** Instrumentation and Control Systems *Functional Description*	3.4.4	No text provided. (Provide functional description of instrumentation and control systems.)
Licensing, Operation, Design	5.4.1.2	**OPERATION SYSTEMS** **Operation Support Systems** Instrumentation and Control Systems *MaJor Components*	3.4.4	No text provided. (Describe the major components of the instrumentation and control systems and explain the interrelationship of the individual components and the means by which these are combined with the system.)
Licensing, Operation, Design	5.4.1.3	**OPERATION SYSTEMS** **Operation Support Systems** Instrumentation and Control Systems *Detection System and Locations*	3.4.4	No text provided. (Describe detection systems, their locations and how they are interconnected into the instrumentation and control systems.)
Licensing, Operation, Design	5.4.1.4	**OPERATION SYSTEMS** **Operation Support Systems** Instrumentation and Control Systems *Operating Characteristics*	3.4.4	No text provided. (Describe the operating characteristics of the instrumentation and controls systems.)

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	5.4.1.5	**OPERATION SYSTEMS** **Operation Support Systems** Instrumentation and Control Systems *Safety Criteria and Assurance*	3.4.4	No text provided. (From the parameters discussed in the preceding sections, summarize the criteria for the means of ensuring a safe system as constructed, operated, and maintained. Summarize those limit(s) selected for commitment to action. Identify those items that can be characterized as being operation safety features that are considered necessary beyond normal operation and control. Emphasis should be placed on personnel exposure considerations.)
Licensing, Operation, Design	5.4.2	**OPERATION SYSTEMS** **Operation Support Systems** System and Component Spares	3.4.4	Describe in detail installation of spare or alternative instrumentation designed to provide continuity of operation under normal and off-normal conditions. Also describe the bases for inspection, preventive maintenance, and testing programs to ensure continued safe functioning.
Licensing, Operation, Design	5.5	**OPERATION SYSTEMS** **Control Room and Control Areas**	3.4.5	Discuss how a control room and control areas are to be designed to permit occupancy and actions to be taken to operate the installation safely under both normal and off-normal conditions. Describe the redundancy that allows the installation to be put into a safe condition and the monitoring of this condition if any control room or control area is removed from service.
Licensing, Operation, Design	5.6	**OPERATION SYSTEMS** **Analytical Sampling**	3.4.6	Provisions for obtaining samples for analysis and controls necessary to ensure that operations are within prescribed limits should be discussed. Describe the facilities and analytical equipment that will be available to perform the analyses as well as the destination of laboratory wastes. Discuss provisions for obtaining samples during off-normal conditions to ensure that prescribed limits have not been violated.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	6	SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT	14.4	By reference to Chapter 3, "Principal Design Criteria," provide the primary design bases and supporting analyses for demonstrating that all radioactive waste materials that are generated as a result of ISFSI or MRS operations will be safely contained until disposal. The considerations for offsite disposal of solid waste materials and contaminated equipment should be included. The waste confinement objectives, equipment, and program should implement, in part, the considerations necessary for protection against radiation, as described in Chapter 7, "Radiation Protection." All reference to waste in this chapter is to waste that is generated as a result of the ISFSI or MRS operation. It does not refer to the high-level radioactive wastes that may be stored in the MRS.
Licensing, Operation, Design	6.1	SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT **Onsite Waste Sources**	14.4.1	Classify all anticipated radioactive wastes with respect to source, chemical and radiological composition, method and design for treatment and handling, and mode of storage prior to disposal. Previous flowsheets and diagrams may be cross-referenced. Waste sources other than those containing radioactive materials should also be identified if they constitute a potential safety problem. Account for combustion products as well as chemical wastes leaving the installation. This information should be included to assist the NRC staff in ascertaining that no radioactive material will be added to such sources, particularly effluents.
Licensing, Operation, Design	6.2	SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT **Offgas Treatment and Ventilation**	14.4.2	For all offgas and ventilation systems, indicate those radioactive wastes that will be produced as a result of their removal from the gases cleaned by those systems. Such items as filters and scrubbers, which collect wastes, should be discussed to indicate the destination of the wastes upon regeneration or replacement. If the wastes enter other waste treatment systems, indicate how such transfers are made and any possible radiological effects of the transfer. The actual operation of the gas-cleaning equipment and its minimum expected performance should be discussed in this section.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	6.3	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention**	14.4.3	Show how all liquid wastes are generated and how they enter liquid treatment systems. Include such items as laboratory wastes, cask washdown, liquid spills, decontamination, and cleanup solutions. As part of the design objectives, a statement should be made concerning the inventory levels expected, provisions for interim storage, and identification of those streams that will be processed to achieve volume reduction or solidification. Relate the discussion on process and equipment to the radiation levels of the various types of wastes to be handled. A description of the solidification of liquid wastes should be provided.
Licensing, Operation, Design	6.3.1	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention** Design Objectives	14.4.3.1	Describe the design objectives for the system under discussion. Identify, in particular, criteria that incorporate backup and special features to ensure that the waste will be safely contained and personnel doses will be minimized.
Licensing, Operation, Design	6.3.2	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention** Equipment and System Description	14.4.3.2	Provide a description of the equipment and systems to be installed. Accompany the description with appropriate drawings adequate to show location of equipment, flow paths, piping, valves, instrumentation, and other physical features. Describe features, systems, or special handling techniques that are important to safety included in the systems to provide for the safety of the operation.
Licensing, Operation	6.3.3	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention** Operating Procedures	14.4.3.3	Provide a narrative description of the procedures associated with operation of the systems. State whether the procedures will include performance tests, action levels, action to be taken under normal and off-normal conditions, and methods for testability to ensure functional operation.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	6.3.4	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention** Characteristics, Concentrations, and Volumes of Solidified Wastes	14.4.3.4	Describe the physical, chemical, and thermal characteristics of the solidified wastes, and provide an estimate of concentrations and volumes generated.
Licensing, Operation, Design	6.3.5	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention** Packaging	14.4.3.5	Describe the means for packaging the solidified wastes, and identify aspects that will be incorporated in the operating quality assurance program. The package itself should be described in detail to show (1) materials of construction, including welding information, (2) maximum temperatures for waste and container at the highest design heat loads, (3) homogeneity of the waste contents, (4) corrosive characteristics of the waste on the materials of construction, (5) means to prevent overpressurization of the package, and (6) confinement provided by the package under off-normal conditions.
Licensing, Operation, Design	6.3.6	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Liquid Waste Treatment and Retention** Storage Facilities	14.4.3.6	Describe the operation of the storage facilities demonstrating that the likelihood of accidental puncture or other damage to a package from natural phenomena or other causes is very low. Discuss external corrosion of the package from storage surroundings, if applicable. Show how packages will be moved safely into and out of storage locations and how the packages will be monitored over their storage life on site.
Licensing, Operation, Design	6.4	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes**	14.4.4	List and characterize all solid wastes that are produced during installation operation. Describe the systems used to treat, package, and contain these solid wastes.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	6.4.1	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes** Design Objectives	14.4.4.1	Describe the objectives of the methods and the equipment selected for minimizing the generation of solid wastes and for safe management of the solid waste that is generated.
Licensing, Operation, Design	6.4.2	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes** Equipment and System Description	14.4.4.2	Provide a description of the equipment and systems to be installed. Accompany the description with appropriate engineering drawings to show the location of the equipment and associated features that will be used for volume reduction, confinement and/or packaging, storage, and disposal.
Licensing, Operation, Design	6.4.3	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes** Operating Procedures	14.4.4.3	Describe the procedures associated with operation of the equipment, including performance tests, process limits, and means for monitoring and controlling to these limits.
Licensing, Operation, Design	6.4.4	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes** Characteristics, Concentrations, and Volumes of Solid Wastes	14.4.4.4	Describe the physical, chemical, and thermal characteristics of the solid wastes, and provide an estimate of concentrations and volumes generated.
Licensing, Operation, Design	6.4.5	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes** Packaging	14.4.4.5	Describe the means for packaging the solid wastes where required, and identify aspects that will be incorporated in the operating quality assurance program.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	6.4.6	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Solid Wastes** Storage Facilities	14.4.4.6	For solid wastes of the type to be retained on site for extended periods of time, show in detail the confinement methods used. Discuss corrosion aspects and monitoring of the confinement. Show how these wastes will be handled at the time the installation is permanently decommissioned.
Licensing, Operation, Design	6.5	**SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT** **Radiological Impact of Normal Operations - Summary**	14.4.5	For the gaseous and liquid effluents and solid wastes, provide the following: 1. A summary identifying each effluent and type of waste; 2. Amount generated per metric ton (MT) of spent fuel or high-level radioactive waste handled and stored per unit of time; 3. Quantity and concentration of each radionuclide in each stream; 4. Identification of the locations defined beyond the restricted areas [as in paragraph 20.3(a)(14) of 10 CFR Part 20] and beyond the controlled area* that are potentially impacted by radioactive materials in effluents; 5. For the locations identified in item 4, the amount of each radionu- clide and its person-rem contribution of radiation dose to human occupants that can accrue under normal operating conditions; 6. Discussion and sample calculations showing the reliability of the estimated values presented; and 7. For each effluent, the constraints imposed on process systems and equipment to ensure a safe operation. *"Controlled area" means that area immediately surrounding an ISFSI or MRS for which the licensee exercises authority over its use and within which ISFSI or MRS operations are performed (§ 72.3 of 10 CFR Part 72).

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7	**RADIATION PROTECTION**	11.4	This chapter of the SAR should provide information on methods for radiation protection and on estimated radiation exposures to operating personnel during normal operation and anticipated operational occurrences (including all types of radioactive material handling, transfer, processing, storage, and disposal; maintenance; routine operational surveillance; inservice inspection; and calibration). This chapter should also provide information on layout and equipment design, the planning and procedures programs, and the techniques and practices employed by the applicant in meeting the standards of 10 CFR Part 20 for protection against radiation and the guidance given in the appropriate regulatory guides. Reference to other chapters for information needed in this chapter should be specifically made where required.
Licensing, Operation, Design	7.1	**RADIATION PROTECTION** **Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)**	11.4.1	No text provided.
Licensing, Operation	7.1.1	**RADIATION PROTECTION** **Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)** Policy Considerations	11.4.1.1	Describe the management policy and organizational structure related to ensuring that occupational exposures to radiation and radiation-producing sources are ALARA. Describe the applicable activities to be conducted by the individuals having responsibility for radiation protection. Describe policy with respect to designing and operating the installation to achieve ALARA objectives. Indicate how the guidance given in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and, where appropriate, Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," will be followed. If this guidance will not be followed, indicate the specific alternative approaches to be used.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	7.1.2	**RADIATION PROTECTION** **Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)** Desgn Considerations	11.4.1.2	Describe layout and equipment design considerations that are directed toward ensuring that occupational radiation exposures are ALARA. Describe how experience from any past designs is used to develop improved design for ensuring that occupational radiation exposures are ALARA and that contamination incidents are minimized. Include any design guidance (both general and specific) given to the individual designers. Describe how the design is directed toward reducing the (1) need for maintenance of equipment, (2) radiation levels and time spent where maintenance is required, and (3) contamination control in handling, transfer, and storage of all radioactive materials. These descriptions, including an indication of how the applicable design consideration guidance provided in regulatory position 2 of Regulatory Guide 8.8 will be followed, should be provided. If it will not be followed, indicate the specific alternative approaches to be used. Also state whether, and if so how, relevant design experience from existing facilities is being used. Discuss the arrangements and plans for decontamination of the installation and individual items of equipment in case of need. Discuss how the ALARA goals are to be met and the alternatives considered with regard to occupational exposures to. radiation.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.1.3	**RADIATION PROTECTION** **Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)** Operational Considerations	11.4.1.3	Describe the methods used to develop the detailed plans and procedures for ensuring that occupational exposures to radiation are ALARA and that operational safeguards are provided to ensure that contamination levels are ALARA. Describe how these plans, procedures, and safeguards will impact on the design of the installation and how such planning has incorporated information from other designs and follows the applicable guidance given in regulatory position 4 of Regulatory Guide 8.8. If the guidance will not be followed, describe the specific alternative approaches to be used. Identify and describe procedures and methods of operation that are used to ensure that occupational radiation exposures are ALARA such as those pertinent procedures in regulatory position 4 of Regulatory Guide 8.8 and in Regulatory Guide 8.10. Describe how operational requirements are reflected in the design considerations described in Section 7.1.2 and the radiation protection design features described in Section 7.3. Provide the criteria and/or conditions under which various procedures and techniques are implemented for ensuring that occupational exposures to radiation are ALARA and residual contamination levels are ALARA for all systems that contain, collect, store, or transport radioactive solids and liquids generated as a result of the ISFSI or MRS operations, including those from the radioactive waste treatment, handling, and storage systems.
Licensing, Operation, Design	7.2	**RADIATION PROTECTION** **Radiation Sources**	7.4.1	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.2.1	**RADIATION PROTECTION** **Radiation Sources** Characterization of Sources	7.4.1	The sources of radiation that are the bases for the radiation protection design and the bases for their curie values should be described in the manner needed as input to the shielding design calculations. For shielding calculations, the description should include a tabulation of all sources by isotopic composition, X- and gamma-ray energy groups from zero to the maximum photon energy and the respective photon yield, and source geometry. In addition to the spent fuel or high-level radioactive waste in storage, the sources should include radioactive materials contained in equipment and storage containers or tanks throughout the installation. Indicate the physical and chemical forms of all sources.
Licensing, Operation, Design	7.2.2	**RADIATION PROTECTION** **Radiation Sources** Airborne Radioactive Material Sources	11.4.2.5	The sources of radioactive material that may become airborne in areas easily accessible to, or normally occupied by, operating personnel should be described with the provisions made for personnel protective measures. The description should include a tabulation of the calculated concentrations of airborne radioactive material by nuclides expected during normal operation and anticipated operational occurrences in areas normally occupied by operating personnel. Provide the models and parameters for calculating airborne concentrations of radioactive materials.
Licensing, Design	7.3	**RADIATION PROTECTION** **Radiation Protection Design Features**	11.4.2	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	7.3.1	**RADIATION PROTECTION** **Radiation Protection Design Features** Installation Design Features (Part 1 of 3)	11.4.2.1	Describe equipment and installation design features used for ensuring that occupational exposures to radiation are ALARA and that a high degree of integrity is obtained for the confinement of radioactive materials. Indicate how the applicable design feature guidance given in regulatory position 2 of Regulatory Guide 8.8 has been followed. If it was not followed, describe the specific alternative approaches used. Provide illustrative examples of the features used in the design as applied to the systems addressed in Section 7.1.3. An illustrative example should be provided for components of each of the following systems: shipping cask receiving, preparation, and transfer; cask decontamination and unloading; spent fuel and high-level radioactive waste transfer; spent fuel and high-level radioactive waste storage array servicing SSSC or drywell sealing; and site-generated waste treatment packaging, storage, and shipment. Reference other chapters and sections as appropriate. Provide scaled layout and arrangement drawings of the installation showing the locations of all sources described in Section 7.2. Include specific activity, physical and chemical characteristics, and expected concentrations. Provide on the layouts the radiation area designations, including area boundaries and type of interface (e.g., partitions, locked doors, barriers).

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	7.3.1	**RADIATION PROTECTION** **Radiation Protection Design Features** Installation Design Features (Part 2 of 3)	11.4.2.1	The layouts should show shield wall thicknesses, controlled access areas, personnel and equipment decontamination areas, contamination control areas and type of controls, traffic patterns, location of the health physics facilities, location of airborne radioactive material monitors and area radiation monitors, location of control panel(s) for radiological waste equipment and components, location of the onsite laboratory for analysis of chemical and radioactive samples, and location of the counting room. Provide the design radiation dose rate for each area and activity. Describe the facilities and equipment involved, including any special equipment provided specifically for radiation protection. Describe the function and performance objectives of the building ventilation systems. Discuss the areas and equipment serviced and the design for each unit system. Include in the description, by referring to drawings, the interface considerations between systems. Discuss the design limits selected for operation and the performance limits that must be met for safety. Discuss the program for measuring the efficiency of filters and other gaseous effluent treatment devices over the lifetime of the installation. Provide criteria for changing filters. Discuss how the ventilation system design will allow filter changes to be compatible with the ALARA principle.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	7.3.1	**RADIATION PROTECTION** **Radiation Protection Design Features** Installation Design Features (Part 3 of 3)	11.4.2.1	Estimate the concentrations and quantities of radioactive materials discharged by each system. List source terms by type of material, concentration, activity, and total quantity per unit time to be used in determining radiation exposure data presented in Section 7.4. Provide a detailed discussion of the evaluations made to show that unit ventilation systems by themselves and in conjunction with other ventilation systems will be operable. Show that sufficient margins exist so that a single component failure will not result in an uncontrolled release of radioactivity. Reference the discussions of offgas treatment in Section 4.3.1 and appropriate equipment and process flow drawings to show that: 1. ALARA radioactivity releases will be achieved during normal operation; 2. Capacity is sufficient to confine radioactive material during projected operating conditions; 3. Provisions are incorporated to adequately monitor performance; and 4. Satisfactory design features are incorporated to interface with other effluent and ventilation systems.
Licensing, Design	7.3.2	**RADIATION PROTECTION** **Radiation Protection Design Features** Shielding	11.4.2.3	Provide information on the shielding for each of the radiation sources identified in Section 7.2. Show the design of penetrations, the material, the method by which the shield parameters (e.g., attenuation coefficients, buildup factors) were determined, and the assumptions, codes, and techniques used in the calculations. Describe special protective features that use shielding, geometric arrangement (including equipment separation), or remote handling to ensure that occupational exposures to radiation will be ALARA in normally occupied areas. Describe the use of portable shielding, if applicable.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Design	7.3.3	**RADIATION PROTECTION** **Radiation Protection Design Features** Ventilation	11.4.2.4	The personnel protection features incorporated in the design of the ventilation systems should be described by amplifying the discussions on building ventilation and offgas treatment provided in Chapters 4, "Installation Design," and 5, "Operation Systems," to show that the designs selected will satisfy the ALARA provisions of paragraph 20.1(c) of 10 CFR Part 20 and of appropriate guides. The discussion should also show that expenditures for additional design work and equipment will not result in an accompanying reduction of released radioactive materials or personnel dose. Reference the discussion on building ventilation in Section 4.3.1 and appropriate engineering drawings to show the interrelationship of component parts and controls to the following: 1. Maintaining levels of exposure to radiation ALARA; 2. Preventing spread of radioactive materials and controlling contamination between areas; 3. Interfacing with process offgases (e.g., waste treatment, cask venting); and 4. Limiting the spread of radioactive materials within the ventilation systems.
Licensing, Design	7.3.4	**RADIATION PROTECTION** **Radiation Protection Design Features** Area Radiation and Airborne Radioactivity Monitoring Instrumentation	11.4.2.5	Describe the fixed area radiation monitors and continuous airborne monitoring instrumentation and the placement of each. Describe the criteria and methods used for determining setpoints for alarms from the radiological monitoring system. Provide information on the auxiliary and emergency power supply, range, sensitivity, accuracy, energy dependence calibration methods and frequency, alarm setpoints, recording devices, and location of detectors, readouts, and alarms for the monitoring instrumentation. Also provide the location of the continuous airborne monitor sample collectors, 6nd give details of sampling line pump location and for obtaining representative samples of effluent monitors.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.4	**RADIATION PROTECTION** **Estimated Onsite Collective Dose Assessment**	11.4.3.1	Provide the estimated annual occupancy times, including the maximum expected total hours per year for any individual and total person-hours per year for all personnel for each radiation area, including the storage areas, during normal operation and anticipated operational occurrences. For areas with expected airborne concentrations of radioactive material (as identified in Section 7.2.2), provide estimated maximum individual and total person-hours of occupancy. Also provide the objectives and criteria for design dose rates in various areas and an estimate of the annual collective person-rem doses associated with major functions such as spent fuel or high-level radioactive waste transfer and storage operations and ancillary activities (e.g., offgas handling, waste treatment), maintenance, radwaste handling, decontamination, and inservice inspection. Supply the bases, models, and assumptions for the above values. The estimated annual occupancy for each radiation area in the installation should be tabulated and the bases for the values provided. Provide estimates of annual collective doses (person-rems) for the functions listed above and the assumptions used in determining these values.
Licensing, Operation	7.5	**RADIATION PROTECTION** **Health Physics Program**	11.4.4	No text provided.
Licensing, Operation	7.5.1	**RADIATION PROTECTION** **Health Physics Program** Organization	11.4.4.1	Describe the administrative organization of the health physics program, including the authority and responsibility of each position identified. Indicate how the applicable guidance in regulatory position 2 of Regulatory Guide 8.8 and in Regulatory Guide 8.10 has been followed. If it was not followed, describe the specific alternative approaches used. Describe the experience and qualification of the personnel responsible for the health physics program.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.5.2	**RADIATION PROTECTION** **Health Physics Program** Equipment, Instrumentation, and Facilities	11.4.4.2	Describe portable and laboratory equipment and instrumentation for (1) performing radiation and contamination surveys, (2) sampling airborne radioactive material, (3) area radiation monitoring, ing and (4) personnel monitor- during normal operation, anticipated operational occurrences, and accident conditions. Describe the instrument storage, calibration, and maintenance facilities. Also describe the health physics facilities, laboratory facilities for radioactive material analyses, protective clothing, respiratory protective equipment, decontamination facilities (for equipment and personnel), and other contamination control equipment and areas that will be available. Indicate how the guidance provided by Regulatory Guides 8.4, "Direct-Reading and Indirect- Reading Pocket Dosimeters," and 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," will be followed. If it will not be followed, describe the specific alternative methods to be used. Describe the location of the respiratory protective equipment, protective clothing, and portable and laboratory equipment and instrumentation. Describe the type of detectors and monitors and the quantity, sensitivity, range, and frequency and methods of calibration for all the equipment and instrumentation mentioned above.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.5.3	**RADIATION PROTECTION** **Health Physics Program** Procedures	11.4.4.3	Describe the methods, frequencies, and plans for conducting radiation surveys. Describe the health physics plans that have been developed for ensuring that occupational radiation exposures will be ALARA. Describe the physical and administrative measures for controlling access and stay time for designated radiation areas. Reference may be made to Section 7.1, as appropriate. Describe the bases and methods for monitoring and controlling personnel, equipment, and surface contamination. Describe radiation protection training programs. Indicate how the guidance given in Regulatory Guides 8.9, 8.10, and 8.15, "Acceptable Programs for Respiratory Protection," will be followed. If it will not be followed, describe the specific alternative approaches to be used. Describe the methods and plans for personnel dosimetry, including methods for recording and reporting results. Describe how dosimetric results are used as a guide to operational planning. The criteria for performing routine and nonroutine whole-body and/or lung counting and bioassays should be provided. Describe the methods and procedures for evaluating and controlling potential airborne radioactive material concentrations, including any requirements for special air sampling. Discuss the use of respiratory protective devices, including the respiratory protective equipment fitting programs and training of personnel.
Licensing, Operation, Design	7.6	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment**	11.4.3.2	Describe the program and the analytical approach taken to monitor the radioactive material content of the effluent streams of the installation. Relate the monitoring program to process flow diagrams and the discussions presented in Chapter 5, "Operation Systems," and Chapter 6, "Site-Generated Waste Confinement and Management." An estimate of the contribution by the operations of the ISFSI or MRS to the offsite radiation level should be provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.6.1	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Effluent and Environmental Monitoring Program	9.4.3.2	The program for monitoring and estimating the contribution of radioactive materials to the environment should be described. Present the details of the approach, the results obtained for determining the background levels, and the estimate of subsequent contribution of the installation.
Licensing, Operation, Design	7.6.1.1	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Effluent and Environmental Monitoring Program *Gas Effluent Monitoring*	9.4.3.2	Describe the features of the monitoring systems to be used, their locations, and the release paths to be monitored. For each system, show the expected reliability and sensitivity. The selection of each system and instrument should be justified. The frequency of sampling, the limits for action, and the plans to be used to maintain continued integrity of analyses should also be discussed.
Licensing, Operation, Design	7.6.1.2	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Effluent and Environmental Monitoring Program *Liquid Effluent Monitoring*	9.4.3.2	Describe the features of the liquid monitoring systems to be used, their locations, and the items to be monitored. For each system, show the expected reliability and sensitivity. The selection of each system and instrument should be justified. Whenever sampling is used, the frequency of sampling, the limits for action, and the plans to be used to maintain continued integrity of analyses should also be discussed.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.6.1.3	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Effluent and Environmental Monitoring Program *Solid Waste Monitoring*	9.4.3.2	Describe the procedures, equipment, and instrumentation used to monitor all solid radioactive waste.
Licensing, Operation, Design	7.6.1.4	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Effluent and Environmental Monitoring Program *Environmental Monitoring*	9.4.3.2	Describe in detail the environmental monitoring program for those pathways that lead to the highest potential external and internal radiation exposures of individuals resulting from ISFSI or MRS operations. Provide a table showing the type of sample (e.g., water, soil, vegetable), number of samples, sample location, collection frequency, and sample analysis to be performed and its frequency. Identify the sampling locations on a map of suitable scale to show distance and direction of monitoring stations, with the site boundary also indicated on this map. This section should include the program for continuing meteorological data collection and evaluation to supplement the estimates previously developed.
Licensing, Operation, Design	7.6.2	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Analysis of Multiple Contribution	11.4.3	An analysis should be presented of the incremental collective doses (person-rems) that would result from the impact of present or projected nuclear facilities in the vicinity of the ISFSI or MRS (i.e., within an 8-kilometer (5-mile) radius) as compared with the collective doses from background for the same population.

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Licensing, Operation, Design	7.6.3	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Estimated Dose Equivalents	11.4.3	Present the annual whole-body collective doses (person-rem) estimated to be attributable to installation effluents in each of 16 compass sectors about the installation between each of the arcs having the radii of 1.5, 3, 5, 6.5, and 8 kilometers (approximately 1, 2, 3, 4, and 5 miles). Provide details of assumptions, and give sample calculations with emphasis on critical pathways to man. Relate to the meteorological data presented in Chapter 2, "Site Characteristics," and the radioactive material release rates in Chapter 6, "Site-Generated Waste confinement and Management." In addition to the person-rem whole-body determinations, details on uptakes by the critical organ should be provided.
Licensing, Operation, Design	7.6.3.1	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Estimated Dose Equivalents *Identification of Sources*	11.4.3	For each radioisotope that utes contrib- more than 10 percent of total dose, include a description of the char- acteristics of the isotope pertinent to its release and eventual biological impact.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.6.3.2	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Estimated Dose Equivalents *Analysis of Effects and Consequences* (Part 1 of 2)	11.4.3	An analysis of biological effects and the attendant risk factors should be supported by information that includes the following: 1. Joint frequency distribution of wind speed, wind direction, and atmospheric stability; 2. Methods, assumptions, and conditions employed; 3. Biological pathways and the critical organ; and 4. Dose models. The risk factors should be given for each isotope that contributes more than 10 percent of total dose and the critical organ in terms of maximum dose commit-ment (rem) per year, average dose commitment (rem) per year, and total collective dose (person-rem) per year for the population within an 8-kilometer (5-mile) radius. The considerations of uncertainties in the calculational methods and equipment performance should be discussed. Conservatism existing in assumptions should also be described. Reference published data associated with the analysis. The mathematical or physical model employed, including any simplification or approximation to perform the analyses, should be discussed. The parameters for postulated chronic releases should be tabulated. The tabulation should include conservative realistic values for each assumption used. List the parameters in a table similar to Table 7-1 (See Part 2 of 2) Any digital computer programs or analog simulation used in the analysis should also be identified. Adequate figures should be included on the analytical model, computer listing, and input data. Reference to computer models already available to the Commission may be made by summary only.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.6.3.2	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Estimated Dose Equivalents *Analysis of Effects and Consequences* (Part 2 of 2)	11.4.3	**TABLE 7-1** **Parameters To Be Tabulated* for Postulated Chronic Releases** Assumptions A. Data and Assumptions Used to Estimate Radioactive Source 1. Form (physical, chemical) 2. Particle size 3. Physical and chemical data related to transport or removal functions B. Data and Assumptions Used to Estimate 1. Leakage fractions 2. Absorption and filtration efficiencies 3. Release flow rates and pathways C. Dispersion Data 1. Stack or building leakage source 2. Building wake (ground source) 3. Boundary distances 4. x/Qs (annual average by sectors) 5. Deposition, decay, and washout coefficients D. Dose Data 1. Dose model (code) 2. Liquid and gaseous source terms 3. Biological pathways 4. Dose model (code) parameters and input used

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.6.4	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release	2.4.4.9	Describe radioactive liquid effluents. Refer to Chapter 6, "Site-Generated Waste Confinement and Management," for a discussion of how liquid wastes are treated. Describe the contribution that the liquid discharged to the atmosphere as water vapor makes to the gaseous radioactive source terms. Describe the radioactive and nonradioactive site-generated wastes from the following sources, and include the same type of information (as applicable) as described in Section 7.6.3.2.
Licensing, Operation, Design	7.6.4.1	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Treated Process Effluent (from Waste Treatment Are* a)	2.4.4.9	No text provided.
Licensing, Operation, Design	7.6.4.2	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Sewage*	2.4.4.9	No text provided.
Licensing, Operation, Design	7.6.4.3	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Drinking Water*	2.4.4.9	No text provided.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	7.6.4.4	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Rain Runoff*	2.4.4.9	No text provided.
Licensing, Operation, Design	7.6.4.5	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Laundry Waste*	2.4.4.9	No text provided.
Licensing, Operation, Design	7.6.4.6	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Items Requiring Further Development*	2.4.4.9	No text provided.
Licensing, Operation, Design	7.6.4.7	**RADIATION PROTECTION** **Estimated Offsite Collective Dose Assessment** Liquid Release *Changes Since Initial Submittal*	2.4.4.9	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8	**ACCIDENT ANALYSES**	15.4	The evaluation of the safety of an ISFSI or MRS is accomplished in part by analyzing the response of the installation to postulated accident events in terms of (1) minimizing the causes of such events, (2) the quantitative identification and mitigation of the consequences, and (3) the ability to cope with each situation if it occurs. These analyses are an important aspect of the reviews made by the NRC prior to issuing a license to store spent fuel in an ISFSI or to store spent fuel and high-level radioactive waste in an MRS. An in-depth discussion of accident analysis should be presented. This analysis should be updated to present details that have been revised or developed since the initial submittal. In previous chapters, features important to safety have been identified and discussed. The purpose of this chapter is to identify and analyze a range of credible accident occurrences (from minor to the design basis accidents) and their causes and consequences. For each situation, reference should be made to the appropriate chapter and section describing the considerations to prevent or mitigate the accident. ANSI/ANS 57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)," defines four categories of design events that provide a means of establishing design requirements to satisfy operational and safety criteria. The first design event is associated with normal operation. The second and third design events apply to events that are expected to occur during the life of the installation. The fourth design event is concerned with natural phenomena or low probability events.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.1	**ACCIDENT ANALYSES** **Off-Normal Operations**	15.4.1	In this section, design events of the first or second type as defined in ANSI/ANS 57.9-1984 are considered. They may include malfunctions of systems, minor leakage, limited loss of external power, or operator error. In general, the consequences of the events discussed in this section would not have a significant effect beyond the controlled area. The following format should be used to present the desired detail.
Licensing, Operation, Design	8.1.1	**ACCIDENT ANALYSES** **Off-Normal Operations** Event	15.4.1	Identify the event, including the location of event, type of failure or maloperation, and system or systems involved.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.1.1.1	**ACCIDENT ANALYSES** **Off-Normal Operations** Event *Postulated Cause of Event*	15.4.1	Describe the sequence of occurrences that could initiate the event under consideration and the bases upon which credibility or probability of each occurrence in the sequence is determined. The following should be provided: 1. Starting conditions and assumptions; 2. A step-by-step sequence of the course of each event, identifying all protection systems required to function at each step; and 3. Identification of any operator actions necessary. The discussion should show the extent to which protective systems should function, the effect of failure of protective functions, and the credit taken for operation safety features. The performance of backup protection systems during the entire course of the event should be analyzed. The discussion also should include credit taken for the functioning of other systems and consequences of failure. The analysis given should permit an independent evaluation of the adequacy of the protection system as related to the event under study. The results can be used to determine which functions, systems, interlocks, and controls are important to safety and what actions are required by the operator under anticipated operational occurrence and off-normal conditions.
Licensing, Operation, Design	8.1.1.2	**ACCIDENT ANALYSES** **Off-Normal Operations** Event *Detection of Event*	15.4.1	Discuss the means or methods such as visual or audible alarms or routine inspections performed on a stated frequency to be provided to detect the event. Provide for each an assessment of response time.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.1.1.3	**ACCIDENT ANALYSES** **Off-Normal Operations** Event *Analysis of Effects and Consequences*	15.4.1	Analyze the effects and particularly any radiological consequences of the event. The analysis should: 1. Show the methods, assumptions, and conditions used in estimating the course of events and the consequences; 2. Identify the time-dependent characteristics and release rate of radioactive materials within the confinement system that could escape to the environment; and 3. Describe the margin of protection provided by whatever system is depended on to limit the extent or magnitude of the consequences.
Licensing, Operation, Design	8.1.1.4	**ACCIDENT ANALYSES** **Off-Normal Operations** Event *Corrective Actions*	15.4.1	For each event, give the corrective actions necessary to return to a normal situation.
Licensing, Operation, Design	8.1.2	**ACCIDENT ANALYSES** **Off-Normal Operations** Radiological Impact from Off-Normal Operations	15.4.1	The capability of the installation to operate safely within the range of anticipated operating variations, malfunctions of operating equipment, and operator error should be shown. The information may be presented in tabular form with the situations analyzed listed in one column accompanied by other columns that identify: 1. Estimated doses (person-rem); 2. Method or means available for detecting the respective situations; 3. Causes of the particular situation; 4. Corrective actions; and 5. Effects and consequences.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.2	**ACCIDENT ANALYSES** **Accidents**	15.4.2	Provide a rigorous analysis of accident potential for the proposed ISFSI or MRS. Include any incident that would potentially result in a dose of >25 mrem beyond the controlled area. If there are no such credible potential accidents, show that this is true. Such analyses should address situations wherein direct radiation or radioactive materials may be released in such quantity as to endanger personnel within the controlled area. Design events of the third and fourth types as defined in ANSI/ANS 57.9-1984 are included in this section. The following (subsections') format should be used to provide the desired detail.
Licensing, Operation, Design	8.2.1	**ACCIDENT ANALYSES** **Accidents** Accidents Analyzed	15.4.2	Identify the accident, the location or portion of the facility involved, and the type of accident. Discuss each accident sequentially (e.g., 8.2.2, 8.2.3 ...).
Licensing, Operation, Design	8.2.1.1	**ACCIDENT ANALYSES** **Accidents** Accidents Analyzed *Cause of Accident*	15.4.2	For each accident analyzed, describe and list the sequence of events leading to the initiation of the accident. Identify, with respect to natural phenomena, human error, equipment malfunction, or equipment failure. Include an estimate of probability, and show how this probability estimate was determined.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.2.1.2	**ACCIDENT ANALYSES** **Accidents** Accidents Analyzed *Accident Analysis* (Part 1 of 2)	15.4.2	Analyze the effects and particularly any radiological consequences of each accident. Show the methods, assumptions, and conditions used in estimating the consequences, the recovery from the consequences, and the steps used to mitigate each accident. Assess the consequences of the accident to persons and property both on site and off site. In addition to the assumptions and conditions employed in the course of events and consequences, support the following by sufficient information: 1. The mathematical or physical models employed, including a description of any simplification introduced to perform the analyses. Identify assumptions used that are known to differ from those used by the NRC staff. 2. Identification of any digital computer program or analog simulation used in the analysis with principal emphasis on the input data and the extent or range of variables investigated. This information should include figures showing the analytical models, flow path identification, actual computer listing, and complete listing of input data. The detailed description of mathematical models and digital computer programs or listings may be included by reference with only summaries provided in the SAR. 3. The physical or mathematical models used in the analyses and the bases for their use with specific reference to: a. The distribution and fractions of the radioactive material inventory assumed to be released from the source into offgas systems; b. The concentrations of airborne radioactive materials in the confinement atmosphere and buildup on filters during the postaccident time intervals analyzed; and c. The conditions of meteorology, topography, or other circumstances, and combinations of adverse conditions considered in the analyses.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.2.1.2	**ACCIDENT ANALYSES** **Accidents** Accidents Analyzed *Accident Analysis* (Part 2 of 2)	15.4.2	4. The time-dependent characteristics, activity, and release transmissible rate of radioactive materials within the confinement system that could escape to the environment via leakages in the confinement boundaries and leakage through lines that could exhaust to the environment. 5. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects that should be taken into account in the evaluation of the results. 6. The conditions and assumptions associated with the events analyzed, including any reference to published data or research and development investigations in substantiation of the assumed or calculated conditions. 7. The extent of system interdependency (confinement system and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the confinement systems or other sources such as the contribution of confinement air systems and air purification and cleanup systems. 8. The results and consequences derived from each analysis and the margin of protection provided by whatever system is depended on to limit the extent or magnitude of the consequences.
Licensing, Operation, Design	8.2.1.3	**ACCIDENT ANALYSES** **Accidents** Accidents Analyzed *Accident Dose Calculations*	15.4.2	1. For each accident analyzed, provide and discuss the results of conservative calculations of potential integrated whole-body and critical-organ doses to an individual from exposure to radiation as a function of distance and time after the accident. Present in terms of a 50-year dose commitment. Discuss the results and consequences derived from the analysis and the margin of protection provided by whatever system is depended on (i.e., remains operative) to limit the extent or magnitude of the consequences. 2. For each accident analyzed, provide and discuss the results of con- servative calculations of potential integrated whole-body and critical-organ integrated population doses from exposure to radiation as a function of population distribution at the time of initial operation to a distance of 8 kilo-meters (5 miles). Present results in terms of a 50-year dose commitment.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	8.3	**ACCIDENT ANALYSES** **Site Characteristics Affecting Safety Analysis**	2.4	Describe in summary form the site characteristics that have a bearing on the safety analysis, and show how these have been considered in developing suitable margins of safety.
Licensing, Operation	9	**CONDUCT OF OPERATIONS**	10.4	The plan for operation of the installation should be described. Sufficient detail should be provided to indicate how the applicant intends to conduct all operations to ensure that a technically competent staff will be maintained to provide continued implementation of administrative and operating procedures and programs, all of which are considered necessary to ensure.safe operation.
Licensing, Operation	9.1	**CONDUCT OF OPERATIONS** **Organizational Structure**	10.4.1	The following format should be used to present the organizational structure through the construction phase and through the preoperational testing, startup, installation operation, and decommissioning phases of the project.
Licensing, Operation	9.1.1	**CONDUCT OF OPERATIONS** **Organizational Structure** Corporate Organization	10.4.1.1	Describe the corporate arrangement or organization responsible for the spent fuel or high-level radioactive waste storage installation. If the corporation is made up from two or more existing identities, the relationship and responsibilities between each should be explained. Provide sufficient information to demonstrate the financial capabilities for construction, operation, and decommissioning of the installation.
Licensing, Operation	9.1.1.1	**CONDUCT OF OPERATIONS** **Organizational Structure** Corporate Organization *Corporate Functions, Responsibilities, and Authorities*	10.4.1.1	Describe corporate functions, responsibilities, and authorities with respect to installation engineering and design, construction, quality assurance, testing, operation, and other applicable activities.

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Licensing, Operation	9.1.1.2	**CONDUCT OF OPERATIONS** **Organizational Structure** Corporate Organization *Applicant's In-House Organization.*	10.4.1.1	A description should be provided of the applicant's management and technical staffing and in-house organizational relationships established for the design and construction review and quality assurance functions and of the responsibilities and authorities of personnel and organizations described in Section 9.1.1.1. Establish the extent of dependence on offsite personnel.
Licensing, Operation	9.1.1.3	**CONDUCT OF OPERATIONS** **Organizational Structure** Corporate Organization *Interrelationships with Contractors and Suppliers*	10.4.1.1	The working interrelationships and organizational interfaces among the applicant, the architect-engineer, and other suppliers and contractors should be described.
Licensing, Operation	9.1.1.4	**CONDUCT OF OPERATIONS** **Organizational Structure** Corporate Organization *Applicant's Technical Staff*	10.4.1.1	Describe the applicant's technical staff specifically supporting the engineering, construction, and operation of the ISFSI or MRS. Include a description of the duties, responsibilities, and authority of the engineering technical staff; and state numbers of personnel, qualifications, educational backgrounds (disciplines), and technical experience. Indicate technical support for the applicant's technical staff to be provided by outside consultants. If such arrangements are to be used, the specific areas Qf responsibility and functional working arrangements of these support groups should be provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	9.1.2	**CONDUCT OF OPERATIONS** **Organizational Structure** Operating Organiza-tion, Management, and Administrative Control System	10.4.1.2	This section should describe the structure, functions, and responsibilities of the operating organization. The following specific information should be included items outlined in subsections 9.1.2.1 and 9.1.2.2:
Licensing, Operation	9.1.2.1	**CONDUCT OF OPERATIONS** **Organizational Structure** Operating Organiza-tion, Management, and Administrative Control System *Onsite Organization*	10.4.1.2	Provide a comprehensive description of the organizational arrangement of the facility showing the title of each position, the flow of responsibility as depicted by an organizational chart, and the number of personnel in each unit. Describe the organizational arrangement for ensuring safe operation, the mode of operation, and assigned responsibilities.
Licensing, Operation	9.1.2.2	**CONDUCT OF OPERATIONS** **Organizational Structure** Operating Organiza-tion, Management, and Administrative Control System *Personnel Functions, Responsibilities, and Authorities*	10.4.1.2	Describe the functions, responsibilities, and authorities of major personnel positions, including a discussion of specific succession to responsibility for overall operation of the facility in the event of absences, incapacitation, or other emergencies.

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Licensing, Operation	9.1.3	**CONDUCT OF OPERATIONS** **Organizational Structure** Personnel Qualification Requirements	10.4.4.2	Describe the proposed minimum qualification requirements for onsite personnel and the qualifications of available supporting personnel. The following specific information should be included items outlined in subsections 9.1.3.1 and 9.1.3.2 :
Licensing, Operation	9.1.3.1	**CONDUCT OF OPERATIONS** **Organizational Structure** Personnel Qualification Requirements *Minimum Qualification Requirements*	10.4.4.2	The minimum qualification requirements should be stated for major operating, technical, and maintenance supervisory personnel.
Licensing, Operation	9.1.3.2	**CONDUCT OF OPERATIONS** **Organizational Structure** Personnel Qualification Requirements *Qualifications of Personnel*	10.4.4.2	The qualifications of the persons assigned to the managerial and technical positions described should be presented in resume' form. The resumes should identify persons by position title and, as a minimum, should describe the formal education, training, and pertinent experience of the persons.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	9.1.4	**CONDUCT OF OPERATIONS** **Organizational Structure** Liaison with Outside Organizations	10.4.1.1	Discuss arrangements made with outside organizations, including those providing expertise on technical facets of details concerning site selection and evaluation, installation design and construction, process and equipment selection or development, and safety evaluations. Additionally, any arrangements made with other government agencies should be presented. The method or system used to monitor the interfaces between each participant should be included.
Licensing, Operation	9.2	**CONDUCT OF OPERATIONS** **Preoperational Testing and Operation**	10.4.2	Describe the preoperational testing and operating startup plans. Emphasize those plans demonstrating that the layout, equipment, and planned operations meet safety and design criteria discussed in previous chapters. Test plans should be presented to verify the integrity of the structures and equipment and to substantiate the safety analysis. Results obtained from carrying out the planned tests are to be reported as a supplement to the SAR.
Licensing, Operation	9.2.1	**CONDUCT OF OPERATIONS** **Preoperational Testing and Operation** Administrative Procedures for Conducting Test Program	10.4.2	Describe the system used for (1) preparing, reviewing, approving, and executing all testing procedures and instructions and (2) evaluating, documenting, and approving the test results, including the organizational responsibilities and personnel qualifications of the applicant and his contractors. Describe the administrative procedures for incorporating any needed system modifications or procedure changes, based on the results of the tests (e.g., test procedure inadequacies or test results contrary to expected test results).
Licensing, Operation	9.2.2	**CONDUCT OF OPERATIONS** **Preoperational Testing and Operation** Test Program Description	10.4.2.1	Describe the test objectives and the general methods for accomplishing these objectives, the acceptance criteria that wi11 be used to evaluate the test results, and the general prerequisites for performing the tests, including special conditions to simulate normal and off-normal operating conditions of the tests listed.

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Licensing, Operation	9.2.2.1	**CONDUCT OF OPERATIONS** **Preoperational Testing and Operation** Test Program Description *Physical Facilities*	10.4.2.1	For the physical facilities, components, and equipment, identify the items to be tested, type of test, response, and validation.
Licensing, Operation	9.2.2.2	**CONDUCT OF OPERATIONS** **Preoperational Testing and Operation** Test Program Description *Operations*	10.4.2.1	Identify those operations to be tested, type of test, response, and validation.
Licensing, Operation	9.2.3	**CONDUCT OF OPERATIONS** **Preoperational Testing and Operation** Test Discussion	10.4.2.1	For each preoperational test, provide the following information: 1. Describe the purpose of the test. 2. Define the response expected in terms of design bases and criteria discussed in previous chapters, and indicate the margin of difference acceptable for safe operation. 3. Discuss necessary corrective action if the results of the preoperational test do not confirm the expected response.
Licensing, Operation	9.3	**CONDUCT OF OPERATIONS** **Training Programs**	10.4.4	No text provided.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	9.3.1	**CONDUCT OF OPERATIONS** **Training Programs** Program Description	10.4.4	Describe the proposed training program, including the scope of training in (1) installation operations and design, instrumentation and control, methods of dealing with operating malfunctions, decontamination procedures, and emergency procedures and (2) health physics subjects such as nature and sources of radiation, methods of controlling contamination, interactions of radiation with matter, biological effects of radiation, use of monitoring equipment, and principles of criticality hazards control. Identify personnel classification with level of instruction.
Licensing, Operation	9.3.2	**CONDUCT OF OPERATIONS** **Training Programs** Retraining Program	10.4.4	Describe the program for continued training that provides additional materials and refresher training.
Licensing, Operation	9.3.3	**CONDUCT OF OPERATIONS** **Training Programs** Admintstration and Records	10.4.4.1	Identify personnel in the organization responsible for the training programs and for maintaining up-to-date records on the status of trained personnel, training of new employees, and refresher or upgrading training of present personnel.
Licensing, Operation	9.4	**CONDUCT OF OPERATIONS** **Normal Operations**	10.4.3	No text provided.
Licensing, Operation	9.4.1	**CONDUCT OF OPERATIONS** **Normal Operations** Procedures	10.4.3.1	The applicant should make a commitment to conduct operations that are important to safety in accordance with detailed written procedures. Include a list of procedures that, by title or subject, clearly indicates their purpose and applicability. Also include a description of the review, change, and approval practices for all installation operating, maintenance, and testing procedures.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	9.4.2	**CONDUCT OF OPERATIONS** **Normal Operations** Records	10.4.3.2	Present the detailed management system for maintaining records relating to the historical operation of the installation. This system should include quality assurance records; operating records, including principal maintenance, alterations, or additions made; records of off-normal occurrences and events associated with radioactive releases; environmental survey records; and the identity and pertinent information of the spent fuel or high-level radioactive waste stored.
Licensing, Operation	9.5	**CONDUCT OF OPERATIONS** **Emergency Planning**	10.4.5	Describe plans for coping with emergencies. Refer to § 72.32* of 10 CFR Part 72 for a description of the kind of information to be provided and the minimum information to be included in the emergency plan. *The text of this section will be promulgated in final form as a conforming amendment. It will be based on the Emergency Preparedness rule (54 FR 14051).
Licensing, Decommissioning	9.6	**CONDUCT OF OPERATIONS** **Decommissioning Plan**	13.4.3	Describe plans for decommissioning to ensure that at the end of the facility's useful life decommissioning will be carried out in a safe and efficient manner. Refer to § 72.30 of 10 CFR Part 72 for the kind of information that should be provided on the decommissioning method that has been tentatively selected, the plans for facilitating the decommissioning process, and recordkeeping. Show how this plan has been used in designing the installation. The plan should be in sufficient detail to provide the basis for an estimate of the decommissioning costs. Such cost estimates are to be used in conjunction with financial qualification requirements to provide reasonable assurance for obtaining funds for decommissioning.
Licensing, Decommissioning, Design	9.6.1	**CONDUCT OF OPERATIONS** **Decommissioning Plan** Decommissioning Program	13.4.3	Present a tentative selection and description of the planned program for decommissioning the installation, based on the design provisions for decommissioning and the present state of the art. Indicate the basis used in selecting the program to-be used such as costs, radiation safety, or other considerations.

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Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Decommissioning, Design	9.6.2	**CONDUCT OF OPERATIONS** **Decommissioning Plan** Cost of Decommissioning	13.4.3	Based on the assumed decommissioning program, identify the approximate cost of the decommissioning activity. This estimate should be used in conjunction with the financial qualification requirement to indicate that there is reasonable assurance that decommissioning funds will be provided.
Licensing, Decommissioning, Design	9.6.3	**CONDUCT OF OPERATIONS** **Decommissioning Plan** Decommissioning Facilitation	13.4.1, 13.4.2	Describe installation design and operational features that are intended to facilitate decommissioning by reducing health and safety impacts of decommissioning and reducing the volume of radioactive wastes.
Licensing, Decommissioning	9.6.4	**CONDUCT OF OPERATIONS** **Decommissioning Plan** Recordkeeping for Decommissioning	13.4.3	Describe plans to obtain and safeguard records and archive files that will support decommissioning.
Licensing, Decommissioning	9.7	**CONDUCT OF OPERATIONS** **Physical Security and Safeguards and Contingency Plans**	10.4.6	Physical security and safeguards contingency plans should be submitted as separate documents. Subpart H of 10 CFR Part 72 sets forth requirements for the content of these plans.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	10	**OPERATING CONTROLS AND LIMITS**	16.4	Throughout the previous sections of this guide, the need to identify safety limits, limiting conditions, and surveillance requirements has been indicated. It is from such information that the operating controls, limits, and supporting bases should be developed. The operating controls and limits for spent fuel storage in an ISFSI or spent fuel and high-level radioactive waste storage in an MRS are derived from the safety assessment of the installation and include all important safety, environmental, and materials and plant protection aspects of installation operation. The safety and environmental analyses should support the conclusion that the health and safety of the public and operating personnel and the environmental values will be protected during ISFSI or MRS operation if all operations are performed within certain prescribed limits. These limits are defined and established in the operating controls and limits. Except for changes that involve license conditions or safety questions that have not been reviewed, changes can be made without amending the license. Changes in operating controls and limits would require NRC staff review and approval before being instituted. The operating controls and limits should be proposed by the applicant. These are reviewed and issued by the NRC in the form of license conditions, including technical specifications.

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	10.1	**OPERATING CONTROLS AND LIMITS** **Proposed Operating Controls and Limits**	16.4	Identify and justify the selection of those variable conditions or other items based on the design criteria of the installation or determined, as a result of safety assessment and evaluation, to be probable subjects of operating controls and limits for the installation. The operating controls and limits and bases proposed by an applicant should be included in this Chapter 10 of the SAR. The operating controls and limits should be complete; i.e., to the fullest extent possible, numerical values and other pertinent data should be provided, including the technical and operating conditions supporting the selection. For each control or limit, the applicable sections that develop, through analysis and evaluation, the details and bases for the control or limit should be referenced. Each license to store spent fuel in an ISFSI or to store spent fuel and high-level radioactive waste in an MRS issued by the NRC will contain technical operating limits, conditions, and requirements imposed on the conduct of operations in the interest of the health and safety of the public. The operating controls and limits are proposed by the applicant. A statement of the bases or reasons for proposed controls or limits should be included in the SAR. After review by the NRC staff, they are modified as necessary before becoming part of the license.
Licensing, Operation, Design	10.1.1	**OPERATING CONTROLS AND LIMITS** **Proposed Operating Controls and Limits** Content of Operating Controls and Limits	16.4	Operating controls and limits should include both technical and administrative matters. Operating controls and limits related to technical matters should consist of-those features of the installation that are of controlling importance to safety (operating variables, systems, or components). In addition, operating controls and limits related to technical matters should include effluent and environmental monitoring and controls or limits addressed to the attainment of ALARA levels of releases and exposures. Operating controls and limits related to administrative matters should be addressed to those organizational and functional requirements that are important to the achievement and maintenance of safe operation of the installation.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	10.1.2	**OPERATING CONTROLS AND LIMITS** **Proposed Operating Controls and Limits** Bases for Operating Controls and Limits	16.4	When an operating control and limit has been selected, the bases for its selection and its significance to safety of operation should be defined. This can be done by the provision of a summary statement of the, technical and operational considerations justifying the selection. The SAR should fully develop, through analysis and evaluation, the details of these bases. Therefore, the physical format for operating controls and limits assumes importance since the collection of controls or limits and their written bases form a document that delineates those features and actions important to safety of operation, the reasons for their importance, and their relationships to each other.
Licensing, Operation, Design	10.2	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits**	16.4	Refer to § 72.44, "License Conditions," of 10 CFR Part 72 for guidance on the categories of activities and conditions requiring operating controls and limits. Additional categories may be designated by the applicant or the NRC if deemed necessary to ensure the protection of the environment or public health and safety.
Licensing, Operation, Design	10.2.1	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Functional and Oper-ating Limits, Moni-toring Instruments, and Limiting Control Settings	16.4	Controls or limits of this category apply to operating variables important to safety that are observable and measurable (e.g., temperatures within the storage structure or evidence of confinement leakage). Control of such variables is directly related to the performance and integrity of equipment and confinement barriers.
Licensing, Operation, Design	10.2.2	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Limiting Conditions for Operation	16.4	This category of operating controls and limits covers two general classes: (1) equipment and (2) technical conditions and characteristics of the installation necessary for continued operation, as discussed below.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation, Design	10.2.2.1	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Limiting Conditions for Operation *Equipment*	16.4	Operating controls and limits should establish the lowest acceptable level of performance for a system or component and the minimum number of components or the minimum portion of the system that should be operable or available.
Licensing, Operation, Design	10.2.2.2	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Limiting Conditions for Operation *Technical Conditions and Characteristics*	16.4	Technical conditions and characteristics should be stated in terms of allowable quantities, e.g., storage structure-temperatures; radioactivity levels in gas samples; area radiation levels; or allowable configurations of equipment, spent fuel assemblies, and high-level radioactive waste canisters during transfer operations.
Licensing, Operation, Design	10.2.3	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Surveillance Requirements	16.4	Major emphasis in surveillance specifications should be placed on those systems and components essential to safety during all modes of operation or necessary to prevent or mitigate the consequences of accidents. Tests, calibrations, or inspections should verify performance and availability of important equipment and should detect incipient deficiencies.
Licensing, Operation, Design	10.2.4	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Design Features	16.4	These operating controls and limits cover design characteristics of special importance to each of the physical barriers and to the maintenance of safety margins in the design. The principal objective of this category is to control changes in the design of essential equipment.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	10.2.5	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Administrative Controls	10.4.1.3, 16.4	The SAR should contain a full description and discussion of organization and administrative systems and procedures, recordkeeping, review and audit, and the reporting necessary to ensure that the operations involved in the storage of spent fuel or high-level radioactive waste are performed in a safe manner.
Licensing, Operation	10.2.6	**OPERATING CONTROLS AND LIMITS** **Development of Operating Controls and Limits** Suggested Format for Operating Controls and Limits (Part 1 of 2)	16.4	1. Title: (e.g., maximum radiation level at any surface of a storage structure). 2. Specification: (limits). 3. Applicability: System(s) or operations to which the control or limit applies should be clearly defined. 4. Objective: The reason(s) for the control or limit and the specific unsafe condition(s) it is intended to prevent. 5. Action: What is to be done if the control or limit is exceeded; clearly define specific actions. 6. Surveillance Requirements: What maintenance and tests are to be performed and when? 7. Bases: The SAR should contain all pertinent information and an explicit detailed analysis and assessment supporting the choice of the item and its specific value or characteristics. The basis for each control or limit should contain a summary of the information in sufficient depth to indicate thecompleteness and validity of the supporting information and to provide justification for the-control or limit.

Category	REG 3.48 Section	REG Guide 3.48 <u>SECTION TITLE</u> Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Operation	10.2.6	**<u>OPERATING CONTROLS AND LIMITS</u>** **Development of Operating Controls and Limits** Suggested Format for Operating Controls and Limits (Part 2 of 2)	16.4	The following subjects may be appropriate for discussion in the bases section: a. *Technical Basis* . The technical basis is derived from technical knowledge of the process and its characteristics and should support the choice of the particular variable as well as the value of the variable. The results of computations, experiments, or judgments should be stated, and analysis and evaluation should be summarized. b. *Equipment* . A safety limit often is protected by or closely related to certain equipment. Such a relationship should be noted, and the means by which the variable is monitored and controlled should be stated. For controls or limits in categories referenced in Sections 10.2.2 through 10.2.4, the bases are particularly important. The function of the equipment and how and why the requirement is selected should be noted here. In addition, the means by which surveillance is accomplished should be noted. If surveillance is required periodically, the basis for frequency of required action should be given. c. *Operation* . The margins and the bases that relate to the safety limit(s) and the normal operating zone(s) should be stated. The roles of operating procedures and of protective systems in guarding against exceeding a limit or condition should be stated. Include a brief discussion of such factors as system response(s), process or operational transients, malfunctions, and procedural errors. Reference to related controls or limits should be made.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Quality Assurance	11	**QUALITY ASSURANCE** (Part 1 of 2)	12.4	Section 72.140(c) of 10 CFR Part 72 requires the applicant to provide a description of its quality assurance (QA) program based on the criteria in Subpart G of 10 CFR Part 72. The application of the QA program to identified activities, including operations, and to identified structures, systems, and components must be commensurate to the importance to safety of such identified activities and items. The program should cover all activities identified as being important to safety throughout the life of the project, from site selection and preliminary design through final decommissioning. National standard ANSI/ASME NQA-1-1983, "Quality Assurance Program Requirements for Nuclear Power Facilities," is specifically applicable to an ISFSI or MRS. The organization of this standard is consistent with the presentation of the 18 criteria in Subpart G of 10 CFR Part 72.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Licensing, Quality Assurance	11	**QUALITY ASSURANCE** (Part 2 of 2)	12.4	This chapter on QA should be similarly organized. The 18 criteria are as follows: 1. Organization 2. Quality assurance program 3. Design control 4. Procurement document control 5. Instructions, procedures, and drawings 6. Document control 7. Control of purchased material, equipment, and services 8. Identification and control of materials, parts, and components 9. Control of special processes 10. Licensee inspection 11. Test control 12. Control of measuring and test equipment 13. Handling, storage, and shipping control 14. Inspection, test, and operating status 15. Nonconforming materials, parts, or components 16. Corrective action 17. Quality assurance records 18. Audits Note that the Basic and Supplemental Requirements in ANSI/ASME NQA-1-1983 reflect the regulatory requirements. The guidance material presented in the appendices is optional. However, an applicant should follow such guidance where applicable with any deviations fully explained and justified.

Categories:
Construction
Decommissioning
Design
General
Licensing

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.48

Category	REG 3.48 Section	REG Guide 3.48 SECTION TITLE Subsection Title	NUREG 1567 Section	Criteria Description
Operaton				
Quality Assurance				

Category	REG 3.60 Section	REG Guide 3.60 SECTION TITLE	Criteria Description
Design	A	**REGULATORY POSITION**	Subpart F, "General Design Criteria," of 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," presents the general design criteria that are applicable to an ISFSI. This regulatory guide provides guidance acceptable to the NRC staff for use in the design of a dry storage ISFSI that will comply with these general design criteria. Any information collection activities mentioned in this regulatory guide are contained in requirements in 10 CFR Part 72, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 72 have been cleared under OMB Clearance No. 3150-0132.
Design	B	**DISCUSSION**	Group 57.7 of Subcommittee ANS-55 (Fuel and Waste Management) of the American Nuclear Society has developed ANSI/ANS-57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)." It was approved as an American National Standard in December 1984, and it defines the design criteria for a dry-type independent spent fuel storage installation.
Design	C	**REGULATORY POSITION**	ANSI/ANS-57.9-1984 is acceptable to the NRC staff for use in the design of an ISFSI that uses a dry environment as the mode of storage subject to the items described in subsections C.1 through C.9.
Design	C.1	**REGULATORY POSITION** **No Title for subsection 1.**	ANSI/ANS-57.9-1984 refers to a companion standard for design input of siting parameters, ANSI/ANS-2.19-1981, "Guidelines for Establishing Site-Related Parameters for Site Selection and Design of an Independent Spent Fuel Storage Installation (Water-Pool Type)," which has not been endorsed by the NRC. Until ANSI/ANS-2.19-1981 is endorsed by the NRC, the users of ANSI/ANS-57.9-1984 should seek guidance from the NRC staff on siting parameters that are used as design input.
Design	C.2	**REGULATORY POSITION** **No Title for subsection 2.**	Section 2.8 of ANSI/ANS-57.9-1984 (ANSI/ANS-57.9-1984 does not actually have a Section 2.8. Section 2 is Definitions.) defines a new term, "important confinement features," to classify structures, systems, and components with regard to the degree of their importance to safety. An existing term, "important to safety," is presently defined in 10 CFR Part 72 and is applicable for use. The term "important to safety" should be used instead of "important confinement features" as used in the standard.

Design	C.3	**REGULATORY POSITION** **No Title for subsection 3.**	Section 5.4.5.1 requires that engineered foundations take into account the potential for soil liquefaction during Design Event IV occurrences. This section should not be applied to portable storage system modules (i.e., metal storage casks) that can continue to provide confinement independently of storage foundation failure.
Design	C.4	**REGULATORY POSITION** **No Title for subsection 4.**	Section 5.4.6.1.1 should be supplemented to ensure that the storage system module will either withstand the maximum credible drop without compromising the integrity of the shielding structure or provide the capability for unloading the module. This section should be supplemented with the following: "If the storage system module is portable, provide the capability for withstanding the maximum credible drop during transport without compromising the integrity of the shielding structure or provide the capability for unloading an individual storage module for Design Event III."
Design	C.5	**REGULATORY POSITION** **No Title for subsection 5.**	Section 6.2.2.1.2(2) indicates that, in the design analysis of sealed containers, material properties should be determine at a temperature 50°C above the design temperature for the fuel unit handling and the storage areas. Rather than using 50°C, material properties of the sealed containers should be determined at temperatures appropriate for off-normal and accident conditions.
Design	C.6	**REGULATORY POSITION** **No Title for subsection 6.**	Section 6.4.1.3 of ANSI/ANS-57.9-1984 indicates that, if the fuel unit heat generation rate is calculated, ANSI/ANS-5.1-1979 is to be used as the basis for the calculation. However, the calculational methods in Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," should be used rather than ANSI/ANS-5.1.
Design	C.7	**REGULATORY POSITION** **No Title for subsection 7.**	Section 6.17.2.2.2 should be supplemented to add Section III of the ASME Boiler and Pressure Vessel Codes, as appropriate, in addition to Section VIII.
Design	C.8	**REGULATORY POSITION** **No Title for subsection 8.**	Section 7 of ANSI/ANS-57.9-1984 lists the codes and standards that are referenced in the standard. Endorsement of ANSI/ANS-57.9-1984 by this regulatory guide does not constitute an endorsement of the referenced codes and standards.
Design	C.9	**REGULATORY POSITION** **No Title for subsection 9.**	ANSI/ANS-57.9-1984 includes a number of appendices. Endorsement of the standard by this regulatory guide does not constitute an endorsement of the appendices.
Design	D	**IMPLEMENTATION**	The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described in this guide will be used in the evaluation of applications for a dry storage ISFSI submitted under 10 CFR Part 72.

Categories:

Construction
Decommissioning
Design
General
Licensing
Operaton
Quality Assurance

Category	REG 3.50 Section	REG Guide 3.50 SECTION TITLE Subsection Title	Criteria Description
Licensing	A	**INTRODUCTION** (Part 1 of 3)	**Purpose** This regulatory guide provides a description of a standard format and content that the U.S. Nuclear Regulatory Commission (NRC) staff considers acceptable for specific license application for Independent Spent Fuel Storage Installations (ISFSIs) and Monitored Retrievable Storage (MRS) facilities. **Applicable Rules and Regulations** • Title 10, Part 72, of the Code of Federal Regulations(10 CFR 72), “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” Subpart B, “License Application, Form, and Contents,” specifies the information that must be in an application for a license to store spent nuclear fuel, highlevel radioactive waste, and power-reactor-related greater than Class C (GTCC) waste in an ISFSI or in a MRS facility.

Licensing	A	**INTRODUCTION** (Part 2 of 3)	**Related Guidance** • Regulatory Guide (RG) 3.62, "Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks," provides guidance on the preparation of the Safety Analysis Report (SAR) for an ISFSI or MRS facility using dry storage. It also provides information on technical specifications that is useful for ISFSIs and MRS facilities • NUREG-1757, Volume 3, Revision 1, "Consolidated Decommissioning Guidance – Financial Assurance, Record Keeping and Timeliness," contains guidance on financial assurance for ISFSIs licensed under 10 CFR Part 72. • RG 5.55, "Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities," although it does not specifically address ISFSIs or MRS facilities, contains information that could be useful in developing safeguards contingency plans for these facilities. • RG 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," provides format and technical content information for emergency plans which are required by 10 CFR 72.32. • RG 5.44, "Perimeter Intrusion Alarm Systems," although it does not specifically address ISFSIs or MRS facilities, contains information that could be useful in developing physical security plans for these facilities. • NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs," provides format and technical content information for environmental reports which are required by 10 CFR 72.34.

Licensing	A	**INTRODUCTION** (Part 3 of 3)	**Purpose of Regulatory Guides** The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission. **Paperwork Reduction Act** This regulatory guide contains information collection requirements covered by 10 CFR Part 72 that the Office of Management and Budget (OMB) approved under OMB control number 3150-0132. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

Licensing	B	**DISCUSSION** (Part 1 of 2)	**Reason for Revision** This revision to RG 3.50 (Revision 2) was issued to conform to the format and content requirements in 10 CFR Part 72, which has been revised several times since Revision 1 was issued, and to update guidance on electronic submissions of applications. In addition, Revision 2 includes editorial changes to improve clarity. **Background** RG 3.50 was originally issued in January 1982 to provide an acceptable format for the content of license applications for spent fuel facilities. Revision 1 of this guide was published in September 1989 to include MRS's and updates to 10 CFR 72. Revision 1 of RG 3.50 became outdated because it discussed how to submit forms on microfilm and the agency has now moved most of its document submission to electronic form. Most of the guidance that was referenced in Revision 1 has been withdrawn, such as Regulatory Guide 3.44 "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type)," and American Nuclear Society Institute (ANSI) (correction, ANSI stands for American National Standards Institute) Standard N299-1976 "Administrative and Managerial Control for the Operation of Nuclear Fuel Reprocessing Plants." The information from these referenced documents has been captured in RG 3.62 "Standard Format and Content for the Safety Analysis Report for onsite Storage of Spent Fuel Storage Casks" and the current version of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

Licensing	B	**DISCUSSION** (Part 1 of 2)	**Harmonization with International Standards** The International Atomic Energy Agency (IAEA) has established a series of safety guides and standards constituting a high level of safety for protecting people and the environment. IAEA safety guides present international good practices and increasingly reflects best practices to help users striving to achieve high levels of safety. Pertinent to this regulatory guide, the IAEA safety series does not contain a similar document. The only format and content guidance that IAEA has issued is found in GS-G-4.1, “Format and Content of the Safety Analysis Report for Nuclear Power Plants." This document specifically describes format and content for SARs for nuclear plants, but does not cover spent fuel storage facilities. IAEA Safety Guide SSG-12, “Licensing Process for Nuclear Installations,” contains a brief section that lists required contents of a license, but it is not specific to spent fuel storage facilities and it does not provide any detail or format. Lastly, IAEA Safety Guide SSG -15, “Storage of Spent Nuclear Fuel,” addresses the design, operation and safety assessment of spent fuel storage facilities as well as the application of safety objectives, principles and criteria to the storage of spent nuclear fuel, but does not provide guidance on the format and content of applications.
Licensing	C	**STANDARD FORMAT AND CONTENT**	This regulatory guide provides a format that the NRC considers acceptable for submitting the information for 10 CFR Part 72 license applications to store spent nuclear fuel, high-level radioactive waste, and reactor-related Greater than Class C (GTCC) waste pursuant to a specific license. Conformance with this guide is not mandatory and the NRC staff will consider a license application with different formats acceptable if it provides an adequate basis for the findings required for the issuance of a license. The staff recommends using the format suggested in this regulatory guide because doing so will allow for a more efficient review by the staff and a potential reduction in the extent or the number of requests for additional information.

Licensing	C.1	<u>**STANDARD FORMAT AND CONTENT**</u> **Contents of the License Application**	The license application is the document that should address each of the requirements of 10 CFR Part 72 and should be completed upon submittal. The application is required to contain general information about the applicant, pursuant to 10 CFR 72.22. The license application should also include the following documents: • SAR (see §72.24 and RG 3.62); • Quality assurance (QA) program (see 10 CFR 72.24(n) and 10 CFR Part 72.140 (d)); • Physical security plan (including guard training) (see 10 CFR 72.24(o), and 10 CFR 72.180); • Safeguards contingency plan (see 10 CFR 72.184) ; • Proposed technical specifications (see 10 CFR 72.26 and RG 3.62); • Applicant's technical qualifications (see 10 CFR 72.28); • Personnel training program (see 10 CFR 72.28(b)); • Decommissioning plan and decommissioning funding plan (see 10 CFR 72.30 (a) and 10 CFR 72.54(g),; • Emergency plan (see 10 CFR 72.32),; • Environmental report (see 10 CFR 72.34); and • Proposed license conditions (see 10 CFR 72.44).

Licensing	C.2	**STANDARD FORMAT AND CONTENT** **Format and Style** (Part 1 of 2)	The applicant should strive for a clear, concise presentation of the information provided in the license application. Confusing or ambiguous statements and unnecessarily verbose descriptions do not contribute to expeditious technical review. Claims about the adequacy of designs or design methods should be supported by technical bases (i.e., an appropriate engineering evaluation or description of actual tests). Terms should be used as defined in 10 CFR Part 72, specifically or including 10 CFR 72.3. If a particular regulatory requirement does not apply to the proposed storage facility, the applicant should use the term “Not Applicable” instead of omitting the corresponding section. In addition, applicants should justify their decision not to address a particular requirement when its applicability is questionable. Appendices to each document in an application should include any appropriate detailed information that was omitted from the main text. The first appendix to a given document in an application should provide a list of documents that are referenced in the text of that application, including page numbers, if appropriate. If a license application references a proprietary document, it should also reference the nonproprietary summary description of that document. Applicants may also use appendices to provide supplemental information such as calculational methods or design approaches used by the applicant. When a license application cites numerical parameters or values, the number of significant figures should reflect the accuracy or precision to which the number is known. When possible, the applicant should specify estimated limits of error or uncertainty. Applicants should not drop or round off significant figures if this action would affect subsequent conclusions.

Licensing	C.2	**<u>STANDARD FORMAT AND CONTENT</u>** **Format and Style** (Part 2 of 2)	Applicants should use acronyms, abbreviations, symbols, and special terms consistently throughout a license application and in a manner that is consistent with generally accepted usage. Each document in an application should define any acronyms, abbreviations, symbols, or special terms used in the given section that are unique to the proposed storage system or not common in general usage. Applicants should use drawings, diagrams, sketches, and charts when these media would more accurately or conveniently convey the information. However, applicants should ensure that drawings, diagrams, sketches, and charts present information in a legible and consistent form and define relevant symbols. In addition, applicants should not reduce drawings, diagrams, sketches, and charts to the extent that readers need visual aids to interpret pertinent information. Applicants should number pages sequentially within each document, section, and appendix. For example, the fourth page of Section six would be numbered 6-4. A title page should identify key individuals responsible for the preparation of the license application and should include the oath or affirmation as required by 10 CFR 72.16(b). A table of contents should also be included. Applications that do not contain the information described in the regulations may be rejected for review by the NRC.

Licensing	C.3	**STANDARD FORMAT AND CONTENT** **Submissions and Revisions** (Part 1 of 3)	**Procedures for Submissions** Applications may be submitted either electronically, by mail, or by hand delivery to NRC headquarters. For details on communications with the NRC, including submitting applications, see 10 CFR 72.4. Detailed guidance on submitting electronic applications and supplements can be found on the NRC's Web site at http://www.nrc.gov/site-help/e-submittals.html; by e-mail to MSHD.Resource@nrc.gov; or by writing to the Office of Information Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. The guidance on electronic submissions discusses, among other topics, the formats that the NRC can accept, the use of electronic signatures, and the treatment of nonpublic information. If electronic submissions are utilized, applicants are encouraged to also send an electronic copy to the pertinent NRC project manager.

Licensing	C.3	**STANDARD FORMAT AND CONTENT** **Submissions and Revisions** (Part 2 of 3)	**Procedures for Updating or Revising Pages** For applicants making electronic submissions, a consolidated document is preferable to submission of individually edited pages and will enable reviewers to have the latest information with minimal effort to print and replace pages. For paper submissions, applicants should update data and text by replacing entire pages whenever a change is made to that page. Applicants should also highlight the updated or revised portion of each page using a "change indicator" consisting of a bold vertical line drawn in the margin opposite the binding margin. All pages submitted to update, revise, or add pages to an application should show the date of the revision and the corresponding change or amendment number. A transmittal letter, including a guide page listing the pages to be inserted and removed, should accompany the revised pages. When applicable, supplemental pages may follow the revised page, with the pages still being numbered sequentially. Applicants should distinguish between changes made under the change authority in 10 CFR 72.48 (c)(1) and amendment to the license or Certificate of Compliance as required by 72.48 (c)(2). All statements on a revised page should be accurate as of the date of each submittal. Applicants should take special care to ensure that they revise the documents submitted as part of the application to reflect any changes to the design, contents, analysis, and tests reported in supplemental information (e.g., responses to NRC staff requests for information or responses to regulatory positions).

Licensing	C.3	**STANDARD FORMAT AND CONTENT** **Submissions and Revisions** (Part 3 of 3)	**Referenced Materials** Under 10 CFR 72.18, applicants may avoid repetition by incorporating by reference material previously filed with the NRC. However, applicants should use caution in making such references and should ensure that they are pertinent to the subject discussed, contain current information, and are readily obtainable or extractable from the referenced documents. It may be more efficient in some cases to repeat, or summarize, information furnished in the previously submitted document. **Protection of Proprietary Information** The applicant should identify and submit under separate cover any information that it considers proprietary. The requirements in 10 CFR 2.390(b) should be followed for such information. For safeguards information, applicants should also adhere to requirements in 10 CFR 73.21, 10 CFR 73.22, and 10 CFR 73.23 as applicable.
Licensing	C.4	**REGULATORY POSITION** **Further Information** (Part 1 of 6)	**General and Financial Information** Information on the contents of applications is found in 10 CFR 72.22. Applicants, except for DOE, must provide sufficient information to demonstrate to the Commission that they can satisfy the financial qualifications of activities associated with an ISFSI or MRS facility. This includes but is not limited to: estimated construction costs, estimated operating costs over the planned life of the facility, and estimated decommissioning costs. **Safety Analysis Report** Each application for a license should include a SAR as described in 10 CFR 72.24. The information should describe the proposed ISFSI or MRS facility for the receipt, handling, packaging, and storage of spent fuel, high-level radioactive waste and/or reactor related GTCC waste. Regulatory Guide 3.62, "Standard Format and Content for a Safety Analysis Report for Dry Storage of Spent Fuel at an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility," (title error, actual title is "Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fue Storage Casks") provides additional guidance on the preparation of the SARs for ISFSIs and MRS facilities using dry storage.

Licensing	C.4	**REGULATORY POSITION** **Further Information** (Part 2 of 6)	**Quality Assurance Program** The application should contain either the QA program required by 10 CFR Part 72, Subpart G, "Quality Assurance" (as an enclosure), or should reference a currently NRC-approved QA program. The SAR should briefly describe the QA program. A QA program that has been approved by the NRC as meeting Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," may be applied to the ISFSI. Note that 10 CFR 72.140(d) states, "A quality assurance program previously approved by the Commission as satisfying the requirements of Appendix B of Part 50 of this chapter, Subpart H to part 71 of this chapter, or Subpart G to this part will be accepted as satisfying the requirements of paragraph (b) of this section, except that a licensee, applicant for a license, certificate holder, and applicant for a CoC who is using an Appendix B or Subpart H quality assurance program shall also meet the record keeping requirement of 72.174. In filing the description of the quality assurance program required by paragraph (c) of this section, each licensee, applicant for license, certificate holder, and applicant for a COC shall notify the NRC, in accordance with section 72.4, of its intent to apply its previously-approved quality assurance program to ISFSI activities or spent fuel storage casks activities. The notification shall identify the previously-approved quality assurance program by date of submittal to the Commission, docket number, and date of Commission approval."

Licensing	C.4	**REGULATORY POSITION** **Further Information** (Part 3 of 6)	**Physical Protection Plan** As discussed in 10 CFR 72.24(o), as part of the licensing process, the applicant must submit a physical protection program that satisfies the requirements in 10 CFR Part 72, Subpart H, "Physical Protection." Because the details of the provisions for physical protection are withheld from public disclosure, the applicant may submit this document(s) separately from the rest of the application. Thelicense application should contain a reference to the submission for the physical security program and the date of NRC approval if the NRC had approved the program before submittal of the application. The physical protection plan should describe the design criteria for the physical protection of the proposed ISFSI or MRS facility, the design bases, and how the design bases relate to the design criteria, and should ensure that the physical protection plan meets the requirements in 10 CFR 73.51, "Requirements for the physical protection of stored spent nuclear fuel and high-level radioactive waste." **Safeguards Contingency Plan** A safeguard contingency plan is a documented plan to give guidance to licensee personnel in order to accomplish specific defined objectives in the event of threats, theft, or radiological sabotage relating to special nuclear materials or nuclear facilities. As required by 10 CFR 72.184, the licensee shall prepare and maintain a safeguards contingency plan in accordance with Appendix C to 10 CFR Part 73 "Nuclear Power Plant Safeguards Contingency Plans." Although RG 5.55, "Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities" relates to a fuel cycle plant, it provides information that might be useful for creating safeguards contingency plans for ISFSIs and MRS facilities.

Licensing	C.4	**<u>REGULATORY POSITION</u>** **Further Information** (Part 4 of 6)	**<u>Proposed Technical Specifications</u>** The regulations in 10 CFR 72.26 "Contents of application: Technical specifications," require applications to include proposed technical specifications in accordance with requirements of 10 CFR 72.44 "License conditions," in addition to a summary statement of the bases of and justifications for these technical specifications. For more information on technical specifications, please review RG 3.62. **<u>Technical Qualifications</u>** Title 10 CFR 72.40(a)(4) requires a finding by the NRC that the applicant is qualified through training and experience to operate an ISFSI or MRS facility. Information that the application must include for this purpose can be found in 10 CFR 72.28, "Contents of application: Applicant's technical qualifications." The licensee is responsible for implementing the proposed project as described in the license application. This means that, even though a contractor may perform much of the actual work involved during the site selection, design, procurement, construction, and even the operating phases of the project, the licensee must have staff that is knowledgeable in all aspects of the project. The application should include the applicant's experience to show that it has the technical qualifications to construct and operate (or oversee the construction and operation of) the ISFSI or MRS facility. Note that if previous sections have discussed the operating organization and delegations and/or adequately described the minimum skills and experience, the information need not be repeated but may be referenced as appropriate.

Licensing	C.4	**<u>REGULATORY POSITION</u>** **Further Information** (Part 5 of 6)	**<u>Personnel Training Program</u>** Applicants should describe a training program in their application as discussed in 10 CFR 72.28(b). Requirements for the personnel training program are in 10 CFR Part 72, Subpart I "Training and Certification of Personnel." **<u>Decommissioning Plan and Decommissioning Funding Plan</u>** The proposed final decommissioning plan should include all the criteria discussed in 10 CFR 72.54(g). Updated and detailed plans must be submitted and approved by the Commission prior to the start of any decommissioning activity. Each application should include a decommissioning plan and decommissioning funding plan that contains sufficient information on proposed practices and procedures for decontamination and decommissioning and associated funding in accordance with the requirements of 10 CFR 72.30, "Financial assurance and recordkeeping for decommissioning." NUREG-1757, Volume 3, Revision 1, "Consolidated Decommissioning Guidance -- Financial Assurance, Recordkeeping and Timeliness, contains additional guidance on financial assurance for ISFSI's licensed under 10 CFR Part 72.

Licensing	C.4	**REGULATORY POSITION** **Further Information** (Part 6 of 6)	**Emergency Plan** The applicant should submit a plan for coping with emergencies as discussed in 10 CFR 72.32. If the ISFSI is located on the site of a facility licensed under 10 CFR Part 50, the emergency plan required by 10 CFR 50.47, "Emergency plans," satisfies the requirements in 10 CFR 72.32, "Emergency Plan." Additionally for ISFSIs or MRS facilities that are not located on the site of a nuclear power plant, the guidance in Regulatory Guide 1.101, "Emergency Response Planning and Preparedness for Nuclear Power Reactors," provides useful information for applicants when developing the Emergency Plan for a site-specific ISFSI. **Environmental Report** The regulations at 10 CFR 72.34, "Environmental report," require applicants to submit as part of the license application, an environmental report that satisfies the requirements in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)." Chapter 6 of NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs," issued August 2003, provides format and technical content information for the environmental report that is required by 10 CFR 72.34. **Proposed License Conditions** According to 10 CFR 72.44, license conditions are required to be included with the license. Applicants may propose license conditions to address design, construction and operation of the facilities.

Licensing	D	**IMPLEMENTATION** (Part 1 of 2)	The purpose of this section is to provide information on how applicants and licensees1 may use this regulatory guide and information regarding the NRC staff's plans for using this guide. In addition, it describes how the NRC staff has complied with the backfitting provisions in 10 CFR 72.62 and issue finality provisions of 10 CFR Part 52. The staff recommends that applicants use the format suggested in this regulatory guide because doing so will allow for a more efficient review by the staff and potentially reduce the extent or the number of staff requests for additional information. Conformance with this guide is not mandatory and the NRC staff will consider license applications with different formats acceptable if they provide an adequate basis for the findings required for the issuance of a license. This regulatory guide applies only to applicants who are not within the scope of entities protected by § 72.62. In addition, the subject matter of this regulatory guide does not concern matters dealing with either of the structures, systems and components of an ISFSI or MRS, or the procedures or organization for operating an ISFSI or MRS. Therefore, the matters addressed in this regulatory guide are not within the scope of the backfitting provisions in § 72.62(a)(1) or (2).
Licensing	D	**IMPLEMENTATION** (Part 2 of 2)	This regulatory guide does not apply to entities protected by issue finality provisions in 10 CFR Part 52 with respect to the matters addressed in this regulatory guide. Although Part 52 combined license applicants and holders may apply for specific ISFSI licenses, the guidance in this regulatory guide is directed to ISFSI applicants and does not make a distinction between ISFSI applicants who are also combined license applicants or holders and ISFSI applicants who are not combined license applicants and holders, and presents no more onerous guidance for ISFSI applicants who are also combined license applicants or holders versus ISFSI applicants who are not combined license applicants and holders. Accordingly, the NRC concludes that the staff's use of this regulatory guide is not inconsistent with any Part 52 issue finality provisions.

Categories:
Construction
Decommissioning
Design
General

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A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53

Category	REG 3.53 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design, Operation	A	**INTRODUCTION** (Part 1 of 2)	The storage of spent fuel in an independent spent fuel storage installation (ISFSI) pending its ultimate disposal is a new step in the nuclear fuel cycle that is licensed pursuant to 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation." The conventional method for such storage is in water basins embedded in the ground with their water surface approximately at grade level. Other methods using dry storage are also being considered. These methods may include air-cooled canyons or vaults, underground caissons or dry wells, or surface-storage casks or silos. Applicants may determine that other regulatory guides contain useful information for particular situations. Potential use of other guides may be discussed with the NRC staff.
Design, Operation	A	**INTRODUCTION** (Part 2 of 2)	This regulatory guide identifies existing regulatory guides that may be applicable in whole or in part to the design and operation of an ISFSI. Since the different modes of storage vary widely in design, the guides cited will obviously not all be applicable to all design technologies. Also, as a general rule, the technologies and operating conditions involved in the receipt and storage of aged spent fuel (i.e., spent fuel that has undergone at least 1 year of decay since removal from a reactor core) are not only much less complex and dynamic than those of production and utilization facilities (i.e., reactor and reprocessing facilities), but they are also less complex and dynamic than the technologies and conditions involved in the receipt and storage of spent fuel in reactor basins designed to receive spent fuel directly from a reactor core after a decay of a few days or less. The referenced guides are useful not only because the methods of design and operation cited have been examined by the NRC staff and found to be appropriate as a means of meeting the requirements of NRC regulations, but also because these guides are familiar to licensees and applicants. Thus, while the guides may exceed the requirements of Part 72 in some cases (in particular, those guides written with reference to power reactors), they can be of benefit to applicants and licensees who already have experience with the solutions endorsed in them and who may wish to apply familiar solutions rather than develop alternative solutions less certain of being acceptable to the NRC staff.

Design, Operation	B	**DISCUSSION**	Existing regulatory guides were examined for their potential applicability to the design and operation of an ISFSI that may use either a wet or dry mode of storage. The specific revision of each guide that may be applicable, in whole or in part, is listed in the tables of this guide. This guide will be updated as referenced guides are revised. The user and staff must exercise discretion in using all of the detailed information associated with each regulatory position cited, e.g., not all appendices and examples contained in the cited guides are applicable to an ISFSI. If the guidance in a guide written specifically for an ISFSI differs from that in a guide developed for another facility, (e.g., a Division 1 guide), the guidance in the guide specific to an ISFSI should be followed.
Design, Operation	C	**REGULATORY POSITION**	Tables 1, 2, and 3 list existing regulatory guides that may be applicable to an ISFSI. Table 1 (Excel Sheet: Reg Guide 3.53 Table 1) identifies guides applicable to ISFSI design and Table 2 (Excel Sheet: Reg Guide 3.53 Table 2) identifies guides applicable to ISFSI operation. Table 3 (Excel Sheet: Reg Guide 3.53 Table 3) identifies guides that are specific to an ISFSI. The guides are listed in the tables by number, title, and revision under general subjects that are arranged alphabetically. Relevant regulatory positions in each guide are identified and briefly described. In Tables 1 and 2, the portions of the NRC regulations addressed by the regulatory positions are also identified and other pertinent information is provided.

Categories:
Construction
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A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 1

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Accident Analysis	1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25).	2	Atmospheric diffusion assumptions	72.65(a), 72.74(d)
		3.b	External whole-body approximations assumptions	72.65(a), 72.74(d)
	1.91, Revision 1, Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants	1	Explosive transport	72.63, 72.72(c)
		2	Explosive transport	72.63, 72.72(c)
		3	Explosive transport	72.63, 72.72(c)
	1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants			In applying this guide to an ISFSI, substitute the terms " controlled area" as defined in § 72.3(h) and "ISFSI-emergency planning zone (ISFSI-EPZ)" as defined in § 72.3(n) respectively for the terms "exclusion area" and "low population zone (LPZ)" wherever they appear.
		1.2	Distances for X/Q	72.65(a), 72.74(d)
		1.3	X/Q at the controlled area	72.65(a), 72.74(d)
		1.4	X/Q at EPZ	72.65(a), 72.74(d)
		2	Maximum sector X/Q values	72.65(a), 72.74(d)
		3	5% overall site X/Q value	72.65(a), 72.74(d)
		4	Selection of X/Q	72.65(a), 72.74(d)

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 1

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Control Room-Chemical Release	1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	1	Stored chemicals	72.61(b), 72.72(j)
		2	Transported chemicals	72.61(b), 72.72(j)
		3	Onsite chemicals	72.61(b), 72.72(j)
		4	Toxicity limits	72.72(j)
		5.a	Accident concentration	72.61(b), 72.72(j)
		6	Dilution factor	72.61(b), 72.72(j)
		11	Removal systems	72.72(j)
		12	Natural phenomena and chemical release	72.61(b), 72.72(j)
		15	Emergency procedures	72.15(a)(11)
Criticality Safety	3.41, Revision 1, Validation of Calculational Methods for Nuclear Criticality Safety	All	Endorsement of ANSI N16.9-1975	72.73(a)
Floods	1.59, Revision 2, Design Basis Floods for Nuclear Power Plants	1	Flood conditions	72.33(c)(4), 72.61(c), 72.62, 72.72(b)(2), 72.72(b)(4)
		2.a	Hardened protection alternative-warning time	72.33(c)(4), 72.61(c), 72.62, 72.72(b)(2), 72.72(b)(4)
		2.c	Hardened protection alternative-less severe flood conditions	72.33(c)(4), 72.61(c), 72.62, 72.72(b)(2), 72.72(b)(4)
		3	Unanticipated changes	72.33(c)(4), 72.61(c), 72.62, 72.72(b)(2), 72.72(b)(4)
		4	Data utilization	72.33(c)(4), 72.61(c), 72.62, 72.72(b)(2), 72.72(b)(4)
Flood Protection	1.102, Revision 1, Flood Protection for Nuclear Power Plants	1	Types of protection	72.72(b)(2)
		2	Shutdown specifications	72.72(b)(2)
		3	Vulnerability of safety-related equipment	72.72(b)(2)

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 1

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Physical Hydraulic Models	1.125, Revision 1, Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	1	Model preconstruction submittals	72.1 5(a)(3), 72.33(c)(4), 72.6 1(c), 72.62
		2	Early staff discussions	72.1 5(a)(3), 72.33(c)(4), 72.6 1(c), 72.63
		3	Documentation	72.1 5(a)(3), 72.33(c)(4), 72.6 1(c), 72.64
		4	Comparison of data and model	72.15(a)(3), 72.33(c)(4), 72.61(c), 72.62, 72.72(b)(3)
		5	Design changes	72.15(a)(3), 72.33(c)(4), 72.61(c), 72.62, 72.72(b)(3)
		6	Report contents	72.15(a)(3), 72.33(c)(4), 72.61(c), 72.62, 72.72(b)(3)
Quality Assurance-Design and Construction	1.28, Revision 2, Quality Assurance Program Requirements (Design and Construction)	General	Endorsement of ANSI N45.2-1977	
		1		72.15(a)(14), 72.72(a), 72.80
		4		72.15(a)(14), 72.72(a), 72.80
Quality Assurance-Terms	1.74, Quality Assurance Terms and Definitions	Second paragraph	Procurement documents	72.80

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Radiological Protection-ALARA	8.8, Revision 3, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable			In applying this guide to an ISFSI, substitute the term "ISFSI" for the terms "LWR" and "nuclear power station" wherever they appear. Disregard references to the nuclear steam supply vendor.
		1.d	Review of designs and equipment	20.1 (c), 72.15(a)(5), 72.74(a)
		2.a	Access control of radiation areas	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(1)	Shielding for service personnel	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(2)	Temporary shielding and distance	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(4)	Streaming and scattering	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(5)	Streaming	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(6)	Reduction of exposure from pipes	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(7)	Expeditious design features	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(8)	Laydown space	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(9)	Removal of equipment	20.1 (c), 72.15(a)(5), 72.74(a)
		2.b(10)	Drains	20.1 (c), 72.15(a)(5), 72.74(a)
		2.c	Process instrumentation and controls	20.1 (c), 72.15(a)(5), 72.74(a)
		2.d(1)	Control of airborne contaminants (air flow)	20.1(c), 72.15(a)(5), 72.74(a), 72.74(c), 72.74(d)
		2.d(2)	Ventilation systems	20.1(c), 72.15(a)(5), 72.15(a)(1 2), 72.74(a) 72.74(d)
		2.d(3)	Auxiliary ventilation systems	20.1(c), 72.15(a)(5), 72.15(a)(12), 72.74(a), 72.74(c), 72.74(d)
		2.f	Isolation and decontamination	20.1(c), 72.15(a)(5), 72.74(a), 72.74(b)
		2.g	Radiation monitoring systems	20.1(c), 72.15(a)(5), 72.74(b), 72.74(c)
		2.h(1)	Reduction of accumulation	20.1(c), 72.15(a)(5), 72.74(a)

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 1

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
		2.h(2)	Need for maintenance	20.1(c), 72.15(a)(5), 72.74(a)
		2.h(3)	Pipe bends	20.1(c), 72.15(a)(5), 72.74(a)
		2.h(4)	Pipe surfaces	20.1(c), 72.15(a)(5), 72.74(a)
		2.h(5)	Pipe tees	20.1(c), 72.15(a)(5), 72.74(a)
		2.h(6)	Slurry piping	20.1(c), 72.15(a)(5), 72.74(a)
		2.h(7)	Sparging	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(1)	Radiation-damage-resistant materials	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(2)	Stainless steel piping	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(3)	Pipe routing	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(4)	Filters	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(5)	Servicing valves	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(7)	Valve selection	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(8)	Pumps	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(9)	Sedimentation	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(10)	Spare pipe connections	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(11)	Station design	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(12)	Component removal	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(13)	Working environment	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(14)	Lamp replacement	20.1(c), 72.15(a)(5), 72.74(a)
		2.i(15)	Emergency lighting	20.1(c), 72.15(a)(5), 72.74(a)
		3.a	Radiation protection program preparation and planning	20.1(c), 72.33(c)(5)
		3.b(1)	Health physics technicians	20.1(c), 72.33(c)(5)
		3.b(3)	Communications	20.1(c), 72.33(c)(5)
		3.c	Postoperations	20.1(c), 72.33(c)(5)
		4	Radiation protection facilities instrumentation and equipment	20.1(c), 72.33(c)(5), 72.74

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 1

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Seismic	1.29, Revision 3, Seismic Design Classification			In applying this guide to an ISFSI, substitute the term "ISFSI Design Earthquake (ISFSI-DE)" for the "Safe Shutdown Earthquake (SSE)" wherever it appears.
		1.d	All sections listed for position 1 describe applicable activities to be included	72.72(b)(2)
		1.g		72.72(b)(2)
		1.i		72.72(b)(2)
		1.j		72.72(b)(2)
		1.k		72.72(b)(2), 72.72(i)
		1.l		72.72(b)(2)
		1.n		72.72(j)
		1.p		72.75(a)
		1.q		72.72(b)(2)
		2	Non-safety-related components	72.72(b)(2)
		3	Defined boundaries	72.72(b)(2)
	1.60, Revision 1, Design Response Spectra for Seismic Design of Nuclear Power Plants			In applying this guide to an ISFSI, substitute the term "ISFSI Design Earthquake (ISFSI-DE)" for the term "Safe Shutdown Earthquake (SSE)" wherever it appears. The term "Operating Basis Earthquake (OBE)" is not applicable to an ISFSI.
		1	Horizontal component	72.66(a)(2), 72.66(a)(6), 72.66(b)
		2	Vertical component	72.66(a)(2), 72.66(a)(6), 72.66(b)

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Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
	1.61, Damping Values for Seismic Design of Nuclear Power Plants			In applying this guide to an ISFSI, substitute the term "ISFSI Design Earthquake (ISFSI-DE)" for the term "Safe Shutdown Earthquake (SSE)" wherever it appears. The term "Operating Basis Earthquake (OBE)" is not applicable to an ISFSI.
		1	Modal damping values	72.66(a)(2), 72.66(a)(6), 72.66(b)
		2	High damping values	72.66(a)(2), 72.66(a)(6), 72.66(b)
		3	Combined stress	72.66(a)(2), 72.66(a)(6), 72.66(b)
	1.92, Revision 1, Combining Modal Responses and Spatial Components in Seismic Response Analysis	1	Combining of modal responses	72.66(a)(2), 72.66(a)(6), 72.66(b)
		2	Combination of effects	72.66(a)(2), 72.66(a)(6), 72.66(b)
	1.122, Revision 1, Development of Floor Design Response Spectra for Seismic Design of FloorSupported Equipment or Components	1	Directional analysis	72.66(a)(2), 72.66(a)(6), 72.66(b)
		2	Uncertainties	72.66(a)(2), 72.66(a)(6), 72.66(b)
		3	Response spectrum	72.66(a)(2), 72.66(a)(6), 72.66(b)

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 1

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Site Investigations-Foundations	1.132, Revision 1, Site Investigations for Foundations of Nuclear Power Plants			This guide applies to all types of ISFSI designs (§ 72.66). If massive structures or foundations are required by the specific design (i.e., water basin, vault, canyon, support hot cell), this guide would provide applicable guidance for the site investigation.
		1	General site investigation	72.61(a), 72.66(a)(4)
		2	Boring logs	72.61(a), 72.66(a)(4)
		3	Ground-water investigations	72.61(a), 72.66(a)(4)
		4	Procedures	72.61(a), 72.66(a)(4)
		5	Spacing and depth	72.61(a), 72.66(a)(4)
		6	Sampling	72.61(a), 72.66(a)(4)
		7	Retention of records and samples	72.61(a), 72.66(a)(4)
Site Investigations-Soils	1.138, Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants	1	Requirements for testing program	72.61(a), 72.66(a)(4)
		2	Handling and storage of samples	72.61(a), 72.66(a)(4)
		3	Selection and preparation of specimens	72.61(a), 72.66(a)(4)
		4	Criteria for testing procedures	72.61(a), 72.66(a)(4)
		5	Documentation	72.61(a), 72.66(a)(4)

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Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Structures-Concrete Shields	1.69, Concrete Radiation Shields for Nuclear Power Plants	General	Endorsement of ANSI-N101.6-1972	72.15(a)(3), 72.33(c)(4), 72.74(a)
		1		72.80
		2		72.80
		3		72.15(a)(3), 72.33(c)(4), 72.74(a)
		4		72.15(a)(3), 72.33(c)(4), 72.74(a)
		5		72.15(a)(3), 72.33(c)(4), 72.74(a)
		6		72.15(a)(3), 72.33(c)(4), 72.74(a)
		7		72.15(a)(3), 72.33(c)(4), 72.74(a)
		8		72.15(a)(3), 72.33(c)(4), 72.74(a)
Testing-Diesel Generator	1.108, Revision 1, Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	1	General design	72.72(f), 72.72(k)
		2	Testing	72.72(k)
		3	Records and reports	Reports are not submitted pursuant to Regulatory Guide 1.16. 72.72(k)
Testing-Protective	1.22, Periodic Testing of Protection System Actuation Functions (Safety Guide 22)			In applying this guide to an ISFSI, substitute the expression "ISFSI receiving and storage operations" for the expression "reactor operation" wherever it appears.
		1	Testing requirements	72.72(f)
		2	Testing methods	72.72(f)
		4	Untested equipment	72.72(f)

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Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Tornado	1.76, Design Basis Tornado for Nuclear Power Plants	1	Design basis tornado	72.6 1(c), 72.62
		2	Less conservative design basis tornado	72.6 1(c), 72.62
Waste Management Systems	1.143, Revision 1, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants			This guide does not refer to the spent fuel storage systems such as the basin, cask, vault, etc., but only applies to the systems that are used to collect, store, control, or process waste that is generated during the ISFSI operation.
		Liquid Waste System		
		1.1.1	Design and test requirements	72.72(a)
		1.1.3	Seismic criteria	72.72(b)(1), 72.72(b)(2)
		1.1.4	Seismic criteria	72.72(b)(1), 72.72(b)(2)
		1.2	Tank design	72.75
		Solid Waste System		
		3.1.1	Design and test requirements	72.72(a)
		3.1.3	Seismic criteria	72.72(b)(1), 72.72(b)(2)
		3.1.4	Seismic criteria	72.72(b)(1), 72.72(b)(2)
		4.1	ALARA	72.75
		4.4	Hydrostatic testing	72.75
		4.5	Testing	72.75
		5.2	Buildings housing radwaste systems	72.72(b)(1), 72.72(b)(2)
		5.3	Optional shielding	72.72(b)(1), 72.72(b)(2)

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Atmospheric Releases	4.16, Measuring, Evaluating, and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Airborne Effluents from Nuclear Fuel Processing and Fabrication Plants			Examples in this guide are not applicable to an ISFSI. Environmental reports for an ISFSI should be submitted on an annual basis rather than semiannually as stated in position 5.1.
		1	Methods of sampling analysis	72.74(c)
		2	Methods of sampling analysis	72.74(c)
		3	Methods of sampling analysis	72.74(c)
		4	Precision and accuracy of results	72.74(c)
		5	Precision and accuracy of results	72.33(d)(3)
Atmospheric Transport	1. 111, Revision 1, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	Atmospheric transport and diffusion models	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
		2	Source configuration	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
		3	Removal mechanisms	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
		4	Meteorological data for models	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
Aquatic Dispersion	1.113, Revision 1, Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose	1	Transport and water-use models	72.15(a)(13), 72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
		2	Selection of models	72.15(a)(13), 72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
Dose Assessment	1.109, Revision 1, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	1	Doses from liquid effluent pathways	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
		3	Doses from airborne particulates	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)
		4	Integrated doses to population	72.33(d)(3), 72.61(e), 72.65(a), 72.74(d)

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Effluent Monitoring	1.21, Revision 1, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants			For an ISFSI that is co-located at a reactor site, as opposed to an ISFSI at a separate site, the monitoring requirements may be reduced. Monitoring may be needed at the fuel receiving and storage areas on a continuous basis only when spent fuel is being handled. Otherwise, periodic measurements may suffice, particularly for possible sealed storage modes.
		2	Location of monitoring	72.74(c)(1), 72.74(d)
		3	Type of monitoring	72.33(d), 72.74(c)(1), 72.74(d)
		4	Gross radioactivity measurements	72.33(d), 72.74(c)(1), 72.74(d)
		5	Measurements of specific radionuclides	72.33(d), 72.74(c)(1), 72.74(d)
		6	Representative samples	72.33(d), 72.74(c)(1)
		7	Composite samples	72.33(d), 72.74(c)(1)
		8	Time between collection and analysis	72.33(d), 72.74(c)(1)
		9	Corrections for decay	72.33(d), 72.74(d)
		11.a	Errors in measurement	72.33(d), 72.74(d)
		11.b	Quality controls	72.80(b)
		11.c	Calibrations	72.74(d)
		12.b	Significant figures	72.33(d), 72.74(d)
		12.c	Numerical values	72.33(d), 72.74(d)

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.53 Table 2

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Environmental Monitoring	4. 1, Revision 1, Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants			The preoperational monitoring period as stated in regulatory position 1 should be reduced from 2 years to 1 year.
		1	Preoperational program	72.33(d)(2), 72.67
		2.a	Sample media	72.33(d)(2), 72.67
		2.b	Sample frequency	72.33(d)(2), 72.67
		2.d	Analysis	72.33(d)(2)
		2.e	Quality control	72.33(d)(2), 72.80
		3	Detection capabilities	72.33(d)(2), 72.67
Environmental Monitoring-TLD	4.13, Revision 1, Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications	General	Endorsement of ANSI N545-1979	
		1		72.33(d)(2), 72.74(c)(2)
		2		72.33(d)(2), 72.74(c)(2)
		4		72.33(d)(2), 72.74(c)(2)
		5		72.33(d)(2), 72.74(c)(2)
Quality Assurance-Environmental Monitoring	4.15, Revision 1, Quality Assurance for Radiological Monitoring Programs (Normal Operations)-Effluent Streams and the Environment	1	Organization and responsibilities	72.17
		2	Personnel qualifications	72.17
		4	Records	72.33(c)(5), 72.80
		5	Quality control in sampling	72.33(c)(5), 72.81
		6	Quality control in analysis	72.33(c)(5), 72.82
		7	Quality control for continuous monitoring systems	72.33(c)(5), 72.83
		8	Review and analysis of data	72.33(c)(5), 72.84
		9	Audits	72.33(c)(5), 72.85

Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Radiological Protection--ALARA	8.8, Revision 3, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable			In applying this guide to an ISFSI, substitute the term "ISFSI" for the terms "LWR" and "nuclear power station" wherever they appear. Disregard references to the nuclear steam supply vendor.
		1	General-program goals	20.1(c), 72.33(c)(5)
		1.a	Establishment of program	20.1(c), 72.33(c)(5)
		1.b	Organization and personnel responsibilities	20.1(c), 72.17
		1.c	Training and instruction	20.1(c), 72.17, 72.92
	8.10, Revision I-R, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable	1.a	Management commitment	20.1(c), 72.15(a)(8), 72.17, 72.33(c)(5)
		1.b	Audits	20.1(c), 72.15(aX8), 72.33(c)(5)
		1.c	Responsibilities	20.1(c), 72.15(aX8), 72.17, 72.33(b)(4), 72.33(c)(5), 72.92
		1.d	Training	19.12, 72.17(d), 72.33(b)(4), 72.92
		1.e	RSO authority	20.1(c), 72.15(a)(8), 72.33(c)(5)
		1.f	Procedure modifications	20.1(c), 72.15(a)(8), 72.33(c)(5)
		2	Staff vigilance	20.1(c), 72.15(a)(8), 72.33(c)(3), 72.33(c)(5)
Radiological Protection-Air Sampling Instruments	8.25, Calibration and Error Limits of Air Sampling Instruments for Total Volume of Air Sampled	1	Calibration frequency	20.103(a)(3), 72.74(b)
		2	Error limit	20.103(a)(3), 72.74(b)
		3	Documentation	20.103(a)(3), 72.74(b)
Radiological Protection-Bioassay	8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	All	Assumption, models, concepts	20.108, 72.15(a)(12)
	8.26, Applications of Bioassay for Fission and Activation Products	All	Endorsement of ANSI N343-1978	20.108, 72.15(a)(8)

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Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Radiological Protection-Evacuation Signal	8.5, Revision 1, Criticality and Other Interior Evacuation Signals	All	Endorsement of ANSI/ANS N2.3-1979	72.74(b)
Radiological Protection-Pocket Dosimeters	8.4, Direct-Reading and Indirect-Reading Pocket Dosimeters	1	Testing	20.202(a), 20.401, 72.15(a)(5)
		2	Rejection	20.202(a), 20.401, 72.15(a)(5)
		3	Mixed radiation fields	20.202(a), 20.401, 72.15(a)(5)
Radiological Protection-Prenatal Exposure	8.13, Revision 1, Instruction Concerning Prenatal Radiation Exposure	1	Instruction	19.12
		2	Reasons	19.12
Radiological Protection-Respiratory Protection	8.15, Acceptable Programs for Respiratory Protection	1	Written policy	20.103
		2	Equipment selection	20.103
		3	Individual use of respirator	20.103
		4	Requirements of program	20.103
		5	Equipment approval	20.103
		6	Unapproved equipment	20.103
		7	Protection factors	20.103
		8	Technical requirements	20.103
Radiological Protection-Symbol	8.1, Radiation Symbol	General		20.203
Safeguards-Alarm Systems	5.44, Revision 2, Perimeter Intrusion Alarm Systems	1	Qualification	72.81
		2	Testing	72.81
Safeguards-Contingency Plans	5.55, Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	All	Contingency plans	72.83

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Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Safeguards-Entry/Exit Control	5.7, Revision 1, Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	1	Combination locks	72.81
		2	Combination padlocks	72.81
		3	Key locks	72.81
		4	Key padlocks	82.71(q), 72.81
		5	Electric locks	72.81
		6	Pushbutton mechanical locks	72.81
		7	Mechanical locks	72.81
		8	Combinations	72.81
Safeguards-Security Force	5.20, Training, Equipping, and Qualifying of Guards and Watchmen	1	Preemployment screening	72.81
		2	Training	72.81
		3	Testing and requalification	72.81
		4	Equipment	72.81
	5.43, Plant Security Force Duties	1	Organization	72.81
		2	Duties	72.81
Safeguards-Transportation	5.57, Revision 1, Shipping and Receiving Control of Strategic Special Nuclear Material	1	Preshipment controls on waste	72.81
		2	Overchecks	72.81
		3	Additional shipping controls	72.54, 72.81
		4	Receipts	72.54, 72.81
Safeguards-Visual Surveillance	5.14, Revision 1, Use of Observation (Visual Surveillance) Techniques in Material Access Areas	1	Operational measures	72.81
		2	Aids to effective surveillance	72.81

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Subject	Regulatory Guide	Regulatory Position		Portions of 10 CFR Addressed
Design-Water Basin	3.49, Design of an Independent Spent Fuel Storage Installation (Water-Basin Type)	All	Endorsement of ANSI/ ANS 57.7-1981	
License Application	3.50, Guidance on Preparing a License Application To Store Spent Fuel in an Independent Spent Fuel Storage Installation	All		
Safety Analysis Report-Water Basin	3.44, Revision 1, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type)	All		
Safety Analysis Report-Dry Storage	3.48, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)	All		

* The regulatory basis for all guides in Table 3 is 10 CFR Part 72.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	A	**INTRODUCTION** (Part 1 of 3)	The U.S. Nuclear Regulatory Commission (NRC) has recently published amendments to 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." Section 72.103, "Geological and Seismological Characteristics for Applications for Dry Modes of Storage on or after October 16, 2003," in paragraph (f)(1), requires that the geological, seismological, and engineering characteristics of a site and its environs be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site. The investigation must provide sufficient information to support evaluations performed to arrive at estimates of the design earthquake ground motion (DE) and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. In 10 CFR 72.103, paragraph (f)(2) requires that the geologic and seismic siting factors considered for design include a determination of the DE for the site, the potential for surface tectonic and non-tectonic deformations, the design bases for seismically induced floods and water waves, and other design conditions. In 10 CFR 72.103, Paragraph (f)(2)(i) requires that uncertainties inherent in estimates of the DE be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis (PSHA) or suitable sensitivity analyses. This guide is being developed to provide general guidance on procedures acceptable to the NRC staff for: (1) conducting a detailed evaluation of site area geology and foundation stability; (2) conducting investigations to identify and characterize uncertainty in seismic sources in the site region important for the PSHA; (3) evaluating and characterizing uncertainty in the parameters of seismic sources; (4) conducting PSHA for the site; and (5) determining the DE to satisfy the requirements of Part 72.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	A	**INTRODUCTION** (Part 2 of 3)	This guide contains several appendices that address the objectives stated above. Appendix A contains definitions of pertinent terms. Appendix B discusses determination of the probabilistic ground motion level and controlling earthquakes and the development of a seismic hazard information base, Appendix C discusses site-specific geological, seismological, and geophysical investigations. Appendix D describes a method to confirm the adequacy of existing seismic sources and source parameters as the basis for determining the DE for a site. Appendix E describes procedures for determination of the DE. The basis for the reference probability, an annual probability of exceeding the Design Earthquake Ground Motion (DE), which is stated in Regulatory Position 3.4, is discussed in “Selection of the Design Earthquake Ground Motion Reference Probability” (Ref. 1).
Design	A	**INTRODUCTION** (Part 3 of 3)	This guide applies to the design basis of both dry cask storage Independent Spent Fuel Storage Installations (ISFSIs) and U.S. Department of Energy monitored retrievable storage (MRS) installations, because these facilities are similar in design. The reference probability in Regulatory Position 3.4 does not apply to wet storage because applications for this means of storage are not expected, and it is not cost-effective to allocate resources to deveop the technical bases for such an expansion of the rulemaking. This guide is consistent with Regulatory Guide 1.165 (Ref. 2), but it has been modified to reflect ISFSI and MRS applications, experience in the use of the dry cask storage methodology, and advancements in the state of knowledge in ground motion modeling (for example, the use of spectral ground motion levels at different frequencies, based on NUREG/CR-6728. The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-3011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	B	**DISCUSSION** (Part 1 of 6)	**BACKGROUND** A PSHA has been identified in 10 CFR 72.103 as a means to determine the DE for the seismic design of an ISFSI or MRS facility. Furthermore, the rule recognizes that the nature of uncertainty and the appropriate approach to account for it depend on the tectonic environment of the site and on properly characterizing parameters input to the PSHA, such as seismic sources, the recurrence of earthquakes within a seismic source, the maximum magnitude of earthquakes within a seismic source, engineering estimation of earthquake ground motion, and the level of understanding of the tectonics. Therefore, methods other than probabilistic methods, such as sensitivity analyses, may be adequate to account for uncertainties. Every site and storage facility is unique, and therefore requirements for analysis and investigations vary. It is not possible to provide procedures for addressing all situations. In cases that are not specifically addressed in this guide, prudent and sound engineering judgment should be exercised. PSHA methodology and procedures were developed during the past 20 to 25 years specifically for evaluation of seismic safety of nuclear facilities. Significant experience has been gained by applying this methodology at nuclear facility sites, both reactor and non-reactor sites, throughout the United States. The Western United States (WUS) (west of approximately 104^{o} west longitude) and the Central and Eastern United States (CEUS) (Refs. 4, 5) have fundamentally different tectonic environments and histories of tectonic deformation. Results of the PSHA methodology applications identified the need to vary the fundamental PSHA methodology application depending on the tectonic environment of a site. The experience with these applications also served as the basis for the Senior Seismic Hazard Analysis Committee guidelines for conducting a PSHA for nuclear facilities (Ref. 6).

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	B	**DISCUSSION** (Part 2 of 6)	**APPROACH** The general process to determine the DE at a new ISFSI or MRS site includes: 1. Site- and region-specific geological, seismological, geophysical, and geotechnical investigations, and; 2. A PSHA or suitable sensitivity analyses. For ISFSI sites that are co-located with existing nuclear power generating stations, unless the existing geological and seismological design criteria for the nuclear power plant (NPP) are used [§ 73.103(a)(2), § 73.103(b)], the level of effort will depend on the availability and quality of existing evaluations. In performing this evaluation, the applicant should evaluate whether new data require re-evaluation of previously accepted seismic sources, and earthquake recurrence and ground motion attenuation models.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	B	**DISCUSSION** (Part 3 of 6)	**CENTRAL AND EASTERN UNITED STATES** The CEUS is considered to be that part of the United States east of the Rocky Mountain front, or east of longitude 104^{o} west (Refs. 6, 7). To determine the DE in the CEUS, an accepted PSHA methodology with a range of credible alternative input interpretations should be used. For sites in the CEUS, the seismic hazard methods, the data developed, and seismic sources identified by Lawrence Livermore National Laboratory (LLNL) (Refs. 4, 5, 7) and the Electric Power Research Institute (EPRI) (Ref. 8) have been reviewed and are acceptable to the staff. The LLNL and EPRI studies developed data bases and scientific interpretations of available information and determined seismic sources and source characterizations for the CEUS (e.g., earthquake occurrence rates, estimates of maximum magnitude). In the CEUS, characterization of seismic sources is more problematic than in the active plate-margin region because there is generally no clear association between seismicity and known tectonic structures or near-surface geology. In general, the observed geologic structures were generated in response to tectonic forces that no longer exist and may have little or no correlation with current tectonic forces. Therefore, it is important to account for this uncertainty by the use of multiple alternative seismotectonic models.
Design	B	**DISCUSSION** (Part 4 of 6)	The identification of seismic sources and reasonable alternatives in the CEUS considers hypotheses presently advocated for the occurrence of earthquakes in the CEUS (e.g., the reactivation of favorably oriented zones of weakness or the local amplification and release of stresses concentrated around a geologic structure). In tectonically active areas of the CEUS, such as the New Madrid Seismic Zone, where geological, seismological, and geophysical evidence suggest the nature of the sources that generate the earthquakes, it may be more appropriate to evaluate those seismic sources by using procedures similar to those normally applied in the WUS.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	B	**DISCUSSION** (Part 5 of 6)	**WESTERN UNITED STATES** The WUS is considered to be that part of the United States that lies west of the Rocky Mountain front, or west of approximately 104^{o} west longitude. For the WUS, an information base of earth science data and scientific interpretations of seismic sources and source characterizations (e.g., geometry, seismicity parameters) comparable to the CEUS, as documented in the LLNL and EPRI studies (Refs. 4, 5, 7-9) does not exist. For this region, specific interpretations, on a site-by-site basis, should be applied (Refs. 10, 11). The active plate-margin regions include, for example, coastal California, Oregon, Washington, and Alaska. For the active plate-margin regions, where earthquakes can often be correlated with known faults that have experienced repeated movements at or near the ground surface during the Quaternary, tectonic structures should be assessed for their earthquake and surface deformation potential. In these regions, at least three types of sources may exist: (1) faults that are known to be at or near the surface; (2) buried (blind) sources that may often be manifested as folds at the earth's surface; and (3) subduction zone sources, such as those in the Pacific Northwest. The nature of surface faults can be evaluated by conventional surface and near-surface investigation techniques to assess orientation, geometry, sense of displacements, length of rupture, quaternary history, etc. Buried (blind) faults are often associated with surficial deformation such as folding, uplift, or subsidence. The surface expression of blind faulting can be detected by mapping the uplifted or down-dropped geomorphological features or stratigraphy, survey leveling, and geodetic methods. The nature of the structure at depth can often be evaluated by deep core borings and geophysical techniques.

Category	REG 3.73 Section	REG Guide 3.53 <u>SECTION TITLE</u> Subsection Title	Criteria Description
Design	B	**<u>DISCUSSION</u>** <u>(Part 6 of 6)</u>	Continental U.S. subduction zones are located in the Pacific Northwest and Alaska. Seismic sources associated with subduction zones are sources within the overriding plate, on the interface between the subducting and overriding lithospheric plates, and in the interior of the downgoing oceanic slab. The characterization of subduction zone seismic sources includes consideration of the three-dimensional geometry of the subducting plate, rupture segmentation of subduction zones, geometry of historical ruptures, constraints on the up-dip and down-dip extent of rupture, and comparisons with other subducting plates worldwide. The Basin and Range region of the WUS, and to a lesser extent the Pacific Northwest and the Central United States, exhibit temporal clustering of earthquakes. Temporal clustering is best exemplified by the rupture histories within the Wasatch fault zone in Utah and the Meers fault in central Oklahoma, where several large late Holocene coseismic faulting events occurred at relatively close intervals (hundreds to thousands of years) that were preceded by long periods of quiescence that lasted thousands to tens of thousands of years. Temporal clustering should be considered in these regions or wherever paleoseismic evidence indicates that it has occurred. The non-Poissonian models to account for temporal clustering have not been developed sufficiently to be able to provide a specific guidance. Therefore, judgement would have to be exercised in considering the temporal clustering in the PSHA.
Design	C	**<u>REGULATORY POSITION</u>**	No text provided.
Design	C.1	**<u>REGULATORY POSITION</u>** **GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL**	No text provided.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.1.1	**REGULATORY POSITION** **GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.1	Comprehensive geological, seismological, geophysical, and geotechnical investigations of the site area and region should be performed. For ISFSIs co-located with existing NPPs, the existing technical information should be used, along with all other available information, to plan and determine the scope of additional investigations. The investigations described in this regulatory guide are performed primarily to gather data pertinent to the safe design and construction of the ISFSI or MRS. Appropriate geological, seismological, and geophysical investigations are described in Appendix C to this guide. Geotechnical investigations are described in Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants" (Ref. 12), and NUREG/CR-5738 (Ref. 13). Another important purpose for the site-specific investigations is to determine whether there are any new data or interpretations that are not adequately incorporated into the existing PSHA data bases. Appendix D describes a method for assessing the impact of new information, obtained during the site-specific investigations on the data bases used for the PSHA. Investigations should be performed at four levels, with the degree of detail based on distance from the site, the nature of the Quaternary tectonic regime, the geological complexity of the site and region, the existence of potential seismic sources, the potential for surface deformation, etc. A more detailed discussion of the areas and levels of investigations and the bases for them are presented in Appendix C to this regulatory guide. General guidelines for the levels of investigation are provided in subsections 1.1.1 through 1.1.4.
Design	C.1.1.1	**REGULATORY POSITION** **GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.1.1	Regional geological and seismological investigations are not expected to be extensive nor in great detail, but should include literature reviews, the study of maps and remote sensing data, and, if necessary, ground-truth reconnaissances conducted within a radius of 320 kilometers (km) (200 miles) of the site to identify seismic sources (seismogenic and capable tectonic sources).

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.1.1.2	**REGULATORY POSITION GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.1.2	Geological, seismological, and geophysical investigations should be carried out within a radius of 40 km (25 miles) in greater detail than the regional investigations, to identify and characterize the seismic and surface deformation potential of any capable tectonic sources and the seismic potential of seismogenic sources, or to demonstrate that such structures are not present. Sites with capable tectonic or seismogenic sources within a radius of 40 km (25 miles) may require more extensive geological and seismological investigations and analyses [similar in detail to investigations and analysis usually preferred within an 8-km (5-mile) radius].
Design	C.1.1.3	**REGULATORY POSITION GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.1.3	Detailed geologic, seismological, geophysical, and geotechnical investigations should be conducted within a radius of 8 km (5 miles) of the site, as appropriate, to evaluate the potential for tectonic deformation at or near the ground surface and to assess the transmission characteristics of soils and rocks in the site vicinity. Sites in the CEUS where geologically young or recent tectonic activity is not present may be investigated in less detail. Methods for evaluating the seismogenic potential of tectonic structures and geological features developed in Reference 13 should be followed.
Design	C.1.1.4	**REGULATORY POSITION GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.1.4	Very detailed geological, geophysical, and geotechnical engineering investigations should be conducted within the site [radius of approximately 1 km (0.5 miles)] to assess specific soil and rock characteristics, as described in Reference 12, updated with NUREG/CR-5738 (Ref. 13).

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.1.2	**REGULATORY POSITION GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.2	The areas of investigation may be expanded beyond those specified above in regions that include capable tectonic sources, relatively high seismicity, or complex geology, or in regions that have experienced a large, geologically recent earthquake.
Design	C.1.3	**REGULATORY POSITION GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.3	Data sufficient to clearly justify all assumptions and conclusions should be presented. Because engineering solutions cannot always be satisfactorily demonstrated for the effects of permanent ground displacement, it is prudent to avoid a site that has a potential for surface or nearsurface deformation. Such sites normally will require extensive additional investigations.
Design	C.1.4	**REGULATORY POSITION GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS** No Title for subsection 1.4	For the site and for the area surrounding the site, lithologic, stratigraphic, hydrologic, and structural geologic conditions should be characterized. The investigations should include the measurement of the static and dynamic engineering properties of the materials underlying the site as well as an evaluation of the physical evidence concerning the behavior during prior earthquakes of the surficial materials and the substrata underlying the site. The properties needed to assess the behavior of the underlying material during earthquakes should be measured. These include the potential for liquefaction and the characteristics of the underlying material in transmitting earthquake ground motions to the foundations of the facility (such as seismic wave velocities, density, water content, porosity, elastic moduli, and strength).
Design	C.2	**REGULATORY POSITION GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD**	No text provided.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.2.1	**REGULATORY POSITION** **GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD** No Title for subsection 2.1	For sites in the CEUS, the EPRI or LLNL PSHA methodologies and data bases may be used to determine the DE provided the site seismic sources that were not included in these data bases are appropriately characterized and provided sensitivity analyses are performed to assess their significance to the seismic hazard estimate. The results of the investigation discussed in Regulatory Position 1 should be used, in accordance with Appendix D, to determine whether the LLNL or EPRI seismic sources and their characterization should be updated. The guidance in Regulatory Positions 2.2 and 2.3 and the methods in Appendix C of this guide may be used if additional seismic sources are to be developed as a result of investigations.
Design	C.2.2	**REGULATORY POSITION** **GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD** No Title for subsection 2.2	When the LLNL or EPRI PSHA methods are not used or are not applicable, the guidance in Regulatory Position 2.3 should be used for identification and characterization of seismic sources. The uncertainties in the characterization of seismic sources should be addressed as appropriate. “Seismic sources” is a general term referring to both seismogenic sources and capable tectonic sources. The main distinction between these two types of seismic sources is that a seismogenic source would not cause surface displacement, but a capable tectonic source causes surface or near-surface displacement. Identification and characterization of seismic sources should be based on regional and site geological and geophysical data, historical and instrumental seismicity data, the regional stress field, and geological evidence of prehistoric earthquakes. Investigations to identify seismic sources are described in Appendix C. The bases for the identification of seismic sources should be described. A general list of characteristics to be evaluated for seismic sources is presented in Appendix C.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.2.3	**REGULATORY POSITION** **GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD** No Title for subsection 2.3	As part of the seismic source characterization, the seismic potential for each source should be evaluated. Typically, characterization of the seismic potential consists of four equally important elements: 1. Selection of a model for the spatial distribution of earthquakes in a source. 2. Selection of a model for the temporal distribution of earthquakes in a source. 3. Selection of a model for the relative frequency of earthquakes of various magnitudes, including an estimate for the largest earthquake that could occur in the source under the current tectonic regime. 4. A complete description of the uncertainty. For example, in the LLNL study, a truncated exponential model was used for the distribution of magnitudes given that an earthquake has occurred in a source. A stationary Poisson process is used to model the spatial and temporal occurrences of earthquakes in a source. For a general discussion of evaluating the earthquake potential and characterizing the uncertainty of a seismic source, refer to Reference 5.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.2.3.1	**REGULATORY POSITION GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD** No Title for subsection 2.3.1	For sites in the CEUS, when the LLNL or EPRI method is not used or not applicable (such as in the New Madrid, MO; Charleston, SC; and Attica, NY, seismic zones), it is necessary to evaluate the seismic potential for each source. The seismic sources and data that have been accepted by NRC in past licensing decisions may be used, along with the data gathered from the investigations carried out as described in Regulatory Position 1. Generally, the seismic sources for the CEUS are area sources because there is uncertainty about the underlying causes of earthquakes. This uncertainty is caused by a lack of active surface faulting, a low rate of seismic activity, or a short historical record. The assessment of earthquake recurrence for CEUS area sources commonly relies heavily on catalogs of historic earthquakes. Because these catalogs are incomplete and cover a relatively short period of time, the earthquake recurrence rate cannot be estimated reliably. Considerable care must be taken to correct for incompleteness and to model the uncertainty in the rate of earthquake recurrence. To completely characterize the seismic potential for a source, it is also necessary to estimate the largest earthquake magnitude that a seismic source is capable of generating under the current tectonic regime. This estimated magnitude defines the upper bound of the earthquake recurrence relationship. Primary methods for assessing maximum earthquakes for area sources usually include a consideration of the historical seismicity record, the pattern and rate of seismic activity, the Quaternary (2 million years and younger) characteristics of the source, the current stress regime (and how it aligns with known tectonic structures), paleoseismic data, and analogs to sources in other regions considered tectonically similar to the CEUS. Because of the shortness of the historical catalog and low rate of seismic activity, considerable judgment is needed. It is important to characterize the large uncertainties in the assessment of the earthquake potential (Refs. 6, 8).

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.2.3.2	**REGULATORY POSITION GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD** No Title for subsection 2.3.2	For sites located within the WUS, earthquakes can often be associated with known tectonic structures with a high degree of certainty. For faults, the earthquake potential is related to the characteristics of the estimated future rupture, such as the total rupture area, the length, or the amount of fault displacement. The following empirical relations can be used to estimate the earthquake potential from fault behavior data and also to estimate the amount of displacement that might be expected for a given magnitude. It is prudent to use several of the following different relations to obtain an estimate of the earthquake magnitude. • Surface rupture length versus magnitude (Refs. 14-18); • Subsurface rupture length versus magnitude (Ref. 19); • Rupture area versus magnitude (Ref. 20); • Maximum and average displacement versus magnitude (Ref. 19); and • Slip rate versus magnitude (Ref. 21). When such correlations as in References 15-21 are used, the earthquake potential is often evaluated as the mean of the distribution. The difficult issue is the evaluation of the appropriate rupture dimension to be used. This is a judgmental process based on geological data for the fault in question and the behavior of other regional fault systems of the same type. In addition to maximum magnitude, the other elements of the recurrence model are generally obtained using catalogs of seismicity, fault slip rate, and other data. All the sources of uncertainty must be appropriately modeled.
Design	C.2.3.3	**REGULATORY POSITION GEOLOGICAL, SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD** No Title for subsection 2.3.3	For sites near subduction zones, such as in the Pacific Northwest and Alaska, the maximum magnitude must be assessed for subduction zone seismic sources. Worldwide observations indicate that the largest known earthquakes are associated with the plate interface, although intraslab earthquakes may also have large magnitudes. The assessment of plate interface earthquakes can be based on estimates of the expected dimensions of rupture or analogies to other subduction zones worldwide.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.3	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES**	A PSHA should be performed for the site, since it allows the use of multiple models to estimate the likelihood of earthquake ground motions occurring at a site and systematically takes into account uncertainties that exist in various parameters (such as seismic sources, maximum earthquakes, and ground motion attenuation). Alternative hypotheses are considered in a quantitative fashion in a PSHA. Alternative hypotheses can also be used to evaluate the sensitivity of the hazard to the uncertainties in the significant parameters and to identify the relative contribution of each seismic source to the hazard. Subsections 3.1 through 3.6 describe a procedure that is acceptable to the NRC staff for performing a PSHA.
Design	C.3.1	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES** No Title for subsection 3.1	Perform regional and site geological, seismological, and geophysical investigations in accordance with Regulatory Position 1 and Appendix C.
Design	C.3.2	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES** No Title for subsection 3.2	For CEUS sites, perform an evaluation of LLNL or EPRI seismic sources, in accordance with Appendix D, to determine whether they are consistent with the site-specific data gathered in Regulatory Position 1 or require updating. The PSHA should only be updated if the new information indicates that the current version significantly overestimates the hazard and there is a strong technical basis that supports such a revision. In most cases, limited-scope sensitivity studies should be sufficient to demonstrate that the existing data base in the PSHA envelops the findings from site-specific investigations. In general, significant revisions to the LLNL and EPRI data base are to be undertaken only periodically (every 10 years), or when there is an important new finding or occurrence. Any significant update should follow the guidance of Reference 5.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.3.3	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES** No Title for subsection 3.3	For CEUS sites only, perform the LLNL or EPRI PSHA using original or updated sources as determined in Regulatory Position 2. For sites in the WUS, perform a site-specific PSHA (Ref. 6). The ground motion estimates should be made for rock conditions in the free-field or by assuming hypothetical rock conditions for a non-rock site to develop the seismic hazard information base discussed in Appendix B.
Design	C.3.4	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES** No Title for subsection 3.4	Using the mean reference probability of 5E-4/yr (Ref. 1), determine the 5 percent of critically damped mean spectral ground motion levels for 1 Hz ($S_{a,1}$) and 10 Hz ($S_{a,10}$).
Design	C.3.5	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES** No Title for subsection 3.5	Deaggregate the mean probabilistic hazard characterization in accordance with Appendix B to determine the controlling earthquakes (i.e., magnitudes and distances) and document the hazard information base as described in Appendix B.
Design	C.3.6	**REGULATORY POSITION** **PROBABILISTIC SEISMIC HAZARD ANALYSIS PROCEDURES** No Title for subsection 3.6	Instead of the controlling earthquake approach described in Regulatory Positions 3.4 and 3.5, an alternative approach is as follows: a. Using the mean reference probability of 5E-4/yr (Ref. 1), determine the 5 percent of critically damped mean spectral ground motion levels for a sufficient number of frequencies significant to an ISFSI or an MRS facility; and b. Envelope the ground motions to determine the DE.
Design	C.4	**REGULATORY POSITION** **PROCEDURES FOR DETERMINING THE DE**	After completing the PSHA (see Regulatory Position 3) and determining the controlling earthquakes, the procedures described in subsections 4.1 through 4.5 should be used to determine the DE. Appendix E contains an additional discussion of some of the characteristics of the DE.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	C.4.1	**REGULATORY POSITION PROCEDURES FOR DETERMINING THE DE** No Title for subsection 4.1	With the controlling earthquakes determined as described in Regulatory Position 3, and by using the procedures in Revision 3 of Reference 22 (which may include the use of ground motion models not included in the PSHA but that are more appropriate for the source, region, and site under consideration, or which represent the latest scientific development), develop 5 percent of critical damping response spectral shapes for the actual or assumed rock conditions. The same controlling earthquakes are also used to derive vertical response spectral shapes.
Design	C.4.2	**REGULATORY POSITION PROCEDURES FOR DETERMINING THE DE** No Title for subsection 4.2	Use $S_{a,10}$ to scale the response spectrum shape corresponding to the controlling earthquake. If there is a controlling earthquake for $S_{a,1}$, determine that the Sa,10 scaled response spectrum also envelopes the ground motion spectrum for the controlling earthquake for $S_{a,1}$. Otherwise, modify the shape to envelope the low-frequency spectrum or use two spectra in the following steps. For a rock site, go to Regulatory Position 4.4.
Design	C.4.3	**REGULATORY POSITION PROCEDURES FOR DETERMINING THE DE** No Title for subsection 4.3	For non-rock sites, perform a site-specific soil amplification analysis considering uncertainties in site-specific geotechnical properties and parameters to determine response spectra at the free ground surface in the free field for the actual site conditions. Procedures described in Appendix C of this guide and in Reference 22 may be used to perform soil-amplification analyses.
Design	C.4.4	**REGULATORY POSITION PROCEDURES FOR DETERMINING THE DE** No Title for subsection 4.4	Compare the smooth DE spectrum or spectra used in design at the free field with the spectrum or spectra determined in Regulatory Position 2 for rock sites, or determined in Regulatory Position 3 for the non-rock sites, to assess the adequacy of the DE spectrum or spectra.
Design	C.4.5	**REGULATORY POSITION PROCEDURES FOR DETERMINING THE DE** No Title for subsection 4.5	To obtain an adequate DE based on the site-specific response spectrum or spectra, develop a smooth spectrum or spectra or use a standard broad band shape that envelopes the spectra of Regulatory Position 2 or 3.

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	D	**IMPLEMENTATION**	The purpose of this section is to provide guidance to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. Except when the applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, this guide will be used in the evaluation of applications for new dry cask ISFSI or MRS licenses submitted after October 16, 2003. This guide will not be used in the evaluation of an application for dry cask ISFSI or MRS licenses submitted before October 16, 2003.

APPENDICES

Only the title and a brief description of the contents of the appendix is provided below.

Category	Appen-dix	Appendix Title	Brief Description of Appendix
Design	App A	**DEFINITIONS**	Appendix A provides definitions for seismic terminology used in Regulatory Guide 3.73.
Design	App B	**DETERMINATION OF CONTROLLING EARTHQUAKES AND DEVELOPMENT OF SEISMIC HAZARD INFORMATION BASE**	Appendix B elaborates on the steps described in Regulatory Position 3 of this regulatory guide to determine the controlling earthquakes used to define the design earthquake ground motion (DE) at the site and to develop a seismic hazard information base.
Design	App C	**GEOLOGICAL, SEISMOLOGICAL, AND GEOPHYSICAL INVESTIGATIONS TO CHARACTERIZE SEISMIC SOURCES**	Appendix C provides details of the geological, seismological, and geophysical characteristics of a site that need to be investigated for probabilistic seismic hazard analyses (PSHA). The primary objective of geological, seismological, and geophysical investigations is to develop an up-to-date, site-specific earth science data base that supplements existing information so that a PSHA of a site can be properly performed.

A-2. DETAILED REQUIREMENTS OF Reg Guide 3.73

Category	REG 3.73 Section	REG Guide 3.53 SECTION TITLE Subsection Title	Criteria Description
Design	App D	**PROCEDURE FOR THE EVALUATION OF NEW GEOSCIENCES INFORMATION OBTAINED FROM THE SITE-SPECIFIC INVESTIGATIONS**	Appendix D provides methods acceptable to the U. S. Nuclear Regulatory Commission staff for assessing the impact of new information obtained during site-specific investigations on the data base used for the probabilistic seismic hazard analyses (PSHA).
Design	App E	**PROCEDURE TO DETERMINE THE DESIGN EARTHQUAKE GROUND MOTION**	Appendix E elaborates on Regulatory Position 4 of this guide, which describes an acceptable procedure to determine the design earthquake ground motion (DE).

Categories:
Construction
Decommissioning
Design
General
Licensing
Operaton
Quality Assurance

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
DR	*Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*	ANSI/ANS 8.1	N/A	April 15, 2014	American National Standards Institute, American Nuclear Society	N/A		ANSI Standard
DR	*Design Fabrication, and Construction of Freight Cars*	M-1001 (Manual of Standards and Recommended Practices Section C, Part II)	N/A	April 13, 2011	Association of American Railroads (AAR)	N/A	R 1.0.3	
DR	*Radioactive Materials - Leakage Tests on Packages for Shipment*	N14.5	N/A	January 1, 1997	American National Standards Institute	N/A		
DR	*Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more*	N14.6	N/A	201X	American National Standards Institute	N/A		ANSI Standard is being developed by Institute of Nuclears Materials Management
DR	*Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)*	NOG-1	N/A	2015	American Society of Mechanical Engineers	N/A		ASME Standard
DR	*Quality Assurance Requirements for Nuclear Facility Applications*	NQA-1	N/A	2015	American Society of Mechanical Engineers	N/A		ASME Standard
NR	*Transporting Spent Fuel: Protection Provided Against Severe Highway and Railroad Accidents*	NUREG/BR-0111	N/A	March 1987	U.S. Nuclear Regulatory Commission	William R. Lahs		
NR	*Safety of Spent Fuel Transportation*	NUREG/BR-0292	N/A	March 2003	U.S. Nuclear Regulatory Commission	N/A		NRC brochure.
NR	*Shock and Vibration Environments for a Large Shipping Container during Truck Transport Part II)*	NUREG/CR-0128	N/A	May 1978	U.S. Nuclear Regulatory Commission	Clifford F. Magnuson		Prepared by Sandia Laboratories as document SAND78-0337, RT.
NR	*SCALE Ver 4.4: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation (Control Modules C4, C6;Volume 1, Part 1), (Control Modules D1, S1-S5, H1; Volume 1, Part 2), (Functional Models F1 - F8; Volume 2, Part 1), (Functional Models F9 - F11; Volume 2, Part 2), (Functional Models F16 - F17; Volume 2, Part 3), (Miscellaneous M1 - M17; Volume 3)*	NUREG/CR-0200	6	May 2000	U.S. Nuclear Regulatory Commission	N/A		Prepared by Oak Ridge National Laboratory as document number ORNLINUREG/CSD-2/R6.
NR	*An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers*	NUREG/CR-0481	N/A	September 1978	U.S. Nuclear Regulatory Commission	Henry J. Rack, Gerald A. Knorovsky		Prepared by Sandia Laboratories as document SAND77-1872, R-7.
NR	*Fission Product Release from Highly Irradiated LWR Fuel*	NUREG/CR-0722	N/A	February 1980	U.S. Nuclear Regulatory Commission	R. A. Lorenz J. L. Collins A. P. Malinauskas O. L. Kirkland R. L. Towns		Prepared by Oak Ridge National Laboratory as document number ORNL/NUREG/TM-287/R1.
NR	*SKYSHINE II Procedure: Calculation of the Effects of Structure Design Assessment on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air (Main Report and Supplement 1)*	NUREG/CR-0781	N/A	May 1979 (Main Report) January 1981 (Supplement 1)	U.S. Nuclear Regulatory Commission	C. M. Lampley		Prepared by Radiation Research Associates, Fort Worth, TX as document number RRA-T7901 (Main Report) and RRA-T8008 (Supplement 1).

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*Approaches to Acceptable Risk: A Critical Guide*	NUREG/CR-1614	N/A	December 1980	U.S. Nuclear Regulatory Commission	Baruch Fischhoff, Sarah Lichtenstein, and Paul Slovic, ORNL Ralph Keeney, Woodward-Clyde Consultants Stephen Derby, Stanford University		Prepared by Oak Ridge National Laboratory as document number ORNL/Sub-7656/1.
NR	*Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick*	NUREG/CR-1815	N/A	June 15, 1981	U.S. Nuclear Regulatory Commission	W. R. Holman, R. T. Langland		Prepared by Lawrence Livermore National Laboratory as document number UCRL-53013. NUREG-1815 provides basis for NRC Reg. Guide 7.11.
NR	*Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages, Volume 1, Volume 2, and Volume 3*	NUREG/CR-2146	N/A	November 1981 (V1) July 1983 (V2) October 1983 (V3)	U.S. Nuclear Regulatory Commission	S. R. Fields		Prepared by Hanford Engineering Development Laboratory as documents: HEDL-TME 81-15, RT (Vol 1, Quarterly Progress Report January 1, 1981 - March 31, 1981); HELD-TME 83-8, RT (Vol 2, Quartly Progress Report April 1, 1981 - June 30, 1981); and HELD 83-18, RT, (Vol 3, Final Summary Report)
NR	*Probabilistic Safety Analysis Procedures Guide*	NUREG/CR-2815	N/A	January 1984	U.S. Nuclear Regulatory Commission	I.A. Papazoglou, R.A. Bari, A.J. Buslik, R.E. Hall, D. Ilberg, P.K. Samanta, T. Teichmann, and R.W. Youngblood, BNL A. El-Bassioni, NRC, J. Fragola and E. Lofgren, Science Applications, Inc. W. Vesely, Battelle Columbus Laboratories		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-51559.
NR	*Recommended Welding Criteria For Use in the Fabrication of Shipping Containers for Radioactive Materials*	NUREG/CR-3019	N/A	March 1984	U.S. Nuclear Regulatory Commission	R. E. Monroe, H. H. Woo, R. G. Sears		Prepared by Lawrence Livermore National Laboratory as document number UCRL-53044, RM.
NR	*User's Manual for FIRIN: A Computer Code to Estimate Accidental Fire and Airborne Releases in Nuclear Fuel Cycle Facilities*	NUREG/CR-3037	N/A	February 1989	U.S. Nuclear Regulatory Commission	M.K. Chan, M.Y. Ballinger, and P.C. Owczarski		Prepared by Pacific Northwest National Laboratory as document number PNL-4532.
NR	*Measures of Risk Importance and Their Applications*	NUREG/CR-3385	N/A	July 1983	U.S. Nuclear Regulatory Commission	W.E. Vesely, T.C. Davis, R.S. Denning, and N. Saltos		Prepared by Battelle Columbus Laboratories as document number BMI-2103.
NR	*Material Transport Analysis for Accident-Induced Flow in Nuclear Facilities*	NUREG/CR-3527	N/A	October 1983	U.S. Nuclear Regulatory Commission	R.A. Martin, P.K. Tang, A.P. Harper, J.D. Novat, and W.S. Gregory		Prepared by Los Alamos National Laboratory as document number LA-9913-MS.

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*The Effect of Thermal and Irradiation Aging Simulation Procedures on Polymer Properties*	NUREG/CR-3629	N/A	April 1984	U.S. Nuclear Regulatory Commission	L.D. Bustard and E. Minor, SNL J. Chenion, F. Carlin, C. Alba, and G. Gaussens, CEA-ORIS LABRA at Saclay, France, M. LeMeur, CEA-DAS-SAF, France		Prepared by Sandia National Laboratory as document number SAND83-2651.
NR	*Accident-Induced Flow and Material Transport in Nuclear Facilities: A Literature Review*	NUREG/CR-3735	N/A	April 1984	U.S. Nuclear Regulatory Commission	J.W. Bolstad, W.S. Gregory, R.A. Martin, P.K. Tang, R.G. Merryman, J.D. Novat, and H.L. Whitmore		Prepared by Los Alamos National Laboratory as document number LA-10079-MS.
NR	*A Study on Ductile and Brittle Failure Design Criteria for Ductile Cast Iron Spent-Fuel Shipping Containers*	NUREG/CR-3760			U.S. Nuclear Regulatory Commission	M.W. Schwartz		Prepared by Lawrence Livermore National Laboratory as document number TI84-014381.
NR	*Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than Four Inches Thick*	NUREG/CR-3826	N/A	April 1984	U.S. Nuclear Regulatory Commission	M.W. Schwartz		Prepared by Lawrence Livermore National Laboratory as document number UCRL 53538.
NR	*Fabrication Criteria for Shipping Containers*	NUREG/CR-3854	N/A	March 1985	U.S. Nuclear Regulatory Commission	L.E. Fischer, W. Lai		Prepared by Lawrence Livermore National Laboratory as document number UCRL-53544.
NR	*Methods for Impact Analysis of Shipping Containers*	NUREG/CR-3966	N/A	November 1987	U.S. Nuclear Regulatory Commission	T.A. Nelson, R. C. Chun		Prepared by Lawrence Livermore National Laboratory as document number UCID-20639.
NR	*A Study on Fabrication Criteria for Ductile Cast Iron Spent-Fuel Shipping Containers*	NUREG/CR-4363	N/A	February 1986	U.S. Nuclear Regulatory Commission	M.W. Schwartz		Prepared by Lawrence Livermore National Laboratory as document number UCRL 53662.
NR	*Tornado Climatology of the Contiguous United States*	NUREG/CR-4461	2	February 2007	U.S. Nuclear Regulatory Commission	J.V. Ramsdell, Jr. and J.P. Rishel		Prepared by Pacific Northwest National Laboratory as document number PNNL-15112.
NR	*Design Features for Enhancing International Safeguards of Away-from- Reactor Dry Storage for Spent LWR Fuel*	NUREG/CR-4517	N/A	October 1986	U.S. Nuclear Regulatory Commission	F.P. Roberts and N.L. Harms		Prepared by Pacific Northwest National Laboratory as document number PNL-5321.
NR	*SCANS, A Microcomputer Based Analysis System for Shipping Cask Design Review*	NUREG/CR-4554	2	March 1998	U.S. Nuclear Regulatory Commission	G. C. Mok, G. R. Thomas, M. A. Gerhard, D. J. Trummer, G. L. Johnson		Vol. 1 of User's Manual to Shipping Cask Analysis System (SCANS) Version 3a. Prepared by Lawrence Livermore National Laboratory as document number UCID-20674.
NR	*FIRAC User's Manual: A Computer Code to Simulate Fire Accidents in Nuclear Facilities*	NUREG/CR-4561	N/A	April 1986	U.S. Nuclear Regulatory Commission	B.D. Nichols and W.S. Gregory		Prepared by Los Alamos National Laboratory as document number LA-10678-M.

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*Guide for Preparing Operating Procedures for Shipping Packages*	NUREG/CR-4775	N/A	December 1998	U.S. Nuclear Regulatory Commission	M. C. Witte		Prepared by Lawrence Livermore National Laboratory as document number UCID-20820.
NR	*Shipping Container Response to Severe Highway and Railway Accident Conditions*	NUREG/CR-4829	N/A	February 1987	U.S. Nuclear Regulatory Commission	L. E. Fischer, C. K. Chou, M. A. Gerhard, C. Y. Kimura, R. W. Martin, R. W. Mensing, M. E.		Vol. 2. Prepared by Lawrence Livermore National Laboratory as document number UCID-20733.
NR	*Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses Within 10 CFR Part 71*	NUREG/CR-5342	N/A	July 1998	U.S. Nuclear Regulatory Commission	C. V. Parks, C. M. Hopper, and J. J. Lichtenwalter		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-13607.
NR	*Recommendations for Resolution of Public Comments on USI A-40, "Seismic Design Criteria"*	NUREG/CR-5347	N/A	May 1989	U.S. Nuclear Regulatory Commission	A.J. Philippacopoulos		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52191.
NR	*Elements of an Approach to Performance-Based Regulatory Oversight*	NUREG/CR-5392	N/A	January 1999	U.S. Nuclear Regulatory Commission	R.W. Youngblood, R.N.M. Hunt, E.R. Schmidt J. Bolin, F. Dombek, and D. Prochnow		Prepared by SCIENTECH, Inc. as document number SCIE-NRC-373-98.
NR	*Predicting the Pressure Dimension Flow of Gasses Through Micro-Capillaries and Micro-Orifices*	NUREG/CR-5403	N/A	November 18, 1994	U.S. Nuclear Regulatory Commission	B. L. Anderson, R. W. Carlson, L. E. Fischer		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-118245.
NR	*Engineering Drawings for 10 CFR Part 71 Package Approvals*	NUREG/CR-5502	N/A	May 1998	U.S. Nuclear Regulatory Commission	M. K. Sheaffer, G. R. Thomas, R. K. Dann, E. W. Russell		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-130438.
NR	*Residual Radioactive Contamination From Decommissioning: User's Manual DandD Version 2.1 (Volume 2)*	NUREG/CR-5512	N/A	April 2001	U.S. Nuclear Regulatory Commission	K. McFadden, Sigma Software LLC D.A. Brosseau, and W.E.		Sigma Software, LLC and Gram, Inc worked under contract to Sandia National Laboratories on
NR	*AUTOCASK: A Microcomputer Based System for Shipping Cask Design Review Analysis*	NUREG/CR-5657	N/A	April 1995	U.S. Nuclear Regulatory Commission	M. A. Gerhard, S. C. Sommer		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-118766.
NR	*Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages*	NUREG/CR-5661	N/A	April 1997	U.S. Nuclear Regulatory Commission	H. R. Dyer, C. V. Parks		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-11936.
NR	*Technical Bases for Regulatory Guide for Soil Liquefaction*	NUREG/CR-5741	N/A	March 2000	U.S. Nuclear Regulatory Commission	J. P. Koester, M. K. Sharp, and M. E. Hynes		Prepared by U.S. Army Corps of Engineers, 3909 Halls Ferry Road Vicksburg, MS 39180-6199
NR	*Stress Analysis of Closure Bolts for Shipping Casks*	NUREG/CR-6007	N/A	April 1992	U.S. Nuclear Regulatory Commission	G. C. Mok and L. E. Fischer, LLNL S. T. Hsu, Kaiser Engineering		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-110637.
NR	*CASKS (Computer Analysis of Storage casKS): A microcomputer based analysis system for storage cask design review. User`s manual to Version 1b (including program reference)*	NUREG/CR-6242	N/A	February 1995	U.S. Nuclear Regulatory Commission	T.F. Chen, M.A.; Gerhard, D.J.Trummer, G.LJohnson, and G.C. Mok		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID--117418.

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NR	*Multidisciplinary Framework for Human Reliability Analysis with an Application to Errors of Commission and Dependency*	NUREG/CR-6265	N/A	August 1995	U.S. Nuclear Regulatory Commission	M.T. Barriere and W.J. Luckas, Jr., BNL, J. Wreathall, JW&Co., S.E. Cooper, SAlC, D.C. Bley, PLG, A. Ramey-Smith, NRC		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52431.
NR	*Quality Assurance Inspections for Shipping and Storage Containers*	NUREG/CR-6314	N/A	April 1996	U.S. Nuclear Regulatory Commission	H.M. Stromberg, G.D. Roberts, and J.H. Bryce		Prepared by Idaho National (Engineering) Laboratory as document number INEL-95/0061.
NR	*Buckling Analysis of Spent Fuel Basket*	NUREG/CR-6322	N/A	May 1995	U.S. Nuclear Regulatory Commission	A. S. Lee, S. E. Bumpas		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-119697.
NR	*A Technique for Human Error Analysis (ATHEANA)*	NUREG/CR-6350	N/A	May 1996	U.S. Nuclear Regulatory Commission	S.E. Cooper, SAIC A.M. Ramey-Smith, NRC J. Wreathall, and D.C. Bley, WWG		Prepared by Brookhaven National Laboratory as document BNL-NUREG-52467 with support from subcontractors:
NR	*Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*	NUREG/CR-6361	N/A	March 1997	U.S. Nuclear Regulatory Commission	J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, D. M. Hopper		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-13211.
NR	*Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts (Main Report, Volume 1) (Appendices, Volume 2)*	NUREG/CR-6372	N/A	April 1997	U.S. Nuclear Regulatory Commission	R.J. Budnitz (Chairman), G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris		Senior Seismic Hazard Analysis Committee (SSHAC) under contract to LLNL to create document UCRL-ID- 122160.
NR	*Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*	NUREG/CR-6407	N/A	February 1996	U.S. Nuclear Regulatory Commission	J. W. McConnell, Jr., A. L. Ayers, M. J. Tyacke		Prepared by Idaho National (Engineering) Laboratory as document number INEL-95/0551.
NR	*Nuclear Fuel Cycle Facility Accident Analysis Handbook*	NUREG/CR-6410	N/A	March 1998	U.S. Nuclear Regulatory Commission	N/A		Prepared by Science Applications International Corporation (SAIC), Reston, Virginia
NR	*A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants*	NUREG/CR-6451	N/A	August 1997	U.S. Nuclear Regulatory Commission	R. J. Travis, R. E. Davis, E. J. Grove, and M. A. Azarm		Prepared by Brookhaven National Laboratory as document BNL-NUREG-52498.
NR	*Containment Analysis for Type B Packages Used to Transport Various Contents*	NUREG/CR-6487	N/A	November 1996	U.S. Nuclear Regulatory Commission	B. L. Anderson, R. W. Carlson, L. E. Fischer		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-124822.

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NR	*Probabilistic Liquefaction Analysis*	NUREG/CR-6622	N/A	November 1999	U.S. Nuclear Regulatory Commission	M. E. Hynes, U.S. Army Corps of Engineers		Prepared by U.S. Army Corps of Engineers, Vicksburg, MS
NR	*Automated Seismic Event Monitoring System (Addendum 1)*	NUREG/CR-6625	N/A	September 2001	U.S. Nuclear Regulatory Commission	I. Henson, R. Wagner, and W. Rivers, Jr., Multimax, Inc.		Prepared by Multimax, Inc. Largo, MD
NR	*Reexamination of Spent Fuel Shipment Risk Estimates (Main Report, Volume1)*	NUREG/CR-6672	N/A	March 2000	U.S. Nuclear Regulatory Commission	J.L. Sprung, D.J. Ammerman, N.L. Breivik, R.J. Dukart, F.L. Kanipe, J.A. Koski, G.S. Mills, K.S. Neuhauser, H.D. Radloff, R.F. Weiner, and H.R. Yoshimura		Prepared by Sandia National Laboratory as document number SAND2000-0234.
NR	*Probabilistic Dose Analysis Using Parameter Distributions Developed For RESRAD and RESRAD-BUILD Codes*	NUREG/CR-6676	N/A	July 2000	U.S. Nuclear Regulatory Commission	S. Kamboj, D. LePoire, E. Gnanapragasam, B. M. Biwer, I. Cheng, J. Arnish, C. Yu, and S. Y. Chen, ANL		Prepared by Argone National Laboratory as document number ANL/EAD/TM- 89. See NUREG/CR-6692 for User Guide and NUREG/CR-6697.
NR	*Proposed Approach for Reviewing Changes to Risk-Important Human Actions*	NUREG/CR-6689	N/A	October 2000	U.S. Nuclear Regulatory Commission	J.C. Higgins, and J.M. O'Hara, BNL		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52598.
NR	*Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes: User Guide*	NUREG/CR-6692	N/A	November 2000	U.S. Nuclear Regulatory Commission	D. LePoire, J. Amish, E. Gnanapragasam, S. Kamboj, B.M. Biwer, J.-J. Cheng, C. Yu, and S.Y. Chen, ANL		Prepared by Argone National Laboratory as document number ANL/EAD/TM-91. See NUREG/CR-6676.
NR	*Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes (Cover – Section 3.13) (Section 4.1 – End)*	NUREG/CR-6697	N/A	December 2000	U.S. Nuclear Regulatory Commission	C. Yu, D. LePoire, E. Gnanapragasam, J. Amish, S. Kamboj, B.M. Biwer, J.-J. Cheng, A. Zielen, and S.Y. Chen, ANL		Prepared by Argone National Laboratory as document number ANL/EAD/TM-98. See NUREG/CR-6672 and NUREG/CR-6676.
NR	*Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel*	NUREG/CR-6700	N/A	January 2001	U.S. Nuclear Regulatory Commission	I.C. Gauld and J.C. Ryman, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/284.
NR	*Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel*	NUREG/CR-6701	N/A	January 2001	U.S. Nuclear Regulatory Commission	I.C. Gauld and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/277.
NR	*Limited Burnup Credit in Criticality Safety Analysis: A Comparison of ISG-8 and Current International Practice*	NUREG/CR-6702	N/A	January 2001	U.S. Nuclear Regulatory Commission	I. C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/72.
NR	*Environmental Effects of Extending Fuel Burnup Above 60 Gwd/MTU*	NUREG/CR-6703	N/A	January 2001	U.S. Nuclear Regulatory Commission	J. V. Ramsdell, Jr., C. E. Beyer, D. D. Lanning, U. P. Jenquin, R. A. Schwarz, D. L. Strenge, P.M. Daling, and R. T. Dahowski, PNNL		Prepared by Pacific Northwest National Laboratory as document number PNNL-13257.

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NR	*Environmental Assessment of Major Revision of 10 CFR Part 71*	NUREG/CR-6711	N/A	2004	U.S. Nuclear Regulatory Commission	D. Hammer and K. Blake, ICF Consulting, Inc.		Prepared by ICF Consulting, Inc, Fairfax, VA. Manuscript Completed: December 2003.
NR	*Summary and Categorization of Public Comments on the Major Revision of 10 CFR Part 71*	NUREG/CR-6712	N/A	April 2002	U.S. Nuclear Regulatory Commission	D. Hammer, K. Blake, L. Massar, S. Matheson, and E. Piendak, ICF Consulting, Inc.		Prepared by ICF Consulting, Inc, Fairfax, VA.
NR	*Regulatory Analysis of Major Revision of 10 CFR Part 71*	NUREG/CR-6713	N/A	2004	U.S. Nuclear Regulatory Commission	D. Hammer and K. Blake, ICF Consulting, Inc.		Prepared by ICF Consulting, Inc, Fairfax, VA. Manuscript Completed: December 2003.
NR	*Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks*	NUREG/CR-6716	N/A	March 2001	U.S. Nuclear Regulatory Commission	S.M. Bowman, I.C. Gauld, and J.C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/385.
NR	*Zinc-Zircaloy Interaction in Dry Storage Casks*	NUREG/CR-6732	N/A	August 2001	U.S. Nuclear Regulatory Commission	H. Tsai and Y. Yan, ANL		Prepared by Argone National Laboratory as document number ANL-01/18.
NR	*Dry Cask Storage Characterization Project - Phase 1: CASTOR V/21 Cask Opening and Examination*	NUREG/CR-6745	N/A	September 2001	U.S. Nuclear Regulatory Commission	W.C. Bare and L.D. Torgerson, INL		Prepared by Idaho National (Engineering an Environmental) Laboratory as document number INEEL/EXT-01-00183.
NR	*Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*	NUREG/CR-6747	N/A	October 2001	U.S. Nuclear Regulatory Commission	J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/306.
NR	*STARBUCS: A Prototypic SCALE Control Module for Automated Criticality Safety Analyses Using Burnup Credit*	NUREG/CR-6748	N/A	October 2001	U.S. Nuclear Regulatory Commission	I. C. Gauld and S. M. Bowman, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/33.
NR	*Parametric Study of Effect of Control Rods for PWR Burnup Credit*	NUREG/CR-6759	N/A	February 2002	U.S. Nuclear Regulatory Commission	C. E. Sanders and J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/69.
NR	*Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit*	NUREG/CR-6760	N/A	March 2002	U.S. Nuclear Regulatory Commission	C. E. Sanders and J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000-321.
NR	*Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit*	NUREG/CR-6761	N/A	March 2002	U.S. Nuclear Regulatory Commission	J.C. Wagner and C.V. Parks		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/373.
NR	*Burnup Credit PIRT Report*	NUREG/CR-6764	N/A	May 2002	U.S. Nuclear Regulatory Commission	G. H. Bidinger, R. J. Cacciapouti, J. M. Conde, P. Cousinou, T. Doering, P. Grimm, H-R Hwang, W. J. Lee, D. Marloye, R. L. Murray, J-C Neuber, M. B. Raap, J. Saptya, D. A. Thomas, S. Turner, R. E. Wilson, B. Boyack, and D. J. Diamond, BNL		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52654.

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NR	*Evaluation of Hydrologic Uncertainty Assessments for Decommissioning Sites Using Complex and Simplified Models*	NUREG/CR-6767	N/A	April 2002	U.S. Nuclear Regulatory Commission	P.D. Meyer and S. Orr, PNNL		Prepared by Pacific Northwest National Laboratory as document number PNNL-13832.
NR	*Spent Nuclear Fuel Transportation Package Performance Study Issues Report*	NUREG/CR-6768		June 2002	U.S. Nuclear Regulatory Commission	J.L. Sprung, D.J. Ammerman, J.A. Koski, R.F. Weiner		Prepared by Sandia National Laboratory as document number SAND2001-08211P.
NR	*Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Development of Hazard- & Risk-Consistent Seismic Spectra for Two Sites (Cover – Page 5-26) and (Section 6.0 – End)*	NUREG/CR-6769	N/A	April 2002	U.S. Nuclear Regulatory Commission	R. K. McGuire, Rick Engineering, Inc. W. J. Silva, Pacific Engineering & Analysis C. J. Costantino, Consultant		Performe by Risk Engineering, Inc.,Boulder, CO with subcontactors: Pacific Engineering & Analysis El Cerrito, CA and Carl J. Costantino, Consultant, Valley, NY.
NR	*Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses*	NUREG/CR-6781	N/A	January 2003	U.S. Nuclear Regulatory Commission	J.C. Wagner and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/272.
NR	*Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs*	NUREG/CR-6800	N/A	March 2003	U.S. Nuclear Regulatory Commission	J.C. Wagner, C.E. Sanders, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2002/6.
NR	*Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*	NUREG/CR-6801	N/A	March 2003	U.S. Nuclear Regulatory Commission	J.C. Wagner, M. D. DeHart, and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/273.
NR	*Recommendations for Shielding Evaluations for Transport & Storage Packages*	NUREG/CR-6802	N/A	May 2003	U.S. Nuclear Regulatory Commission	B.L. Broadhead, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2002/31.
NR	*A Comprehensive Strategy of Hydrogeologic Modeling and Uncertainty Analysis for Nuclear Facilities and Sites*	NUREG/CR-6805	N/A	July 2003	U.S. Nuclear Regulatory Commission	S.P. Neuman and P.J. Wierenga, University of Arizona		Prepared by University of Arizona, Department of Soil, Water and Environmental Science.
NR	*Strategies for Application of Isotopic Uncertainties in Burnup Credit*	NUREG/CR-6811	N/A	June 2003	U.S. Nuclear Regulatory Commission	I. C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/257.
NR	*Examination of Spent PWR Fuel Rods After 15 Years in Dry Storage*	NUREG/CR-6831	N/A	September 2003	U.S. Nuclear Regulatory Commission	R.E. Einziger, H. Tsai, and M.C. Billone, ANL, B.A. Hilton, ANL-West		Prepared by Argone National Laboratory as document number ANL-03/17.
NR	*Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks*	NUREG/CR-6835	N/A	September 2003	U.S. Nuclear Regulatory Commission	K.R. Elam, J.C. Wagner, and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2002/255.
NR	*The Technical Basis for the NRC's Guidelines for External Risk Communication*	NUREG/CR-6840	N/A	January 2004	U.S. Nuclear Regulatory Commission	L. Peterson, E. Specht, and E. Wight, WPI		Prepared by WPI, Gaithersburg, MD.
NR	*A Risk-Informed Basis for Establishing Non-Fixed Surface Contamination Limits for Spent Fuel Transportation Casks*	NUREG/CR-6841	N/A	April 2004	U.S. Nuclear Regulatory Commission	R.R. Rawl, K.F. Eckerman, and J.S. Bogard		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2003/225.

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NR	*Combined Estimation of Hydrogeologic Conceptual Model and Parameter Uncertainty*	NUREG/CR-6843	N/A	March 2004	U.S. Nuclear Regulatory Commission	P.D. Meyer, M. Ye, and K.J. Cantrell, PNNL S.P. Neuman, University of Arizona		Prepared by Pacific Northwest National Laboratory as document number PNNL-14534.
NR	*Sensitivity Analysis Applied to the Validation of the ^{10}B Capture Reaction in Nuclear Fuel Casks*	NUREG/CR-6845	N/A	August 2004	U.S. Nuclear Regulatory Commission	S. Goluoglu, K.R. Elam, B.T. Rearden, B.L. Broadhead, and C.M. Hopper, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2004/48.
NR	*Identification and Analysis of Factors Affecting Emergency Evacuations (Main Report, Volume 1) (Appendices, Volume 2)*	NUREG/CR-6864	N/A	Jnauary 2005	U.S. Nuclear Regulatory Commission	L.J. Dotson and J. Jones		Prepared by Sandia National Laboratory as document number SAND2004-5901.
NR	*Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Cask Storage Systems*	NUREG/CR-6865	N/A	February 2005	U.S. Nuclear Regulatory Commission	V.K. Luk and B.W. Spencer, SNL I.P. Lam, Earth Mechanics, Inc. R.A. Dameron, David Evans & Associates, Inc.		Prepared by Sandia National Laboratory as document number SAND2004-5794P with subontractors: Earth Mechanics, Inc, .Fountain Valley, California and David Evans & Associates, Inc., San Diego, California
NR	*Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario*	NUREG/CR-6886	2	February 2009	U.S. Nuclear Regulatory Commission	H. E. Adkins, Jr., J. M. Cuta, B. J. Koeppel, and A. D. Guzman (PNNL) C. S. Bajwa (NRC)		Prepared by Pacific Northwest National Laboratory as document number PNNL-15313.
NR	*Spent Fuel Transportation Package Response to the Caldecott Tunnel Fire Scenario*	NUREG/CR-6894	1	January 2007	U.S. Nuclear Regulatory Commission	H.E. Adkins, Jr., B.J. Koeppel, J.M. Cuta, and A.D. Guzman, PNNL C. S. Bajwa (NRC)		Prepared by Pacific Northwest National Laboratory as document number PNNL-15346.
NR	*Integrated Ground-Water Monitoring Strategy for NRC-Licensed Facilities and Sites: Logic, Strategic Approach and Discussion (Logic, Strategic Approach and Discussion, Volume 1) (Case Study Applications, Volume 2)*	NUREG/CR-6948	N/A	November 2007	U.S. Nuclear Regulatory Commission	V. Price, T. Temples, J. Tauxe, R. Hodges, and R. Falta, Advanced Environmental Solutions, LLC for V1. V. Price, T. Temples, R. Hodges, Z. Dai, D. Watkins, and J. Imrich, Advanced Environmental Solutions, LLC for V2.		Prepared by Advanced Environmental Solutions, LLC, Lexington, SC.
NR	*Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit*	NUREG/CR-6951	N/A	January 2008	U.S. Nuclear Regulatory Commission	G. Radulescu, D.E. Mueller, and J.C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2006/87.
NR	*Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*	NUREG/CR-6955	N/A	January 2008	U.S. Nuclear Regulatory Commission	J.C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2004/52.
NR	*Spent Fuel Decay Heat Measurements Performed at the Swedish Central Interim Storage Facility*	NUREG/CR-6971	N/A	February 2010	U.S. Nuclear Regulatory Commission	B.D. Murphy and I.C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2008/016.

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NR	*Validation of SCALE 5 Decay Heat Predictions for LWR Spent Nuclear Fuel*	NUREG/CR-6972	N/A	February 2010	U.S. Nuclear Regulatory Commission	I.C. Gauld, G. Ilas, B.D. Murphy, and C.F. Weber, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2008/015.
NR	*Review of Information for Spent Nuclear Fuel Burnup Confirmation*	NUREG/CR-6998	N/A	December 2009	U.S. Nuclear Regulatory Commission	B.B. Bevard, J.C. Wagner and C.V. Parks, ORNL M. Aissa, NRC		Prepared by Oak Ridge National Laboratory as document number ORNLITM-2007/229.
NR	*Technical Basis for a Proposed Expansion of Regulatory Guide 3.54 — Decay Heat Generation in an Independent Spent Fuel Storage Installation*	NUREG/CR-6999	N/A	February 2010	U.S. Nuclear Regulatory Commission	I.C. Gauld and B.D. Murphy, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2007/231.
NR	*Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel*	NUREG/CR-7012	N/A	January 2011	U.S. Nuclear Regulatory Commission	I.C. Gauld, G. Ilas, and G. Radulescu, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2010/41.
NR	*Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor*	NUREG/CR-7013	N/A	January 2011	U.S. Nuclear Regulatory Commission	G. Ilas and I.C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNLITM-2009/321.
NR	*Human Reliability Analysis-Informed Insights on Cask Drops*	NUREG/CR-7016	N/A	February 2012	U.S. Nuclear Regulatory Commission	J. Brewer, S. Hendrickson, and R. Boring, SNL S. Cooper, NRC		Prepared by Sandia National Laboratory as document number SAND2010-8463P.
NR	*Preliminary, Qualitative Human Reliability Analysis for Spent Fuel Handling*	NUREG/CR-7017	N/A	February 2012	U.S. Nuclear Regulatory Commission	Jeffrey D. Brewer and Stacey M. L. Hendrickson, SNL Paul J. Amico, SAIC Susan E. Cooper, NRC		Prepared by Sandia National Laboratory as document number SAND2010-8464P.
NR	*Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments*	NUREG/CR-7030	N/A	October 2010	U.S. Nuclear Regulatory Commission	L. Caseres and T.S. Mintz, Southwest Research Institute		Prepared by Southwest Research Institute, San Antonio, TX
NR	*Analysis of Severe Railway Accidents Involving Long Duration Fires*	NUREG/CR-7034	N/A	February 2011	U.S. Nuclear Regulatory Commission	G. Adams, T. Mintz, M. Necsoiu, and J. Mancillas, Nuclear Waste Regulatory Analyses		Prepared by Center for Nuclear Waste Regulatory Analyses, San Antonio, TX
NR	*Analysis of Severe Roadway Accidents Involving Long Duration Fires*	NUREG/CR-7035	N/A	February 2011	U.S. Nuclear Regulatory Commission	G. Adams, and T. Mintz, Nuclear Waste Regulatory Analyses		Prepared by Center for Nuclear Waste Regulatory Analyses, San Antonio, TX
NR	*Structural Materials Analyses of the Newhall Pass Tunnel Fire, 2007*	NUREG/CR-7101	N/A	June 2011	U.S. Nuclear Regulatory Commission	K. Axler, T.S. Mintz, and K. Das, Nuclear Waste Regulatory Analyses J. Huczek, Southwest Research		Prepared by Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute, San Antonio, TX.

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions*	NUREG/CR-7108	N/A	April 2012	U.S. Nuclear Regulatory Commission	G. Radulescu, I. C. Gauld, G. llas, and J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2011/509.
NR	*An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions*	NUREG/CR-7109	N/A	April 2012	U.S. Nuclear Regulatory Commission	J. M. Scaglione, D. E. Mueller, J. C. Wagner, and W. J. Marshall, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2011/514.
NR	*Performance of Metal and Polymeric O-Ring Seals in Beyond-Design-Basis Temperature Excursions*	NUREG/CR-7115	N/A	April 2012	U.S. Nuclear Regulatory Commission	Jiann C. Yang and Edward J. Hnetkovsky, National Institute of Standards and Technology, Engineering Laboratory		Prepared by National Institute of Standards and Technology, Engineering Laboratory Gaithersburg, Maryland.
NR	*Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel*	NUREG/CR-7116	N/A	November 2011	U.S. Nuclear Regulatory Commission	R. L. Sindelar, A.J. Duncan, M.E. Dupont, P.-S. Lam, M.R. Louthan, Jr., and T.E. Skidmore, SRNL		Prepared by Savannah River National Laboratory as document number SRNL-STI-2011-00005.
NR	*Experimental Studies of Reinforced Concrete Structures Under Multi-Directional Earthquakes and Design Implications*	NUREG/CR-7119	N/A	July 2013	U.S. Nuclear Regulatory Commission	N. Simos and C. H. Hofmayer, BNL		Prepared by Brookhaven National Laboratory.
NR	*Review and Prioritization of Technical Issues Related to Burnup Credit for BWR Fuel*	NUREG/CR-7158	N/A	February 2013	U.S. Nuclear Regulatory Commission	D. E. Mueller, S. M. Bowman, W. J. Marshall, and J. M. Scaglione, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2012/261.
NR	*Analysis of Experimental Data for High Burnup BWR Spent Fuel Isotopic Validation – SVEA-96 and GE14 Assembly Designs*	NUREG/CR-7162	N/A	March 2013	U.S. Nuclear Regulatory Commission	H. J. Smith, I. C. Gauld, and U. Mertyurek, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2013/18.
NR	*Thermal Analysis of Horizontal Storage Casks for Extended Storage Applications*	NUREG/CR-7191	N/A	December 2014	U.S. Nuclear Regulatory Commission	Kaushik Das, Debashis Basu, and Gary Walter, Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute		Prepared by Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute, San Antonio, TX.
NR	*Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems*	NUREG/CR-7194	N/A	Arpil 2015	U.S. Nuclear Regulatory Commission	William (B.J.) Marshall, Brian J. Ade, Stephen M. Bowman, Ian C. Gauld, Germina Ilas, Ugur Mertyurek, and Georgeta Radulescu, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2014/240.
NR	*Final Envronmental Statement on the Transportation of Radioactive Material by Air and Other Modes, Volume 1 and Volume 2*	NUREG-0170	N/A	December 1977 (V1 & V2)	U.S. Nuclear Regulatory Commission	N/A		Docket No. PR-71, 73 (40 FR 23768)
NR	*Directory of Certificates of Compliance for Radioactive Materials Packages (Certificates of Compliance, Volume 2,(Report of NRC-Approved Quality Assurance Programs for Radioactive Materials Packages, Volume 3)*	NUREG-0383	28 (V2) 24 (V3)	October 2013 (V2) January 2009 (V3)	U.S. Nuclear Regulatory Commission	N/A		Volume 1 is for internal (USNRC) use only.
NR	*Physical Protection of Shipments of Irradiated Reactor Fuel: Final Report*	NUREG-0561	2	April 2013	U.S. Nuclear Regulatory Commission	A. G. Garrett, S. L. Garrett, PNNL A. S. Giantelli, R. C. Ragland, USNRC		Prepared by Pacific Northwest National Laboratory

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NR	*Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36*	NUREG-0612	N/A	July 1980	U.S. Nuclear Regulatory Commission	Henry J. George, Task (A-36) Manager, and members of A-36 Task Force		
NR	*Functional Criteria for Emergency Response Facilities*	NUREG-0696	N/A	February 1981	U.S. Nuclear Regulatory Commission	N/A		
NR	*Human-System Interface Design Review Guidelines*	NUREG-0700	2	May 2002	U.S. Nuclear Regulatory Commission	J. M. O'Hara and W. S. Brown, BNL P. M. Lewis and J. J. Persensky, NRC		NUREG-0700, Revision 2, is divided into 3 PDF Documents: Cover – Page 200, Page 201 – Page 400, and Page 401 – End
NR	*Human Factors Engineering Program Review Model*	NUREG-0711	3	November 2012	U.S. Nuclear Regulatory Commission	J. M. O'Hara and J.C. Higgins, BNL S. A. Fleger and P. A. Pieringer, NRC		
NR	*Public Information Circular for Shipments of Irradiated Reactor Fuel*	NUREG-0725	15	May 2010	U.S. Nuclear Regulatory Commission	A. G. Garrett, S. L. Garrett, and R. G. Ostler, PNNL K. Jamgochian, NRC		
NR	*Methodology for Evaluation of Emergency Response Facilities (Draft for Comment)*	NUREG-0814	N/A	August 1981	U.S. Nuclear Regulatory Commission	S. Ramos, NRC		Draft for Comment
NR	*Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data*	NUREG-0917	N/A	July 1982	U.S. Nuclear Regulatory Commission	W. Snell		
NR	*Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment: 10 CFR Part 74, Subpart E*	NUREG-1280	1	April 1995	U.S. Nuclear Regulatory Commission	N/A		
NR	*Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities (Main Report, Volume 1), (Appendices A and B, Volume 2), (Appendices C-H, Volume 3)*	NUREG-1496	N/A	July 1997	U.S. Nuclear Regulatory Commission	N/A		NUREG-1496 is divided into 3 PDF Documents: Main Report (Volume 1), Appendices A & B (Volume 2), Appendices C-H (Volume 3)
NR	*Background as a Residual Radioactivity Criterion for Decommissioning: Appendix A to the Draft Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities (Draft Report of Comment)*	NUREG-1501	N/A	August 1994	U.S. Nuclear Regulatory Commission	A.M. Huffert, R.A. Meck, K. M. Miller, Environmental Measurements Laboratory		Draft for Comment
NR	*A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys (Interim Draft Report for Comment and Use)*	NUREG-1505	1	June 1998	U.S. Nuclear Regulatory Commission	C.V. Gogolak, Environmental Measurements Laboratory G.E. Powers, A.M. Huffert, NRC		Interim Draft Report for Comment and Use
NR	*Integrated Safety Analysis Guidance Document*	NUREG-1513	N/A	May 2001	U.S. Nuclear Regulatory Commission	R.I. Milstein, NRC		
NR	*Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*	NUREG-1520	1	May 2010	U.S. Nuclear Regulatory Commission	N/A		

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NR	*Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*	NUREG-1536	1	July 2010	U.S. Nuclear Regulatory Commission	N/A		Complements 10CFR72.
NR	*Standard Review Plan for Spent Fuel Dry Storage Facilities*	NUREG-1567	N/A	March 2000	U.S. Nuclear Regulatory Commission	N/A		Complements 10CFR72.
NR	*Standard Review Plan for Transportation Packages for Radioactive Material*	NUREG-1609	N/A	March 1999	U.S. Nuclear Regulatory Commission	N/A		Initial Report, Supplement 1 is *Standard Review Plan for Transportation Packages for MOX-Radioactive Material*, Supplement 2 is *Standard Review Plan for Transpor-tation Packages for Irradiated Tritium-Producing Burnable Absorber Rods (TPBARs)*
NR	*Standard Review Plan for Transportation Packages for Spent Nuclear fuel*	NUREG-1617	N/A	March 2000	U.S. Nuclear Regulatory Commission	N/A		
NR	*Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel*	NUREG-1617, Supplemental 1	N/A	September 2005	U.S. Nuclear Regulatory Commission	R.S. Hafner, G.C. Mok, J. Hovingh, C. K. Syn, E.W. Russell, S.C. Keaton, J.L. Boles, D.K. Vogt, P. Prassinos		
NR	*Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)*	NUREG-1624	1	May 2000	U.S. Nuclear Regulatory Commission	N/A		NUREG-1624, Rev 1, is broken out into 2 PDF files: Cover – Section 11 and Appendices A – G. See NUREG-1880 and NUREG/CR-6350.
NR	*Radiological Assessments for Clearance of Materials from Nuclear Facilities (Main Report, Cover through Chapter 8, with Errata, Volume 1), (Appendices A–E, with Errata, Volume 2), (Appendices F–G, Volume 3), and (Appendices H–O, with Errata, Volume 4)*	NUREG-1640	N/A	June 2003(V1) October 2004 (V2) June 3003 (V3) May 2004 (V4)	U.S. Nuclear Regulatory Commission	R. Anigstein, H.J. Chmelynski, D.A. Loomis, J.J. Mauro, R.H. Olsher, and W.C. Thurber, SC&A, Inc., S.F. Marschke,Gemini Consulting Company, R.A. Meck, NRC		Volume 3 is broken out into 2 PDF files: Appendix F and Appendix G.

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NR	*U.S.-Specific Schedules of Requirements for Transport of Specified Types of Radioactive Material Consignments*	NUREG-1660	N/A	January 14, 1999	U.S. Nuclear Regulatory Commission, U.S. Department of Transportation	J. Cook, E. Easton, USNRC R. Boyle, DOT R. Pope, ORNL B. Dodd, D. Harlan, OSU		OSU is Oregen State University.
NR	*Final Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah (Cover to Appendix F, Volume 1), and (Appendix G, "Public Comments and Responses," and Appendix H, "Index of Commenters", Volume 2)*	NUREG-1714	N/A	December 2001	U.S. Nuclear Regulatory Commission	N/A		Private Fuel Storage, L.L.C., Docket 72.22
NR	*NMSS Decommissioning Standard Review Plan*	NUREG-1727	N/A	September 2000	U.S. Nuclear Regulatory Commission	N/A		
NR	*Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation*	NUREG-1736	N/A	October 2001	U.S. Nuclear Regulatory Commission	R.E. Zelac, J.L. Cameron, H. Karagiannis, J.R. McGrath, S.S. Sherbini, M.L. Thomas, and J.E. Wigginton		NUREG-1736 is broken out into 2 PDF files: Cover – Chapter 3 and Appendix A – End.
NR	*Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance*	NUREG-1745	N/A	June 2001	U.S. Nuclear Regulatory Commission	N/A		
NR	*Environmental Review Guidance for Licensing Actions Associated with NMSS Programs*	NUREG-1748	N/A	August 2003	U.S. Nuclear Regulatory Commission	N/A		
NR	*Consolidated Decommissioning Guidance (Decommissioning Process for Materials Licensees, Volume 1), (Characterization, Survey, and Determination of Radiological Criteria, Volume 2), (Financial Assurance, Recordkeeping, and Timeliness, Volume 3)*	NUREG-1757	2 (V1) 1 (V2) 1 (V3)	September 2006 (V1) September 2006 (V2) February 2012 (V3)	U.S. Nuclear Regulatory Commission	K.L. Banovac, J.T. Buckley, R.L. Johnson, G.M. McCann, J.D. Parrott, D.W. Schmidt, J.C. Shepherd, T.B. Smith, P.A. Sobel, B.A. Watson, D.A. Widmayer, T.H. Youngblood for V1; D.W. Schmidt, K.L. Banovac, J.T. Buckley, D.W. Esh, R.L. Johnson, J.J. Kottan, C.A. McKenney, T.G. McLaughlin, S. Schneider for V2; K.M. Kline, C.M. Dean (ICF International), T.L. Fredrichs, M.C. Maier, E.R. Pogue, and R.N. Young (TN Department of Environment and Conservation) for V3.		

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NR	*Radiological Surveys for Controlling Release of Solid Materials(Draft Report for Comment)*	NUREG-1761	N/A	July 2002	U.S. Nuclear Regulatory Commission	E.W. Abelquist and T.J. Bower, Oak Ridge Institute for Science and Education C.V. Gogolak and P. Shebell, U.S. Department of Energy R. Coleman, ORNL G.E. Powers,NRC		Draft Report for Comment
NR	*United States Nuclear Regulatory Commission Package Performance Study Test Protocols (Draft Report for Comment)*	NUREG-1768	N/A	February 2003	U.S. Nuclear Regulatory Commission	N/A		Draft Report for Comment
NR	*Environmental Impact Statement for the Proposed Idaho Spent Fuel Facility at the Idaho National Engineering and Environmental Laboratory in Butte County, Idaho*	NUREG-1773	N/A	January 2004	U.S. Nuclear Regulatory Commission	N/A		
NR	*Good Practices for Implementing Human Reliability Analysis (HRA)*	NUREG-1792	N/A	Arpil 2005	U.S. Nuclear Regulatory Commission	A. Kolaczkowski, Science Applications International Corporation, J. Forester, SNL E. Lois and S. Cooper, NRC		
NR	*Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, pluse ERRATA and Supplement 1, Volume 1 and Volume 2 (Appendices)*	NUREG-1805	N/A	December 2004 (Main Report) July 2013 (Supplement1, V1 & V2)		Naeem Iqbal, Mark Henry Salley, and Sunil Weerakkody, NRC for Main Report. D. Stroup, G. Taylor, G. Hausman, and M. H. Salley, NRC for Supplement 1, Volume 1 and Volume 2		Supplement 1, Volume 2 contains appendices.
NR	*Evaluation of Human Reliability Analysis Methods Against Good Practices*	NUREG-1842	N/A	September 2006	U.S. Nuclear Regulatory Commission	J. Forester, SNL A. Kolaczkowski and D. Kelly, Science Applications International Corporation E. Lois, NRC		
NR	*ATHEANA User's Guide*	NUREG-1880	N/A	June 2007	U.S. Nuclear Regulatory Commission	John Forester, SNL Alan Kolaczkowski, Science Applications International Corporation, Dennis Bley, Buttonwood Consulting, Inc., Susan Cooper and Erasmia Lois, NRC		User's guide for the human reliability analysis (HRA) method known as "A Technique for Human Event Analysis" (ATHEANA) See NUREG-1624 and NUREG/CR-6350.
NR	*Joint Canada - United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages*	NUREG-1886	N/A	March 2009	U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards	M. Rahimi, N.L. Osgood, and M. Sampson, USNRC M. Conroy, DOT S. Faille and K. Glenn, CNSC		This guide is published in Canada as RD-364. Canadian Nuclear Safety Commission (CNSC), U.S. DOT-Pipeline and Hazardous Materials Safety Administration Packaging and Transport Licensing Division

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NR	*RASCAL 3.0.5: Description of Models and Methods*	NUREG-1887	N/A	August 2007	U.S. Nuclear Regulatory Commission	S.A. McGuirea, NRC J.V. Ramsdell, Jr., PNNL G.F. Athey, Athey Consulting		Code used by NRC's emergency operations center for making dose projections for atmospheric releases during radiological emergencies.
NR	*RASCAL 3.0.5 Workbook*	NUREG-1889	N/A	September 2007	U.S. Nuclear Regulatory Commission	G.F. Athey, Athey Consulting S.A. McGuire, NRC J.V. Ramsdell, Jr., PNNL		See NUREG-1887. Workbook contains problems designed to familiarize the user with the RASCAL software through hands-on problem solving.
NR	*Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance*	NUREG-1927	N/A	March 2011	U.S. Nuclear Regulatory Commission	N/A		
NR	*Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,* Draft Report for Comment	NUREG-1927	1 (Draft)	Month? 2015	U.S. Nuclear Regulatory Commission	N/A		Draft Revison 1 became available March 9, 2015 for "Internal Review and Coordination." Purpose is to provide the Advisory Committee on Reactor Safeguards (ACRS) a draft copy for ACRS's Metallurgy & Reacor Fuels subcommittee meeting scheduled, April 8, 2015.
NR	*RASCAL 4.3: Description of Models and Methods (Initial Report and Supplement 1)*	NUREG-1940	N/A	December 2012 (Initial Report) Supplement 1 (May 2015)	U.S. Nuclear Regulatory Commission	J.V. Ramsdell, Jr., PNNL G.F. Athey, Athey Consulting, S.A. McGuire, NRC (Retired) L.K. Brandon, NRC for Initiial Report. J. V. Ramsdell, Jr.Ramsdell Environmental Consulting G. F. Athey, Athey Consulting J. P. Rishel, PNNL for Supplement 1.		Updated Version of RASCAL., See NUREG 1887 and NUREG 1889 for RASCAL 3.0.5 information.
NR	*Intrusion Detection Systems and Subsystems: Technical Information for NRC Licensees*	NUREG-1959	N/A	March 2011	U.S. Nuclear Regulatory Commission	N/A		
NR	*Access Control Systems: Technical Information for NRC Licensees*	NUREG-1964	N/A	Arpil 2011	U.S. Nuclear Regulatory Commission	N/A		
NR	*Central and Eastern United States Seismic Source Characterization for Nuclear Facilities (Chapters 1 to 4, Volume 1) (Chapters 5 to 7, Volume 2), (Chapters 8 to 11, Volume 3), (Appendices A and B, Volume 4), (Appendices C to F, Volume 5), (Appendices G to L, Volume 6)*	NUREG-2115	N/A	January 2012	U.S. Nuclear Regulatory Commission	N/A		NUREG-2115 was co-sponsored by U.S. Department of Energy, Electric Power Research Institute, and U.S. Nuclear Regulatory Commission

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NR	*Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies*	NUREG-2117	1	April 2012	U.S. Nuclear Regulatory Commission	Annie M. Kammerer and Jon P. Ake, NRC		
NR	*Spent Fuel Transportation Risk Assessment*	NUREG-2125	N/A	January 2014	U.S. Nuclear Regulatory Commission	N/A		
NR	*A Proposed Risk Management Regulatory Framework*	NUREG-2150	N/A	April 2012	U.S. Nuclear Regulatory Commission	Commissioner George Apostolakis, M. Cunningham, C.		
NR	*Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications*	NUREG-2152	N/A	March 2013	U.S. Nuclear Regulatory Commission	Ghani Zigh and Jorge Solis, NRC		
NR	*Acceptability of Corrective Action Programs for Fuel Cycle Facilities: Draft Report for Comment*	NUREG-2154	N/A	January 2013	U.S. Nuclear Regulatory Commission	Sabrina Atack, NRC		Draft Report for Comment
NR	*Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel:Final Report, Volume 1 and Volume 2*	NUREG-2157	N/A	September 2014	U.S. Nuclear Regulatory Commission	N/A		Volume 2 includes public comments.
NR	*Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Casks*	NUREG-2174	N/A	February 2015	U.S. Nuclear Regulatory Commission	N/A		
RD	*EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project*	3002000717	N/A	June 2013	Electric Power Research Institute	N/A		Supports NUREG-2115.
RD	*Yankee Atomic Electric Co. v. United States*	*536 F.3d 1268*	N/A	August 7, 2008	United States Court of Federal Claims	James F. Merow	R 0.1.4	
RD	*Civilian Radioactive Waste Management System Requirements Document*	A00000000 – 00811-1708 – 00003	3	November 1996	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		
RD	*Projection of Future Spent Nuclear Fuel Discharges*	CAL-WAT-SE-000007	0	2009	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		Not able to locate copy of document.
RD	*Commission Order: Memorandum and Order on Continued Storage*	CLI-14-08	N/A	August 26, 2014	U.S. Nuclear Regulatory Commission	N/A		
RD	Safety Evaluation Report of the Site-Related Aspects of the Private Fuel Storage Facility, Independent Spent Fuel Storage Installation	Docket No. 72-0022	N/A	Unknown	U.S. Nuclear Regulatory Commission	N/A		Proposed Private Fuel Storage Facility that was to be located in Tooele County, Utah.
RD	*Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (FEIS)*	DOE/EIS-0250	N/A	February 2002	DOE - Office of Civilian Radioactive Waste Management (OCRWM)	N/A	R 1.0.3	
RD	*Civilian Radioactive Waste Management System Requirements Document*	DOE/RW-0406	8	September 12, 2007	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RD	*Acceptance Priority Ranking and Annual Capacity Report*	DOE/RW-0567	N/A	July 2004	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		HQO.20041004.0041
RD	*Used Nuclear Fuel Storage and Transportation Research, Development , and Demonstration Plan*	FCRD-FCT-2012 000053	0	April 2012	U.S. Department of Energy, Fuel Cycle Reseach & Development	N/A	R 2.0.5	
RD	*Status Report on Design Concept Trade Studies: Fleet Maintenance Functional Requirements and Technical Trade Studies*	FCRD-NFST 2014-000542	N/A	?	U.S. Department of Energy, Fuel Cycle Reseach & Development	N/A		
RD	*Used Fuel Management System Architecture Evaluation, Fiscal Year 2012*	FCRD-NFST-2013-000020	0	October 31, 2012	U.S. Department of Energy	M. Nutt, E. Moris, F. Puig, J. Carter, P. Rodwell, A. Delley, R. Howard, D. Giuliano		
RD	*A Project Concept for Nuclear Fuels Storage and Transportation*	FCRD-NFST-2013-000132	1	June 15, 2013	U.S. Department of Energy	J. T. Carter (SRNL), S. Dam & et al (Tech Source, Inc.)		
RD	*Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*	FCRD-NFST-2013-000263	0	August 30, 2013	U.S. Department of Energy	J. T. Carter, D. R. Leduc	R 0.1, R 0.1.4, R 0.2, R 0.11, R 0.12, R 2.0.3	
RD	*Preliminary Site Factors and Considerations for Interim Used Fuel Storage Facilities*	FCRD-NFST-2013-000370	0	September 23, 2013	U.S. Department of Energy, Fuel Cycle Reseach & Development	N/A	R 2.0.4	
RD	*Used Fuel Management System Architecture Evaluation, Fiscal Year 2013*	FCRD-NFST-2013-000377	1	October 31, 2013	U.S. Department of Energy	M. Nutt, F. Puig, E. Morris, Y. Park, R. Joseph III, G. Peterson, D. Giuliano, R. Howard		
RD	*AAR S-2043 Cask Railcar System Requirements Document*	FCRD-NFST-2014-000093	N/A	March 31, 2014	U.S. Department of Energy, Fuel Cycle Research & Development	Feldman, M., Maheras, S. J., Best, R.E.		
RD	*Dry Storage of Used Fuel Transition to Transport*	FCRD-UFD-2012-000253	0	August 2012	Savannah River National Laboratory	D. R. Leduc		
RD	*Used Fuel Research and Development Extended Used Fuel Storage R&D Functions and Requirements*	FCRD-USED-2011-000030	0c	December 2010	U.S. Department of Energy	N/A		
RD	*The Gap Analysis to Support Extended Storage of Used Nuclear Fuel*	FCRD-USED-2011-000136	0	January 31, 2013	U.S. Department of Energy, Used Fuel Division	Brady Hanson & et. al.	R 2.0.5	
RD	*Consolidated Storage Lessons Learned and Background Information*	FCRD-USED-2011-000345	0	September 13, 2011	U.S. Department of Energy	J. Carter, A. Delley, T. Cotton		
RD	*Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States*	ISBN: 0-309-65316-9	N/A	2006	National Research Council, Committee on Transportation of Radioactive Waste	N/A	R 1.0.3	

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RD	*Task Order 11 – Development of Consolidated Fuel Storage Facility Concepts*	N/A	N/A	February 1, 2013	Energy Solutions submitted to DOE under Contract No. DE-NE0000293	N/A		
RD	*Final Report, Task Order 11 – Development of Consolidated Fuel Storage Facility Concepts*	N/A	N/A	January 31, 2013	Shaw Environmental & Infrastructure, Inc. submitted to DOE under Contract No. DE-NE0000292	N/A		
RD	*Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy*	N/A	N/A	January 2012	Blue Ribbon Commission on America's Nuclear Future	N/A	R 0.6, R 2.0.3, R 2.0.5	
RD	*Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel NWTRB report*	N/A	N/A	December 2010	U.S. Department of Energy	N/A		
RD	*Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel*	N/A	N/A	December 2010	United States Nuclear Waste Technical Review Board	Douglas B. Rigby		
RD	*Task Order 11 – Development of Consolidated Fuel Storage Facility Concepts Report*	RPT-3008097-000	N/A	February 12, 2013	Areva Federal Services LLC submitted to DOE under Contract No. DE-NE0000291	N/A		Good Summary of requirement of 10CFR72 on P 7-9.
RD	*Nuclear Waste Administration Act of 2013*	S. 1240 (113th Congress)	N/A	Not Enacted	N/A	N/A	R 2.0.3	Nuclear Waste Administration Act of 2015 introduced to replace S1240.
RD	*Calculation Method for the Projection of Future Spent Nuclear Fuel Discharges*	TDR-WAT-NU-000002	2	August 2005	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	B. McLeod		DOC.20051024.0002 (WBS: 3.2.3) was prepared by Bechtel SAIC Company, LLC under contact DE-AC08-01NV12101
RD, PG	*Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*	N/A	N/A	January 2013	U.S. Department of Energy	N/A	R 0.1, R 0.1.1, R 0.1.2, R 0.1.3, R 0.7, R 0.8, R 2.0.1, R 2.0.2, R 2.0.3	
RD, RR	*Federal Water Pollution Control Amendments of 1972 (Clean Water Act)*	33 U.S.C. § 1251 et seq.	N/A	October 18, 1972	92nd United States Congress	N/A		Public Law 92-500
RD, RR	*Resource Conservation and Recovery Act (Solid Waste Disposal Act)*	42 U.S.C. § 6901 et seq.	N/A	October 21, 1976	94th United States Congress	N/A		Public Law 94-580
RD, RR	*Nuclear Waste Policy Act as Amended*	42 U.S.C. 10101	**N/A**	March 2004	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		
RD, RR	*Comprehensive Environmental Response, Compensation, and Liability Act*	42 U.S.C. § 9601 et seq.	N/A	December 11, 1980	96th United States Congress	N/A		Public Law 96-510

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RD, RR	*Water Quality Act of 1987*	N/A	N/A	February 4, 1987	100th United States Congress	N/A		Public Law 100-4
RG	*Flood Protection for Nuclear Power Plants*	RG 1.102	1	September 1976	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2014
RG	*Tornado Design Classification*	RG 1.117	1	April 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2012
RG	*Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components*	RG 1.122	1	February 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2015
RG	*Spent Fuel Storage Facility Design Basis*	RG 1.13	2	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2010
RG	*Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23)*	RG 1.23	1	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)*	RG 1.25	0	March 1972	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2010
RG	*Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*	RG 1.26	4	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2013
RG	*Quality Assurance Program Criteria (Design and Construction)*	RG 1.28	4	June 2010	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Seismic Design Classification*	RG 1.29	4	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2013
RG	*Quality Assurance Program Requirements (Operation)*	RG 1.33	3	June 2013	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Design Basis Floods for Nuclear Power Plants*	RG 1.59	2	August 1977	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2014
RG	*Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants*	RG 1.76	1	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: August 2011
RG	*Combining Modal Responses and Spatial Components in Seismic Response Analysis*	RG 1.92	3	October 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*	RG 3.48	1	August 1989	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: June 2014
RG	*Standard Format and Content for a Specific License Application for An Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility*	RG 3.50	2	September 2014	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation*	RG 3.53	N/A	July 1982	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: April 2014

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RG	*Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation*	RG 3.54	1	January 1999	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: March 2011
RG	*Design of an Independent Spent Fuel Storage Installation (Dry Storage)*	RG 3.60	N/A	March 1987	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: May 2014
RG	*Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask*	RG 3.61	N/A	February 1989	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: June 2014
RG	*Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks*	RG 3.62	N/A	February 1989	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: June 2014
RG	*Standard Format and Content of Decommissioning Plans for Materials Licensees*	RG 3.65	1	May 2008	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: November 2013
RG	*Standard Format and Content of Financial Assurance Mechanisms*	RG 3.66	1	May 2008	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: November 2013
RG	*Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities*	RG 3.67	1	April 2011	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Nuclear Criticality Safety Standards for Fuels and Material Facilities*	RG 3.71	2	December 2010	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations*	RG 3.73	N/A	October 2003	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: May 2011
RG	*Corrective Action Programs for Fuel Cycle Facilities*	RG 3.75	N/A	July 2014	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment*	RG 4.15	2	July 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: April 2012
RG	*Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities*	RG 4.16	2	December 2010	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors*	RG 4.20	1	April 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RG	*Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning*	RG 4.21	N/A	June 2008	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: July 2013
RG	*Decommissioning Planning During Operations*	RG 4.22	N/A	December 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Training, Equipping, and Qualifying of Guards and Watchmen*	RG 5.20	N/A	January 1974	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2011
RG	*Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities (for Comment)*	RG 5.55	N/A	March 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2011
RG	*Standard Format and Content of Safeguards Contingency Plans for Transportation (for Comment)*	RG 5.56	N/A	March 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2011
RG	*Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas*	RG 5.7	1	May 1980	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: April 2015
RG	*Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material*	RG 7.10	2	March 2005	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011
RG	*Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)*	RG 7.11	N/A	June 1991	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011. Basis for NRC RG 7.11 is provided by NUREG-1815.
RG	*Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)*	RG 7.12	N/A	June 1991	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011
RG	*Transportation Security Plans for Classified Matter Shipments*	RG 7.13	?	Unknown	U.S. Nuclear Regulatory Commission	N/A		Document marked "Official Use Only – Security-Related Information."
RG	*Leakage Tests on Packages for Shipment of Radioactive Materials*	RG 7.4	1	March 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*	RG 7.6	1	March 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: June 2014
RG	*Administrative Guide for Verifying Compliance with Packaging Requirements for Shipments of Radioactive Materials (for Comment)*	RG 7.7	1	March 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*	RG 7.8	1	March 1989	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: June 2011
RG	*Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*	RG 7.9	2	March 2005	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011
RG	*Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable*	RG 8.10	1-R	May 1997	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2014
RG	*Air Sampling in the Workplace*	RG 8.25	1	June 1992	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2011

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RG	*Applications of Bioassay for Fission and Activation Products*	RG 8.26	N/A	September 1980	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: December 2011
RG	*Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants*	RG 8.27	N/A	March 1981	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2011
RG	*Instruction Concerning Risks from Occupational Radiation Exposure*	RG 8.29	1	February 1996	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: December 2011
RG	*Monitoring Criteria and Methods To Calculate Occupational Radiation Doses*	RG 8.34	N/A	July 1992	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*ALARA Levels for Effluents from Materials Facilities*	RG 8.37	N/A	July 1993	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: February 2012
RG	*Control of Access to High and Very High Radiation Areas of Nuclear Plants*	RG 8.38	1	May 2006	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Methods for Measuring Effective Dose Equivalent from External Exposure*	RG 8.40	N/A	July 2010	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Fuel Retrievability*	SFST-ISG-2	1	February 22, 2010	U.S. Nuclear Regulatory Commission, Division of Spent Fuel Storage and Transportation (SFST)	N/A		Interim Staff Guidance
RR	*Standards for Protection against Radiation*	10 CFR 20	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		
RR	*Domestic Licensing of Production and Utilization Facilities*	10 CFR 50	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		Potentially applicable, e.g., in any limited cases where 10 CFR 72 does not govern.
RR	*Packaging and Transportation of Radioactive Material*	10 CFR 71	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A	R 2.1.5	
RR	*Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*	10 CFR 72	N/A	January 1, 2006	U.S. Nuclear Regulatory Commission	N/A	R 2.0.4	
RR	*Physical Protection of Plants and Materials*	10 CFR 73	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		
RR	*Material Control and Accounting of Special Nuclear Material*	10 CFR 74	N/A	January 1, 2006	U.S. Nuclear Regulatory Commission	N/A		
RR	*Safeguards on Nuclear Materials-Implementation of U.S./IAEA Agreement*	10 CFR 75	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		
RR	*Worker Safety and Health Program*	10 CFR 851	N/A	N/A	U.S. Department of Energy	N/A		
RR	*Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste*	10 CFR 961	N/A	N/A	U.S. Department of Energy	N/A	R 0.3	
RR	*Occupational Safety and Health Standards*	29 CFR 1910	N/A	N/A	U.S. Department of Labor, Occupational Safety and Health Administration	N/A		
RR	*Identification and Listing of Hazardous Waste*	40 CFR 261	N/A	July 1, 2012	U.S. Environmental Protection Agency	N/A		

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RR	*National Primary and Secondary Ambient Air Quality Standards*	40 CFR 50	N/A	N/A	U.S. Environmental Protection Agency	N/A		
RR	*National Emission Standards for Hazardous Air Pollutants*	40 CFR 61	N/A	July 1, 2011	U.S. Environmental Protection Agency	N/A		
RR	*Designation of Areas for Air Quality Planning Purposes*	40 CFR 81	N/A		U.S. Environmental Protection Agency	N/A		
RR	*Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, Training Requirements, and Security Plans* , Subpart I, *Safety and Security Plans*	49 CFR 172, Subpart I	N/A	N/A	U.S. Department of Transportation, Pipeline and Hazardous Materials Safety Administration	N/A		
RR	*Transportation: Shippers—General Requirments for Shipments and Packings*	49 CFR 173	N/A	N/A	U.S. Department of Transportation, U.S. Department of Homeland Security	N/A	R 2.1.5	
RR, RD	*Regulations for the Safe Transport of Radioactive Material, 2012 Edition (Specific Safety Requirements)*	No. SSR-6	N/A	October 2012	International Atomic Energy Agency	N/A		IAEA Safety Standard, STI/PUB/1570, ISBN:978-92-0-133310-0

DR = Design Requirement (Design Standard?)
NR = NRC NUREG
PG = Program Guidance
RD = Resource Document
RG = Regulatory Guidance
RR = Regulatory Requirements

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DR	*Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*	ANSI/ANS 8.1	N/A	April 15, 2014	American National Standards Institute, American Nuclear Society	N/A		ANSI Standard
DR	*Design Fabrication, and Construction of Freight Cars*	M-1001 (Manual of Standards and Recommended Practices Section C, Part II)	N/A	April 13, 2011	Association of American Railroads (AAR)	N/A	R 1.0.3	
DR	*Radioactive Materials - Leakage Tests on Packages for Shipment*	N14.5	N/A	January 1, 1997	American National Standards Institute	N/A		
DR	*Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more*	N14.6	N/A	201X	American National Standards Institute	N/A		ANSI Standard is being developed by Institute of Nuclears Materials Management
DR	*Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)*	NOG-1	N/A	2015	American Society of Mechanical Engineers	N/A		ASME Standard
DR	*Quality Assurance Requirements for Nuclear Facility Applications*	NQA-1	N/A	2015	American Society of Mechanical Engineers	N/A		ASME Standard
NR	*Transporting Spent Fuel: Protection Provided Against Severe Highway and Railroad Accidents*	NUREG/BR-0111	N/A	March 1987	U.S. Nuclear Regulatory Commission	William R. Lahs		
NR	*Safety of Spent Fuel Transportation*	NUREG/BR-0292	N/A	March 2003	U.S. Nuclear Regulatory Commission	N/A		NRC brochure.
NR	*Shock and Vibration Environments for a Large Shipping Container during Truck Transport Part II)*	NUREG/CR-0128	N/A	May 1978	U.S. Nuclear Regulatory Commission	Clifford F. Magnuson		Prepared by Sandia Laboratories as document SAND78-0337, RT.
NR	*SCALE Ver 4.4: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation (Control Modules C4, C6;Volume 1, Part 1), (Control Modules D1, S1-S5, H1; Volume 1, Part 2), (Functional Models F1 - F8; Volume 2, Part 1), (Functional Models F9 - F11; Volume 2, Part 2), (Functional Models F16 - F17; Volume 2, Part 3), (Miscellaneous M1 - M17; Volume 3)*	NUREG/CR-0200	6	May 2000	U.S. Nuclear Regulatory Commission	N/A		Prepared by Oak Ridge National Laboratory as document number ORNLINUREG/CSD-2/R6.
NR	*An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers*	NUREG/CR-0481	N/A	September 1978	U.S. Nuclear Regulatory Commission	Henry J. Rack, Gerald A. Knorovsky		Prepared by Sandia Laboratories as document SAND77-1872, R-7.
NR	*Fission Product Release from Highly Irradiated LWR Fuel*	NUREG/CR-0722	N/A	February 1980	U.S. Nuclear Regulatory Commission	R. A. Lorenz J. L. Collins A. P. Malinauskas O. L. Kirkland R. L. Towns		Prepared by Oak Ridge National Laboratory as document number ORNL/NUREG/TM-287/R1.
NR	*SKYSHINE II Procedure: Calculation of the Effects of Structure Design Assessment on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air (Main Report and Supplement 1)*	NUREG/CR-0781	N/A	May 1979 (Main Report) January 1981 (Supplement 1)	U.S. Nuclear Regulatory Commission	C. M. Lampley		Prepared by Radiation Research Associates, Fort Worth, TX as document number RRA-T7901 (Main Report) and RRA-T8008 (Supplement 1).

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*Approaches to Acceptable Risk: A Critical Guide*	NUREG/CR-1614	N/A	December 1980	U.S. Nuclear Regulatory Commission	Baruch Fischhoff, Sarah Lichtenstein, and Paul Slovic, ORNL Ralph Keeney, Woodward-Clyde Consultants Stephen Derby, Stanford University		Prepared by Oak Ridge National Laboratory as document number ORNL/Sub-7656/1.
NR	*Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick*	NUREG/CR-1815	N/A	June 15, 1981	U.S. Nuclear Regulatory Commission	W. R. Holman, R. T. Langland		Prepared by Lawrence Livermore National Laboratory as document number UCRL-53013. NUREG-1815 provides basis for NRC Reg. Guide 7.11.
NR	*Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages, Volume 1, Volume 2, and Volume 3*	NUREG/CR-2146	N/A	November 1981 (V1) July 1983 (V2) October 1983 (V3)	U.S. Nuclear Regulatory Commission	S. R. Fields		Prepared by Hanford Engineering Development Laboratory as documents: HEDL-TME 81-15, RT (Vol 1, Quarterly Progress Report January 1, 1981 - March 31, 1981); HELD-TME 83-8, RT (Vol 2, Quartly Progress Report April 1, 1981 - June 30, 1981); and HELD 83-18, RT, (Vol 3, Final Summary Report)
NR	*Probabilistic Safety Analysis Procedures Guide*	NUREG/CR-2815	N/A	January 1984	U.S. Nuclear Regulatory Commission	I.A. Papazoglou, R.A. Bari, A.J. Buslik, R.E. Hall, D. Ilberg, P.K. Samanta, T. Teichmann, and R.W. Youngblood, BNL A. El-Bassioni, NRC, J. Fragola and E. Lofgren, Science Applications, Inc. W. Vesely, Battelle Columbus Laboratories		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-51559.
NR	*Recommended Welding Criteria For Use in the Fabrication of Shipping Containers for Radioactive Materials*	NUREG/CR-3019	N/A	March 1984	U.S. Nuclear Regulatory Commission	R. E. Monroe, H. H. Woo, R. G. Sears		Prepared by Lawrence Livermore National Laboratory as document number UCRL-53044, RM.
NR	*User's Manual for FIRIN: A Computer Code to Estimate Accidental Fire and Airborne Releases in Nuclear Fuel Cycle Facilities*	NUREG/CR-3037	N/A	February 1989	U.S. Nuclear Regulatory Commission	M.K. Chan, M.Y. Ballinger, and P.C. Owczarski		Prepared by Pacific Northwest National Laboratory as document number PNL-4532.
NR	*Measures of Risk Importance and Their Applications*	NUREG/CR-3385	N/A	July 1983	U.S. Nuclear Regulatory Commission	W.E. Vesely, T.C. Davis, R.S. Denning, and N. Saltos		Prepared by Battelle Columbus Laboratories as document number BMI-2103.
NR	*Material Transport Analysis for Accident-Induced Flow in Nuclear Facilities*	NUREG/CR-3527	N/A	October 1983	U.S. Nuclear Regulatory Commission	R.A. Martin, P.K. Tang, A.P. Harper, J.D. Novat, and W.S. Gregory		Prepared by Los Alamos National Laboratory as document number LA-9913-MS.

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*The Effect of Thermal and Irradiation Aging Simulation Procedures on Polymer Properties*	NUREG/CR-3629	N/A	April 1984	U.S. Nuclear Regulatory Commission	L.D. Bustard and E. Minor, SNL J. Chenion, F. Carlin, C. Alba, and G. Gaussens, CEA-ORIS LABRA at Saclay, France, M. LeMeur, CEA-DAS-SAF, France		Prepared by Sandia National Laboratory as document number SAND83-2651.
NR	*Accident-Induced Flow and Material Transport in Nuclear Facilities: A Literature Review*	NUREG/CR-3735	N/A	April 1984	U.S. Nuclear Regulatory Commission	J.W. Bolstad, W.S. Gregory, R.A. Martin, P.K. Tang, R.G. Merryman, J.D. Novat, and H.L. Whitmore		Prepared by Los Alamos National Laboratory as document number LA-10079-MS.
NR	*A Study on Ductile and Brittle Failure Design Criteria for Ductile Cast Iron Spent-Fuel Shipping Containers*	NUREG/CR-3760			U.S. Nuclear Regulatory Commission	M.W. Schwartz		Prepared by Lawrence Livermore National Laboratory as document number TI84-014381.
NR	*Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than Four Inches Thick*	NUREG/CR-3826	N/A	April 1984	U.S. Nuclear Regulatory Commission	M.W. Schwartz		Prepared by Lawrence Livermore National Laboratory as document number UCRL 53538.
NR	*Fabrication Criteria for Shipping Containers*	NUREG/CR-3854	N/A	March 1985	U.S. Nuclear Regulatory Commission	L.E. Fischer, W. Lai		Prepared by Lawrence Livermore National Laboratory as document number UCRL-53544.
NR	*Methods for Impact Analysis of Shipping Containers*	NUREG/CR-3966	N/A	November 1987	U.S. Nuclear Regulatory Commission	T.A. Nelson, R. C. Chun		Prepared by Lawrence Livermore National Laboratory as document number UCID-20639.
NR	*A Study on Fabrication Criteria for Ductile Cast Iron Spent-Fuel Shipping Containers*	NUREG/CR-4363	N/A	February 1986	U.S. Nuclear Regulatory Commission	M.W. Schwartz		Prepared by Lawrence Livermore National Laboratory as document number UCRL 53662.
NR	*Tornado Climatology of the Contiguous United States*	NUREG/CR-4461	2	February 2007	U.S. Nuclear Regulatory Commission	J.V. Ramsdell, Jr. and J.P. Rishel		Prepared by Pacific Northwest National Laboratory as document number PNNL-15112.
NR	*Design Features for Enhancing International Safeguards of Away-from- Reactor Dry Storage for Spent LWR Fuel*	NUREG/CR-4517	N/A	October 1986	U.S. Nuclear Regulatory Commission	F.P. Roberts and N.L. Harms		Prepared by Pacific Northwest National Laboratory as document number PNL-5321.
NR	*SCANS, A Microcomputer Based Analysis System for Shipping Cask Design Review*	NUREG/CR-4554	2	March 1998	U.S. Nuclear Regulatory Commission	G. C. Mok, G. R. Thomas, M. A. Gerhard, D. J. Trummer, G. L. Johnson		Vol. 1 of User's Manual to Shipping Cask Analysis System (SCANS) Version 3a. Prepared by Lawrence Livermore National Laboratory as document number UCID-20674.
NR	*FIRAC User's Manual: A Computer Code to Simulate Fire Accidents in Nuclear Facilities*	NUREG/CR-4561	N/A	April 1986	U.S. Nuclear Regulatory Commission	B.D. Nichols and W.S. Gregory		Prepared by Los Alamos National Laboratory as document number LA-10678-M.

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*Guide for Preparing Operating Procedures for Shipping Packages*	NUREG/CR-4775	N/A	December 1998	U.S. Nuclear Regulatory Commission	M. C. Witte		Prepared by Lawrence Livermore National Laboratory as document number UCID-20820.
NR	*Shipping Container Response to Severe Highway and Railway Accident Conditions*	NUREG/CR-4829	N/A	February 1987	U.S. Nuclear Regulatory Commission	L. E. Fischer, C. K. Chou, M. A. Gerhard, C. Y. Kimura, R. W. Martin, R. W. Mensing, M. E.		Vol. 2. Prepared by Lawrence Livermore National Laboratory as document number UCID-20733.
NR	*Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses Within 10 CFR Part 71*	NUREG/CR-5342	N/A	July 1998	U.S. Nuclear Regulatory Commission	C. V. Parks, C. M. Hopper, and J. J. Lichtenwalter		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-13607.
NR	*Recommendations for Resolution of Public Comments on USI A-40, "Seismic Design Criteria"*	NUREG/CR-5347	N/A	May 1989	U.S. Nuclear Regulatory Commission	A.J. Philippacopoulos		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52191.
NR	*Elements of an Approach to Performance-Based Regulatory Oversight*	NUREG/CR-5392	N/A	January 1999	U.S. Nuclear Regulatory Commission	R.W. Youngblood, R.N.M. Hunt, E.R. Schmidt J. Bolin, F. Dombek, and D. Prochnow		Prepared by SCIENTECH, Inc. as document number SCIE-NRC-373-98.
NR	*Predicting the Pressure Dimension Flow of Gasses Through Micro-Capillaries and Micro-Orifices*	NUREG/CR-5403	N/A	November 18, 1994	U.S. Nuclear Regulatory Commission	B. L. Anderson, R. W. Carlson, L. E. Fischer		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-118245.
NR	*Engineering Drawings for 10 CFR Part 71 Package Approvals*	NUREG/CR-5502	N/A	May 1998	U.S. Nuclear Regulatory Commission	M. K. Sheaffer, G. R. Thomas, R. K. Dann, E. W. Russell		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-130438.
NR	*Residual Radioactive Contamination From Decommissioning: User's Manual DandD Version 2.1 (Volume 2)*	NUREG/CR-5512	N/A	April 2001	U.S. Nuclear Regulatory Commission	K. McFadden, Sigma Software LLC D.A. Brosseau, and W.E.		Sigma Software, LLC and Gram, Inc worked under contract to Sandia National Laboratories on
NR	*AUTOCASK: A Microcomputer Based System for Shipping Cask Design Review Analysis*	NUREG/CR-5657	N/A	April 1995	U.S. Nuclear Regulatory Commission	M. A. Gerhard, S. C. Sommer		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-118766.
NR	*Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages*	NUREG/CR-5661	N/A	April 1997	U.S. Nuclear Regulatory Commission	H. R. Dyer, C. V. Parks		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-11936.
NR	*Technical Bases for Regulatory Guide for Soil Liquefaction*	NUREG/CR-5741	N/A	March 2000	U.S. Nuclear Regulatory Commission	J. P. Koester, M. K. Sharp, and M. E. Hynes		Prepared by U.S. Army Corps of Engineers, 3909 Halls Ferry Road Vicksburg, MS 39180-6199
NR	*Stress Analysis of Closure Bolts for Shipping Casks*	NUREG/CR-6007	N/A	April 1992	U.S. Nuclear Regulatory Commission	G. C. Mok and L. E. Fischer, LLNL S. T. Hsu, Kaiser Engineering		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-110637.
NR	*CASKS (Computer Analysis of Storage casKS): A microcomputer based analysis system for storage cask design review. User`s manual to Version 1b (including program reference)*	NUREG/CR-6242	N/A	February 1995	U.S. Nuclear Regulatory Commission	T.F. Chen, M.A.; Gerhard, D.J.Trummer, G.LJohnson, and G.C. Mok		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID--117418.

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NR	*Multidisciplinary Framework for Human Reliability Analysis with an Application to Errors of Commission and Dependency*	NUREG/CR-6265	N/A	August 1995	U.S. Nuclear Regulatory Commission	M.T. Barriere and W.J. Luckas, Jr., BNL, J. Wreathall, JW&Co., S.E. Cooper, SAIC, D.C. Bley, PLG, A. Ramey-Smith, NRC		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52431.
NR	*Quality Assurance Inspections for Shipping and Storage Containers*	NUREG/CR-6314	N/A	April 1996	U.S. Nuclear Regulatory Commission	H.M. Stromberg, G.D. Roberts, and J.H. Bryce		Prepared by Idaho National (Engineering) Laboratory as document number INEL-95/0061.
NR	*Buckling Analysis of Spent Fuel Basket*	NUREG/CR-6322	N/A	May 1995	U.S. Nuclear Regulatory Commission	A. S. Lee, S. E. Bumpas		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-119697.
NR	*A Technique for Human Error Analysis (ATHEANA)*	NUREG/CR-6350	N/A	May 1996	U.S. Nuclear Regulatory Commission	S.E. Cooper, SAIC A.M. Ramey-Smith, NRC J. Wreathall, and D.C. Bley, WWG		Prepared by Brookhaven National Laboratory as document BNL-NUREG-52467 with support from subcontractors:
NR	*Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*	NUREG/CR-6361	N/A	March 1997	U.S. Nuclear Regulatory Commission	J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, D. M. Hopper		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-13211.
NR	*Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts (Main Report, Volume 1) (Appendices, Volume 2)*	NUREG/CR-6372	N/A	April 1997	U.S. Nuclear Regulatory Commission	R.J. Budnitz (Chairman), G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris		Senior Seismic Hazard Analysis Committee (SSHAC) under contract to LLNL to create document UCRL-ID- 122160.
NR	*Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*	NUREG/CR-6407	N/A	February 1996	U.S. Nuclear Regulatory Commission	J. W. McConnell, Jr., A. L. Ayers, M. J. Tyacke		Prepared by Idaho National (Engineering) Laboratory as document number INEL-95/0551.
NR	*Nuclear Fuel Cycle Facility Accident Analysis Handbook*	NUREG/CR-6410	N/A	March 1998	U.S. Nuclear Regulatory Commission	N/A		Prepared by Science Applications International Corporation (SAIC), Reston, Virginia
NR	*A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants*	NUREG/CR-6451	N/A	August 1997	U.S. Nuclear Regulatory Commission	R. J. Travis, R. E. Davis, E. J. Grove, and M. A. Azarm		Prepared by Brookhaven National Laboratory as document BNL-NUREG-52498.
NR	*Containment Analysis for Type B Packages Used to Transport Various Contents*	NUREG/CR-6487	N/A	November 1996	U.S. Nuclear Regulatory Commission	B. L. Anderson, R. W. Carlson, L. E. Fischer		Prepared by Lawrence Livermore National Laboratory as document number UCRL-ID-124822.

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NR	*Probabilistic Liquefaction Analysis*	NUREG/CR-6622	N/A	November 1999	U.S. Nuclear Regulatory Commission	M. E. Hynes, U.S. Army Corps of Engineers		Prepared by U.S. Army Corps of Engineers, Vicksburg, MS
NR	*Automated Seismic Event Monitoring System (Addendum 1)*	NUREG/CR-6625	N/A	September 2001	U.S. Nuclear Regulatory Commission	I. Henson, R. Wagner, and W. Rivers, Jr., Multimax, Inc.		Prepared by Multimax, Inc. Largo, MD
NR	*Reexamination of Spent Fuel Shipment Risk Estimates (Main Report, Volume1)*	NUREG/CR-6672	N/A	March 2000	U.S. Nuclear Regulatory Commission	J.L. Sprung, D.J. Ammerman, N.L. Breivik, R.J. Dukart, F.L. Kanipe, J.A. Koski, G.S. Mills, K.S. Neuhauser, H.D. Radloff, R.F. Weiner, and H.R. Yoshimura		Prepared by Sandia National Laboratory as document number SAND2000-0234.
NR	*Probabilistic Dose Analysis Using Parameter Distributions Developed For RESRAD and RESRAD-BUILD Codes*	NUREG/CR-6676	N/A	July 2000	U.S. Nuclear Regulatory Commission	S. Kamboj, D. LePoire, E. Gnanapragasam, B. M. Biwer, I. Cheng, J. Arnish, C. Yu, and S. Y. Chen, ANL		Prepared by Argone National Laboratory as document number ANL/EAD/TM- 89. See NUREG/CR-6692 for User Guide and NUREG/CR-6697.
NR	*Proposed Approach for Reviewing Changes to Risk-Important Human Actions*	NUREG/CR-6689	N/A	October 2000	U.S. Nuclear Regulatory Commission	J.C. Higgins, and J.M. O'Hara, BNL		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52598.
NR	*Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes: User Guide*	NUREG/CR-6692	N/A	November 2000	U.S. Nuclear Regulatory Commission	D. LePoire, J. Amish, E. Gnanapragasam, S. Kamboj, B.M. Biwer, J.-J. Cheng, C. Yu, and S.Y. Chen, ANL		Prepared by Argone National Laboratory as document number ANL/EAD/TM-91. See NUREG/CR-6676.
NR	*Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes (Cover – Section 3.13) (Section 4.1 – End)*	NUREG/CR-6697	N/A	December 2000	U.S. Nuclear Regulatory Commission	C. Yu, D. LePoire, E. Gnanapragasam, J. Amish, S. Kamboj, B.M. Biwer, J.-J. Cheng, A. Zielen, and S.Y. Chen, ANL		Prepared by Argone National Laboratory as document number ANL/EAD/TM-98. See NUREG/CR-6672 and NUREG/CR-6676.
NR	*Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel*	NUREG/CR-6700	N/A	January 2001	U.S. Nuclear Regulatory Commission	I.C. Gauld and J.C. Ryman, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/284.
NR	*Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel*	NUREG/CR-6701	N/A	January 2001	U.S. Nuclear Regulatory Commission	I.C. Gauld and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/277.
NR	*Limited Burnup Credit in Criticality Safety Analysis: A Comparison of ISG-8 and Current International Practice*	NUREG/CR-6702	N/A	January 2001	U.S. Nuclear Regulatory Commission	I. C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/72.
NR	*Environmental Effects of Extending Fuel Burnup Above 60 Gwd/MTU*	NUREG/CR-6703	N/A	January 2001	U.S. Nuclear Regulatory Commission	J. V. Ramsdell, Jr., C. E. Beyer, D. D. Lanning, U. P. Jenquin, R. A. Schwarz, D. L. Strenge, P.M. Daling, and R. T. Dahowski, PNNL		Prepared by Pacific Northwest National Laboratory as document number PNNL-13257.

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NR	*Environmental Assessment of Major Revision of 10 CFR Part 71*	NUREG/CR-6711	N/A	2004	U.S. Nuclear Regulatory Commission	D. Hammer and K. Blake, ICF Consulting, Inc.		Prepared by ICF Consulting, Inc, Fairfax, VA. Manuscript Completed: December 2003.
NR	*Summary and Categorization of Public Comments on the Major Revision of 10 CFR Part 71*	NUREG/CR-6712	N/A	April 2002	U.S. Nuclear Regulatory Commission	D. Hammer, K. Blake, L. Massar, S. Matheson, and E. Piendak, ICF Consulting, Inc.		Prepared by ICF Consulting, Inc, Fairfax, VA.
NR	*Regulatory Analysis of Major Revision of 10 CFR Part 71*	NUREG/CR-6713	N/A	2004	U.S. Nuclear Regulatory Commission	D. Hammer and K. Blake, ICF Consulting, Inc.		Prepared by ICF Consulting, Inc, Fairfax, VA. Manuscript Completed: December 2003.
NR	*Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks*	NUREG/CR-6716	N/A	March 2001	U.S. Nuclear Regulatory Commission	S.M. Bowman, I.C. Gauld, and J.C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/385.
NR	*Zinc-Zircaloy Interaction in Dry Storage Casks*	NUREG/CR-6732	N/A	August 2001	U.S. Nuclear Regulatory Commission	H. Tsai and Y. Yan, ANL		Prepared by Argone National Laboratory as document number ANL-01/18.
NR	*Dry Cask Storage Characterization Project - Phase 1: CASTOR V/21 Cask Opening and Examination*	NUREG/CR-6745	N/A	September 2001	U.S. Nuclear Regulatory Commission	W.C. Bare and L.D. Torgerson, INL		Prepared by Idaho National (Engineering an Environmental) Laboratory as document number INEEL/EXT-01-00183.
NR	*Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*	NUREG/CR-6747	N/A	October 2001	U.S. Nuclear Regulatory Commission	J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/306.
NR	*STARBUCS: A Prototypic SCALE Control Module for Automated Criticality Safety Analyses Using Burnup Credit*	NUREG/CR-6748	N/A	October 2001	U.S. Nuclear Regulatory Commission	I. C. Gauld and S. M. Bowman, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/33.
NR	*Parametric Study of Effect of Control Rods for PWR Burnup Credit*	NUREG/CR-6759	N/A	February 2002	U.S. Nuclear Regulatory Commission	C. E. Sanders and J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/69.
NR	*Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit*	NUREG/CR-6760	N/A	March 2002	U.S. Nuclear Regulatory Commission	C. E. Sanders and J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000-321.
NR	*Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit*	NUREG/CR-6761	N/A	March 2002	U.S. Nuclear Regulatory Commission	J.C. Wagner and C.V. Parks		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2000/373.
NR	*Burnup Credit PIRT Report*	NUREG/CR-6764	N/A	May 2002	U.S. Nuclear Regulatory Commission	G. H. Bidinger, R. J. Cacciapouti, J. M. Conde, P. Cousinou, T. Doering, P. Grimm, H-R Hwang, W. J. Lee, D. Marloye, R. L. Murray, J-C Neuber, M. B. Raap, J. Saptya, D. A. Thomas, S. Turner, R. E. Wilson, B. Boyack, and D. J. Diamond, BNL		Prepared by Brookhaven National Laboratory as document number BNL-NUREG-52654.

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NR	*Evaluation of Hydrologic Uncertainty Assessments for Decommissioning Sites Using Complex and Simplified Models*	NUREG/CR-6767	N/A	April 2002	U.S. Nuclear Regulatory Commission	P.D. Meyer and S. Orr, PNNL		Prepared by Pacific Northwest National Laboratory as document number PNNL-13832.
NR	*Spent Nuclear Fuel Transportation Package Performance Study Issues Report*	NUREG/CR-6768		June 2002	U.S. Nuclear Regulatory Commission	J.L. Sprung, D.J. Ammerman, J.A. Koski, R.F. Weiner		Prepared by Sandia National Laboratory as document number SAND2001-08211P.
NR	*Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Development of Hazard- & Risk-Consistent Seismic Spectra for Two Sites (Cover – Page 5-26) and (Section 6.0 – End)*	NUREG/CR-6769	N/A	April 2002	U.S. Nuclear Regulatory Commission	R. K. McGuire, Rick Engineering, Inc. W. J. Silva, Pacific Engineering & Analysis C. J. Costantino, Consultant		Performe by Risk Engineering, Inc.,Boulder, CO with subcontactors: Pacific Engineering & Analysis El Cerrito, CA and Carl J. Costantino, Consultant, Valley, NY.
NR	*Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses*	NUREG/CR-6781	N/A	January 2003	U.S. Nuclear Regulatory Commission	J.C. Wagner and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/272.
NR	*Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs*	NUREG/CR-6800	N/A	March 2003	U.S. Nuclear Regulatory Commission	J.C. Wagner, C.E. Sanders, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2002/6.
NR	*Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*	NUREG/CR-6801	N/A	March 2003	U.S. Nuclear Regulatory Commission	J.C. Wagner, M. D. DeHart, and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/273.
NR	*Recommendations for Shielding Evaluations for Transport & Storage Packages*	NUREG/CR-6802	N/A	May 2003	U.S. Nuclear Regulatory Commission	B.L. Broadhead, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2002/31.
NR	*A Comprehensive Strategy of Hydrogeologic Modeling and Uncertainty Analysis for Nuclear Facilities and Sites*	NUREG/CR-6805	N/A	July 2003	U.S. Nuclear Regulatory Commission	S.P. Neuman and P.J. Wierenga, University of Arizona		Prepared by University of Arizona, Department of Soil, Water and Environmental Science.
NR	*Strategies for Application of Isotopic Uncertainties in Burnup Credit*	NUREG/CR-6811	N/A	June 2003	U.S. Nuclear Regulatory Commission	I. C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2001/257.
NR	*Examination of Spent PWR Fuel Rods After 15 Years in Dry Storage*	NUREG/CR-6831	N/A	September 2003	U.S. Nuclear Regulatory Commission	R.E. Einziger, H. Tsai, and M.C. Billone, ANL, B.A. Hilton, ANL-West		Prepared by Argone National Laboratory as document number ANL-03/17.
NR	*Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks*	NUREG/CR-6835	N/A	September 2003	U.S. Nuclear Regulatory Commission	K.R. Elam, J.C. Wagner, and C.V. Parks, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2002/255.
NR	*The Technical Basis for the NRC's Guidelines for External Risk Communication*	NUREG/CR-6840	N/A	January 2004	U.S. Nuclear Regulatory Commission	L. Peterson, E. Specht, and E. Wight, WPI		Prepared by WPI, Gaithersburg, MD.
NR	*A Risk-Informed Basis for Establishing Non-Fixed Surface Contamination Limits for Spent Fuel Transportation Casks*	NUREG/CR-6841	N/A	April 2004	U.S. Nuclear Regulatory Commission	R.R. Rawl, K.F. Eckerman, and J.S. Bogard		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2003/225.

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NR	*Combined Estimation of Hydrogeologic Conceptual Model and Parameter Uncertainty*	NUREG/CR-6843	N/A	March 2004	U.S. Nuclear Regulatory Commission	P.D. Meyer, M. Ye, and K.J. Cantrell, PNNL S.P. Neuman, University of Arizona		Prepared by Pacific Northwest National Laboratory as document number PNNL-14534.
NR	*Sensitivity Analysis Applied to the Validation of the ^{10}B Capture Reaction in Nuclear Fuel Casks*	NUREG/CR-6845	N/A	August 2004	U.S. Nuclear Regulatory Commission	S. Goluoglu, K.R. Elam, B.T. Rearden, B.L. Broadhead, and C.M. Hopper, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2004/48.
NR	*Identification and Analysis of Factors Affecting Emergency Evacuations (Main Report, Volume 1) (Appendices, Volume 2)*	NUREG/CR-6864	N/A	Jnauary 2005	U.S. Nuclear Regulatory Commission	L.J. Dotson and J. Jones		Prepared by Sandia National Laboratory as document number SAND2004-5901.
NR	*Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Cask Storage Systems*	NUREG/CR-6865	N/A	February 2005	U.S. Nuclear Regulatory Commission	V.K. Luk and B.W. Spencer, SNL I.P. Lam, Earth Mechanics, Inc. R.A. Dameron, David Evans & Associates, Inc.		Prepared by Sandia National Laboratory as document number SAND2004-5794P with subontractors: Earth Mechanics, Inc, .Fountain Valley, California and David Evans & Associates, Inc., San Diego, California
NR	*Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario*	NUREG/CR-6886	2	February 2009	U.S. Nuclear Regulatory Commission	H. E. Adkins, Jr., J. M. Cuta, B. J. Koeppel, and A. D. Guzman (PNNL) C. S. Bajwa (NRC)		Prepared by Pacific Northwest National Laboratory as document number PNNL-15313.
NR	*Spent Fuel Transportation Package Response to the Caldecott Tunnel Fire Scenario*	NUREG/CR-6894	1	January 2007	U.S. Nuclear Regulatory Commission	H.E. Adkins, Jr., B.J. Koeppel, J.M. Cuta, and A.D. Guzman, PNNL C. S. Bajwa (NRC)		Prepared by Pacific Northwest National Laboratory as document number PNNL-15346.
NR	*Integrated Ground-Water Monitoring Strategy for NRC-Licensed Facilities and Sites: Logic, Strategic Approach and Discussion (Logic, Strategic Approach and Discussion, Volume 1) (Case Study Applications, Volume 2)*	NUREG/CR-6948	N/A	November 2007	U.S. Nuclear Regulatory Commission	V. Price, T. Temples, J. Tauxe, R. Hodges, and R. Falta, Advanced Environmental Solutions, LLC for V1. V. Price, T. Temples, R. Hodges, Z. Dai, D. Watkins, and J. Imrich, Advanced Environmental Solutions, LLC for V2.		Prepared by Advanced Environmental Solutions, LLC, Lexington, SC.
NR	*Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit*	NUREG/CR-6951	N/A	January 2008	U.S. Nuclear Regulatory Commission	G. Radulescu, D.E. Mueller, and J.C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2006/87.
NR	*Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*	NUREG/CR-6955	N/A	January 2008	U.S. Nuclear Regulatory Commission	J.C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2004/52.
NR	*Spent Fuel Decay Heat Measurements Performed at the Swedish Central Interim Storage Facility*	NUREG/CR-6971	N/A	February 2010	U.S. Nuclear Regulatory Commission	B.D. Murphy and I.C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2008/016.

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NR	*Validation of SCALE 5 Decay Heat Predictions for LWR Spent Nuclear Fuel*	NUREG/CR-6972	N/A	February 2010	U.S. Nuclear Regulatory Commission	I.C. Gauld, G. Ilas, B.D. Murphy, and C.F. Weber, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2008/015.
NR	*Review of Information for Spent Nuclear Fuel Burnup Confirmation*	NUREG/CR-6998	N/A	December 2009	U.S. Nuclear Regulatory Commission	B.B. Bevard, J.C. Wagner and C.V. Parks, ORNL M. Aissa, NRC		Prepared by Oak Ridge National Laboratory as document number ORNLITM-2007/229.
NR	*Technical Basis for a Proposed Expansion of Regulatory Guide 3.54 — Decay Heat Generation in an Independent Spent Fuel Storage Installation*	NUREG/CR-6999	N/A	February 2010	U.S. Nuclear Regulatory Commission	I.C. Gauld and B.D. Murphy, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2007/231.
NR	*Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel*	NUREG/CR-7012	N/A	January 2011	U.S. Nuclear Regulatory Commission	I.C. Gauld, G. Ilas, and G. Radulescu, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2010/41.
NR	*Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor*	NUREG/CR-7013	N/A	January 2011	U.S. Nuclear Regulatory Commission	G. Ilas and I.C. Gauld, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNLITM-2009/321.
NR	*Human Reliability Analysis-Informed Insights on Cask Drops*	NUREG/CR-7016	N/A	February 2012	U.S. Nuclear Regulatory Commission	J. Brewer, S. Hendrickson, and R. Boring, SNL S. Cooper, NRC		Prepared by Sandia National Laboratory as document number SAND2010-8463P.
NR	*Preliminary, Qualitative Human Reliability Analysis for Spent Fuel Handling*	NUREG/CR-7017	N/A	February 2012	U.S. Nuclear Regulatory Commission	Jeffrey D. Brewer and Stacey M. L. Hendrickson, SNL Paul J. Amico, SAIC Susan E. Cooper, NRC		Prepared by Sandia National Laboratory as document number SAND2010-8464P.
NR	*Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments*	NUREG/CR-7030	N/A	October 2010	U.S. Nuclear Regulatory Commission	L. Caseres and T.S. Mintz, Southwest Research Institute		Prepared by Southwest Research Institute, San Antonio, TX
NR	*Analysis of Severe Railway Accidents Involving Long Duration Fires*	NUREG/CR-7034	N/A	February 2011	U.S. Nuclear Regulatory Commission	G. Adams, T. Mintz, M. Necsoiu, and J. Mancillas, Nuclear Waste Regulatory Analyses		Prepared by Center for Nuclear Waste Regulatory Analyses, San Antonio, TX
NR	*Analysis of Severe Roadway Accidents Involving Long Duration Fires*	NUREG/CR-7035	N/A	February 2011	U.S. Nuclear Regulatory Commission	G. Adams, and T. Mintz, Nuclear Waste Regulatory Analyses		Prepared by Center for Nuclear Waste Regulatory Analyses, San Antonio, TX
NR	*Structural Materials Analyses of the Newhall Pass Tunnel Fire, 2007*	NUREG/CR-7101	N/A	June 2011	U.S. Nuclear Regulatory Commission	K. Axler, T.S. Mintz, and K. Das, Nuclear Waste Regulatory Analyses J. Huczek, Southwest Research		Prepared by Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute, San Antonio, TX.

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions*	NUREG/CR-7108	N/A	April 2012	U.S. Nuclear Regulatory Commission	G. Radulescu, I. C. Gauld, G. llas, and J. C. Wagner, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2011/509.
NR	*An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions*	NUREG/CR-7109	N/A	April 2012	U.S. Nuclear Regulatory Commission	J. M. Scaglione, D. E. Mueller, J. C. Wagner, and W. J. Marshall, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2011/514.
NR	*Performance of Metal and Polymeric O-Ring Seals in Beyond-Design-Basis Temperature Excursions*	NUREG/CR-7115	N/A	April 2012	U.S. Nuclear Regulatory Commission	Jiann C. Yang and Edward J. Hnetkovsky, National Institute of Standards and Technology, Engineering Laboratory		Prepared by National Institute of Standards and Technology, Engineering Laboratory Gaithersburg, Maryland.
NR	*Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel*	NUREG/CR-7116	N/A	November 2011	U.S. Nuclear Regulatory Commission	R. L. Sindelar, A.J. Duncan, M.E. Dupont, P.-S. Lam, M.R. Louthan, Jr., and T.E. Skidmore, SRNL		Prepared by Savannah River National Laboratory as document number SRNL-STI-2011-00005.
NR	*Experimental Studies of Reinforced Concrete Structures Under Multi-Directional Earthquakes and Design Implications*	NUREG/CR-7119	N/A	July 2013	U.S. Nuclear Regulatory Commission	N. Simos and C. H. Hofmayer, BNL		Prepared by Brookhaven National Laboratory.
NR	*Review and Prioritization of Technical Issues Related to Burnup Credit for BWR Fuel*	NUREG/CR-7158	N/A	February 2013	U.S. Nuclear Regulatory Commission	D. E. Mueller, S. M. Bowman, W. J. Marshall, and J. M. Scaglione, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2012/261.
NR	*Analysis of Experimental Data for High Burnup BWR Spent Fuel Isotopic Validation – SVEA-96 and GE14 Assembly Designs*	NUREG/CR-7162	N/A	March 2013	U.S. Nuclear Regulatory Commission	H. J. Smith, I. C. Gauld, and U. Mertyurek, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2013/18.
NR	*Thermal Analysis of Horizontal Storage Casks for Extended Storage Applications*	NUREG/CR-7191	N/A	December 2014	U.S. Nuclear Regulatory Commission	Kaushik Das, Debashis Basu, and Gary Walter, Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute		Prepared by Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute, San Antonio, TX.
NR	*Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems*	NUREG/CR-7194	N/A	Arpil 2015	U.S. Nuclear Regulatory Commission	William (B.J.) Marshall, Brian J. Ade, Stephen M. Bowman, Ian C. Gauld, Germina Ilas, Ugur Mertyurek, and Georgeta Radulescu, ORNL		Prepared by Oak Ridge National Laboratory as document number ORNL/TM-2014/240.
NR	*Final Envronmental Statement on the Transportation of Radioactive Material by Air and Other Modes, Volume 1 and Volume 2*	NUREG-0170	N/A	December 1977 (V1 & V2)	U.S. Nuclear Regulatory Commission	N/A		Docket No. PR-71, 73 (40 FR 23768)
NR	*Directory of Certificates of Compliance for Radioactive Materials Packages (Certificates of Compliance, Volume 2,(Report of NRC-Approved Quality Assurance Programs for Radioactive Materials Packages, Volume 3)*	NUREG-0383	28 (V2) 24 (V3)	October 2013 (V2) January 2009 (V3)	U.S. Nuclear Regulatory Commission	N/A		Volume 1 is for internal (USNRC) use only.
NR	*Physical Protection of Shipments of Irradiated Reactor Fuel: Final Report*	NUREG-0561	2	April 2013	U.S. Nuclear Regulatory Commission	A. G. Garrett, S. L. Garrett, PNNL A. S. Giantelli, R. C. Ragland, USNRC		Prepared by Pacific Northwest National Laboratory

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36*	NUREG-0612	N/A	July 1980	U.S. Nuclear Regulatory Commission	Henry J. George, Task (A-36) Manager, and members of A-36 Task Force		
NR	*Functional Criteria for Emergency Response Facilities*	NUREG-0696	N/A	February 1981	U.S. Nuclear Regulatory Commission	N/A		
NR	*Human-System Interface Design Review Guidelines*	NUREG-0700	2	May 2002	U.S. Nuclear Regulatory Commission	J. M. O'Hara and W. S. Brown, BNL P. M. Lewis and J. J. Persensky, NRC		NUREG-0700, Revision 2, is divided into 3 PDF Documents: Cover – Page 200, Page 201 – Page 400, and Page 401 – End
NR	*Human Factors Engineering Program Review Model*	NUREG-0711	3	November 2012	U.S. Nuclear Regulatory Commission	J. M. O'Hara and J.C. Higgins, BNL S. A. Fleger and P. A. Pieringer, NRC		
NR	*Public Information Circular for Shipments of Irradiated Reactor Fuel*	NUREG-0725	15	May 2010	U.S. Nuclear Regulatory Commission	A. G. Garrett, S. L. Garrett, and R. G. Ostler, PNNL K. Jamgochian, NRC		
NR	*Methodology for Evaluation of Emergency Response Facilities (Draft for Comment)*	NUREG-0814	N/A	August 1981	U.S. Nuclear Regulatory Commission	S. Ramos, NRC		Draft for Comment
NR	*Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data*	NUREG-0917	N/A	July 1982	U.S. Nuclear Regulatory Commission	W. Snell		
NR	*Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment: 10 CFR Part 74, Subpart E*	NUREG-1280	1	April 1995	U.S. Nuclear Regulatory Commission	N/A		
NR	*Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities (Main Report, Volume 1), (Appendices A and B, Volume 2), (Appendices C-H, Volume 3)*	NUREG-1496	N/A	July 1997	U.S. Nuclear Regulatory Commission	N/A		NUREG-1496 is divided into 3 PDF Documents: Main Report (Volume 1), Appendices A & B (Volume 2), Appendices C-H (Volume 3)
NR	*Background as a Residual Radioactivity Criterion for Decommissioning: Appendix A to the Draft Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities (Draft Report of Comment)*	NUREG-1501	N/A	August 1994	U.S. Nuclear Regulatory Commission	A.M. Huffert, R.A. Meck, K. M. Miller, Environmental Measurements Laboratory		Draft for Comment
NR	*A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys (Interim Draft Report for Comment and Use)*	NUREG-1505	1	June 1998	U.S. Nuclear Regulatory Commission	C.V. Gogolak, Environmental Measurements Laboratory G.E. Powers, A.M. Huffert, NRC		Interim Draft Report for Comment and Use
NR	*Integrated Safety Analysis Guidance Document*	NUREG-1513	N/A	May 2001	U.S. Nuclear Regulatory Commission	R.I. Milstein, NRC		
NR	*Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*	NUREG-1520	1	May 2010	U.S. Nuclear Regulatory Commission	N/A		

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
NR	*Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*	NUREG-1536	1	July 2010	U.S. Nuclear Regulatory Commission	N/A		Complements 10CFR72.
NR	*Standard Review Plan for Spent Fuel Dry Storage Facilities*	NUREG-1567	N/A	March 2000	U.S. Nuclear Regulatory Commission	N/A		Complements 10CFR72.
NR	*Standard Review Plan for Transportation Packages for Radioactive Material*	NUREG-1609	N/A	March 1999	U.S. Nuclear Regulatory Commission	N/A		Initial Report, Supplement 1 is *Standard Review Plan for Transportation Packages for MOX-Radioactive Material*, Supplement 2 is *Standard Review Plan for Transpor-tation Packages for Irradiated Tritium-Producing Burnable Absorber Rods (TPBARs)*
NR	*Standard Review Plan for Transportation Packages for Spent Nuclear fuel*	NUREG-1617	N/A	March 2000	U.S. Nuclear Regulatory Commission	N/A		
NR	*Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel*	NUREG-1617, Supplemental 1	N/A	September 2005	U.S. Nuclear Regulatory Commission	R.S. Hafner, G.C. Mok, J. Hovingh, C. K. Syn, E.W. Russell, S.C. Keaton, J.L. Boles, D.K. Vogt, P. Prassinos		
NR	*Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)*	NUREG-1624	1	May 2000	U.S. Nuclear Regulatory Commission	N/A		NUREG-1624, Rev 1, is broken out into 2 PDF files: Cover – Section 11 and Appendices A – G. See NUREG-1880 and NUREG/CR-6350.
NR	*Radiological Assessments for Clearance of Materials from Nuclear Facilities (Main Report, Cover through Chapter 8, with Errata, Volume 1), (Appendices A–E, with Errata, Volume 2), (Appendices F–G, Volume 3), and (Appendices H–O, with Errata, Volume 4)*	NUREG-1640	N/A	June 2003(V1) October 2004 (V2) June 3003 (V3) May 2004 (V4)	U.S. Nuclear Regulatory Commission	R. Anigstein, H.J. Chmelynski, D.A. Loomis, J.J. Mauro, R.H. Olsher, and W.C. Thurber, SC&A, Inc., S.F. Marschke,Gemini Consulting Company, R.A. Meck, NRC		Volume 3 is broken out into 2 PDF files: Appendix F and Appendix G.

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NR	*U.S.-Specific Schedules of Requirements for Transport of Specified Types of Radioactive Material Consignments*	NUREG-1660	N/A	January 14, 1999	U.S. Nuclear Regulatory Commission, U.S. Department of Transportation	J. Cook, E. Easton, USNRC R. Boyle, DOT R. Pope, ORNL B. Dodd, D. Harlan, OSU		OSU is Oregen State University.
NR	*Final Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah (Cover to Appendix F, Volume 1), and (Appendix G, "Public Comments and Responses," and Appendix H, "Index of Commenters", Volume 2)*	NUREG-1714	N/A	December 2001	U.S. Nuclear Regulatory Commission	N/A		Private Fuel Storage, L.L.C., Docket 72.22
NR	*NMSS Decommissioning Standard Review Plan*	NUREG-1727	N/A	September 2000	U.S. Nuclear Regulatory Commission	N/A		
NR	*Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation*	NUREG-1736	N/A	October 2001	U.S. Nuclear Regulatory Commission	R.E. Zelac, J.L. Cameron, H. Karagiannis, J.R. McGrath, S.S. Sherbini, M.L. Thomas, and J.E. Wigginton		NUREG-1736 is broken out into 2 PDF files: Cover – Chapter 3 and Appendix A – End.
NR	*Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance*	NUREG-1745	N/A	June 2001	U.S. Nuclear Regulatory Commission	N/A		
NR	*Environmental Review Guidance for Licensing Actions Associated with NMSS Programs*	NUREG-1748	N/A	August 2003	U.S. Nuclear Regulatory Commission	N/A		
NR	*Consolidated Decommissioning Guidance (Decommissioning Process for Materials Licensees, Volume 1), (Characterization, Survey, and Determination of Radiological Criteria, Volume 2), (Financial Assurance, Recordkeeping, and Timeliness, Volume 3)*	NUREG-1757	2 (V1) 1 (V2) 1 (V3)	September 2006 (V1) September 2006 (V2) February 2012 (V3)	U.S. Nuclear Regulatory Commission	K.L. Banovac, J.T. Buckley, R.L. Johnson, G.M. McCann, J.D. Parrott, D.W. Schmidt, J.C. Shepherd, T.B. Smith, P.A. Sobel, B.A. Watson, D.A. Widmayer, T.H. Youngblood for V1; D.W. Schmidt, K.L. Banovac, J.T. Buckley, D.W. Esh, R.L. Johnson, J.J. Kottan, C.A. McKenney, T.G. McLaughlin, S. Schneider for V2; K.M. Kline, C.M. Dean (ICF International), T.L. Fredrichs, M.C. Maier, E.R. Pogue, and R.N. Young (TN Department of Environment and Conservation) for V3.		

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NR	*Radiological Surveys for Controlling Release of Solid Materials(Draft Report for Comment)*	NUREG-1761	N/A	July 2002	U.S. Nuclear Regulatory Commission	E.W. Abelquist and T.J. Bower, Oak Ridge Institute for Science and Education C.V. Gogolak and P. Shebell, U.S. Department of Energy R. Coleman, ORNL G.E. Powers,NRC		Draft Report for Comment
NR	*United States Nuclear Regulatory Commission Package Performance Study Test Protocols (Draft Report for Comment)*	NUREG-1768	N/A	February 2003	U.S. Nuclear Regulatory Commission	N/A		Draft Report for Comment
NR	*Environmental Impact Statement for the Proposed Idaho Spent Fuel Facility at the Idaho National Engineering and Environmental Laboratory in Butte County, Idaho*	NUREG-1773	N/A	January 2004	U.S. Nuclear Regulatory Commission	N/A		
NR	*Good Practices for Implementing Human Reliability Analysis (HRA)*	NUREG-1792	N/A	Arpil 2005	U.S. Nuclear Regulatory Commission	A. Kolaczkowski, Science Applications International Corporation, J. Forester, SNL E. Lois and S. Cooper, NRC		
NR	*Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, pluse ERRATA and Supplement 1, Volume 1 and Volume 2 (Appendices)*	NUREG-1805	N/A	December 2004 (Main Report) July 2013 (Supplement1, V1 & V2)		Naeem Iqbal, Mark Henry Salley, and Sunil Weerakkody, NRC for Main Report. D. Stroup, G. Taylor, G. Hausman, and M. H. Salley, NRC for Supplement 1, Volume 1 and Volume 2		Supplement 1, Volume 2 contains appendices.
NR	*Evaluation of Human Reliability Analysis Methods Against Good Practices*	NUREG-1842	N/A	September 2006	U.S. Nuclear Regulatory Commission	J. Forester, SNL A. Kolaczkowski and D. Kelly, Science Applications International Corporation E. Lois, NRC		
NR	*ATHEANA User's Guide*	NUREG-1880	N/A	June 2007	U.S. Nuclear Regulatory Commission	John Forester, SNL Alan Kolaczkowski, Science Applications International Corporation, Dennis Bley, Buttonwood Consulting, Inc., Susan Cooper and Erasmia Lois, NRC		User's guide for the human reliability analysis (HRA) method known as "A Technique for Human Event Analysis" (ATHEANA) See NUREG-1624 and NUREG/CR-6350.
NR	*Joint Canada - United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages*	NUREG-1886	N/A	March 2009	U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards	M. Rahimi, N.L. Osgood, and M. Sampson, USNRC M. Conroy, DOT S. Faille and K. Glenn, CNSC		This guide is published in Canada as RD-364. Canadian Nuclear Safety Commission (CNSC), U.S. DOT-Pipeline and Hazardous Materials Safety Administration Packaging and Transport Licensing Division

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NR	*RASCAL 3.0.5: Description of Models and Methods*	NUREG-1887	N/A	August 2007	U.S. Nuclear Regulatory Commission	S.A. McGuirea, NRC J.V. Ramsdell, Jr., PNNL G.F. Athey, Athey Consulting		Code used by NRC's emergency operations center for making dose projections for atmospheric releases during radiological emergencies.
NR	*RASCAL 3.0.5 Workbook*	NUREG-1889	N/A	September 2007	U.S. Nuclear Regulatory Commission	G.F. Athey, Athey Consulting S.A. McGuire, NRC J.V. Ramsdell, Jr., PNNL		See NUREG-1887. Workbook contains problems designed to familiarize the user with the RASCAL software through hands-on problem solving.
NR	*Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance*	NUREG-1927	N/A	March 2011	U.S. Nuclear Regulatory Commission	N/A		
NR	*Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,* Draft Report for Comment	NUREG-1927	1 (Draft)	Month? 2015	U.S. Nuclear Regulatory Commission	N/A		Draft Revison 1 became available March 9, 2015 for "Internal Review and Coordination." Purpose is to provide the Advisory Committee on Reactor Safeguards (ACRS) a draft copy for ACRS's Metallurgy & Reacor Fuels subcommittee meeting scheduled, April 8, 2015.
NR	*RASCAL 4.3: Description of Models and Methods (Initial Report and Supplement 1)*	NUREG-1940	N/A	December 2012 (Initial Report) Supplement 1 (May 2015)	U.S. Nuclear Regulatory Commission	J.V. Ramsdell, Jr., PNNL G.F. Athey, Athey Consulting, S.A. McGuire, NRC (Retired) L.K. Brandon, NRC for Initiial Report. J. V. Ramsdell, Jr.Ramsdell Environmental Consulting G. F. Athey, Athey Consulting J. P. Rishel, PNNL for Supplement 1.		Updated Version of RASCAL., See NUREG 1887 and NUREG 1889 for RASCAL 3.0.5 information.
NR	*Intrusion Detection Systems and Subsystems: Technical Information for NRC Licensees*	NUREG-1959	N/A	March 2011	U.S. Nuclear Regulatory Commission	N/A		
NR	*Access Control Systems: Technical Information for NRC Licensees*	NUREG-1964	N/A	Arpil 2011	U.S. Nuclear Regulatory Commission	N/A		
NR	*Central and Eastern United States Seismic Source Characterization for Nuclear Facilities (Chapters 1 to 4, Volume 1) (Chapters 5 to 7, Volume 2), (Chapters 8 to 11, Volume 3), (Appendices A and B, Volume 4), (Appendices C to F, Volume 5), (Appendices G to L, Volume 6)*	NUREG-2115	N/A	January 2012	U.S. Nuclear Regulatory Commission	N/A		NUREG-2115 was co-sponsored by U.S. Department of Energy, Electric Power Research Institute, and U.S. Nuclear Regulatory Commission

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NR	*Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies*	NUREG-2117	1	April 2012	U.S. Nuclear Regulatory Commission	Annie M. Kammerer and Jon P. Ake, NRC		
NR	*Spent Fuel Transportation Risk Assessment*	NUREG-2125	N/A	January 2014	U.S. Nuclear Regulatory Commission	N/A		
NR	*A Proposed Risk Management Regulatory Framework*	NUREG-2150	N/A	April 2012	U.S. Nuclear Regulatory Commission	Commissioner George Apostolakis, M. Cunningham, C.		
NR	*Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications*	NUREG-2152	N/A	March 2013	U.S. Nuclear Regulatory Commission	Ghani Zigh and Jorge Solis, NRC		
NR	*Acceptability of Corrective Action Programs for Fuel Cycle Facilities: Draft Report for Comment*	NUREG-2154	N/A	January 2013	U.S. Nuclear Regulatory Commission	Sabrina Atack, NRC		Draft Report for Comment
NR	*Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel:Final Report, Volume 1 and Volume 2*	NUREG-2157	N/A	September 2014	U.S. Nuclear Regulatory Commission	N/A		Volume 2 includes public comments.
NR	*Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Casks*	NUREG-2174	N/A	February 2015	U.S. Nuclear Regulatory Commission	N/A		
RD	*EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project*	3002000717	N/A	June 2013	Electric Power Research Institute	N/A		Supports NUREG-2115.
RD	*Yankee Atomic Electric Co. v. United States*	*536 F.3d 1268*	N/A	August 7, 2008	United States Court of Federal Claims	James F. Merow	R 0.1.4	
RD	*Civilian Radioactive Waste Management System Requirements Document*	A00000000 – 00811-1708 – 00003	3	November 1996	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		
RD	*Projection of Future Spent Nuclear Fuel Discharges*	CAL-WAT-SE-000007	0	2009	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		Not able to locate copy of document.
RD	*Commission Order: Memorandum and Order on Continued Storage*	CLI-14-08	N/A	August 26, 2014	U.S. Nuclear Regulatory Commission	N/A		
RD	Safety Evaluation Report of the Site-Related Aspects of the Private Fuel Storage Facility, Independent Spent Fuel Storage Installation	Docket No. 72-0022	N/A	Unknown	U.S. Nuclear Regulatory Commission	N/A		Proposed Private Fuel Storage Facility that was to be located in Tooele County, Utah.
RD	*Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (FEIS)*	DOE/EIS-0250	N/A	February 2002	DOE - Office of Civilian Radioactive Waste Management (OCRWM)	N/A	R 1.0.3	
RD	*Civilian Radioactive Waste Management System Requirements Document*	DOE/RW-0406	8	September 12, 2007	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		

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RD	*Acceptance Priority Ranking and Annual Capacity Report*	DOE/RW-0567	N/A	July 2004	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		HQO.20041004.0041
RD	*Used Nuclear Fuel Storage and Transportation Research, Development , and Demonstration Plan*	FCRD-FCT-2012 000053	0	April 2012	U.S. Department of Energy, Fuel Cycle Reseach & Development	N/A	R 2.0.5	
RD	*Status Report on Design Concept Trade Studies: Fleet Maintenance Functional Requirements and Technical Trade Studies*	FCRD-NFST 2014-000542	N/A	?	U.S. Department of Energy, Fuel Cycle Reseach & Development	N/A		
RD	*Used Fuel Management System Architecture Evaluation, Fiscal Year 2012*	FCRD-NFST-2013-000020	0	October 31, 2012	U.S. Department of Energy	M. Nutt, E. Moris, F. Puig, J. Carter, P. Rodwell, A. Delley, R. Howard, D. Giuliano		
RD	*A Project Concept for Nuclear Fuels Storage and Transportation*	FCRD-NFST-2013-000132	1	June 15, 2013	U.S. Department of Energy	J. T. Carter (SRNL), S. Dam & et al (Tech Source, Inc.)		
RD	*Nuclear Fuels Storage and Transportation Planning Project Inventory Basis*	FCRD-NFST-2013-000263	0	August 30, 2013	U.S. Department of Energy	J. T. Carter, D. R. Leduc	R 0.1, R 0.1.4, R 0.2, R 0.11, R 0.12, R 2.0.3	
RD	*Preliminary Site Factors and Considerations for Interim Used Fuel Storage Facilities*	FCRD-NFST-2013-000370	0	September 23, 2013	U.S. Department of Energy, Fuel Cycle Reseach & Development	N/A	R 2.0.4	
RD	*Used Fuel Management System Architecture Evaluation, Fiscal Year 2013*	FCRD-NFST-2013-000377	1	October 31, 2013	U.S. Department of Energy	M. Nutt, F. Puig, E. Morris, Y. Park, R. Joseph III, G. Peterson, D. Giuliano, R. Howard		
RD	*AAR S-2043 Cask Railcar System Requirements Document*	FCRD-NFST-2014-000093	N/A	March 31, 2014	U.S. Department of Energy, Fuel Cycle Research & Development	Feldman, M., Maheras, S. J., Best, R.E.		
RD	*Dry Storage of Used Fuel Transition to Transport*	FCRD-UFD-2012-000253	0	August 2012	Savannah River National Laboratory	D. R. Leduc		
RD	*Used Fuel Research and Development Extended Used Fuel Storage R&D Functions and Requirements*	FCRD-USED-2011-000030	0c	December 2010	U.S. Department of Energy	N/A		
RD	*The Gap Analysis to Support Extended Storage of Used Nuclear Fuel*	FCRD-USED-2011-000136	0	January 31, 2013	U.S. Department of Energy, Used Fuel Division	Brady Hanson & et. al.	R 2.0.5	
RD	*Consolidated Storage Lessons Learned and Background Information*	FCRD-USED-2011-000345	0	September 13, 2011	U.S. Department of Energy	J. Carter, A. Delley, T. Cotton		
RD	*Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States*	ISBN: 0-309-65316-9	N/A	2006	National Research Council, Committee on Transportation of Radioactive Waste	N/A	R 1.0.3	

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RD	*Task Order 11 – Development of Consolidated Fuel Storage Facility Concepts*	N/A	N/A	February 1, 2013	Energy Solutions submitted to DOE under Contract No. DE-NE0000293	N/A		
RD	*Final Report, Task Order 11 – Development of Consolidated Fuel Storage Facility Concepts*	N/A	N/A	January 31, 2013	Shaw Environmental & Infrastructure, Inc. submitted to DOE under Contract No. DE-NE0000292	N/A		
RD	*Blue Ribbon Commission on America's Nuclear Future Report to the Secretary of Energy*	N/A	N/A	January 2012	Blue Ribbon Commission on America's Nuclear Future	N/A	R 0.6, R 2.0.3, R 2.0.5	
RD	*Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel NWTRB report*	N/A	N/A	December 2010	U.S. Department of Energy	N/A		
RD	*Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel*	N/A	N/A	December 2010	United States Nuclear Waste Technical Review Board	Douglas B. Rigby		
RD	*Task Order 11 – Development of Consolidated Fuel Storage Facility Concepts Report*	RPT-3008097-000	N/A	February 12, 2013	Areva Federal Services LLC submitted to DOE under Contract No. DE-NE0000291	N/A		Good Summary of requirement of 10CFR72 on P 7-9.
RD	*Nuclear Waste Administration Act of 2013*	S. 1240 (113th Congress)	N/A	Not Enacted	N/A	N/A	R 2.0.3	Nuclear Waste Administration Act of 2015 introduced to replace S1240.
RD	*Calculation Method for the Projection of Future Spent Nuclear Fuel Discharges*	TDR-WAT-NU-000002	2	August 2005	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	B. McLeod		DOC.20051024.0002 (WBS: 3.2.3) was prepared by Bechtel SAIC Company, LLC under contact DE-AC08-01NV12101
RD, PG	*Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*	N/A	N/A	January 2013	U.S. Department of Energy	N/A	R 0.1, R 0.1.1, R 0.1.2, R 0.1.3, R 0.7, R 0.8, R 2.0.1, R 2.0.2, R 2.0.3	
RD, RR	*Federal Water Pollution Control Amendments of 1972 (Clean Water Act)*	33 U.S.C. § 1251 et seq.	N/A	October 18, 1972	92nd United States Congress	N/A		Public Law 92-500
RD, RR	*Resource Conservation and Recovery Act (Solid Waste Disposal Act)*	42 U.S.C. § 6901 et seq.	N/A	October 21, 1976	94th United States Congress	N/A		Public Law 94-580
RD, RR	*Nuclear Waste Policy Act as Amended*	42 U.S.C. 10101	**N/A**	March 2004	U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM)	N/A		
RD, RR	*Comprehensive Environmental Response, Compensation, and Liability Act*	42 U.S.C. § 9601 et seq.	N/A	December 11, 1980	96th United States Congress	N/A		Public Law 96-510

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RD, RR	*Water Quality Act of 1987*	N/A	N/A	February 4, 1987	100th United States Congress	N/A		Public Law 100-4
RG	*Flood Protection for Nuclear Power Plants*	RG 1.102	1	September 1976	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2014
RG	*Tornado Design Classification*	RG 1.117	1	April 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2012
RG	*Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components*	RG 1.122	1	February 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2015
RG	*Spent Fuel Storage Facility Design Basis*	RG 1.13	2	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2010
RG	*Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23)*	RG 1.23	1	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)*	RG 1.25	0	March 1972	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2010
RG	*Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*	RG 1.26	4	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2013
RG	*Quality Assurance Program Criteria (Design and Construction)*	RG 1.28	4	June 2010	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Seismic Design Classification*	RG 1.29	4	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2013
RG	*Quality Assurance Program Requirements (Operation)*	RG 1.33	3	June 2013	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Design Basis Floods for Nuclear Power Plants*	RG 1.59	2	August 1977	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2014
RG	*Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants*	RG 1.76	1	March 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: August 2011
RG	*Combining Modal Responses and Spatial Components in Seismic Response Analysis*	RG 1.92	3	October 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*	RG 3.48	1	August 1989	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: June 2014
RG	*Standard Format and Content for a Specific License Application for An Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility*	RG 3.50	2	September 2014	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation*	RG 3.53	N/A	July 1982	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: April 2014

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RG	*Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation*	RG 3.54	1	January 1999	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: March 2011
RG	*Design of an Independent Spent Fuel Storage Installation (Dry Storage)*	RG 3.60	N/A	March 1987	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: May 2014
RG	*Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask*	RG 3.61	N/A	February 1989	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: June 2014
RG	*Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks*	RG 3.62	N/A	February 1989	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: June 2014
RG	*Standard Format and Content of Decommissioning Plans for Materials Licensees*	RG 3.65	1	May 2008	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: November 2013
RG	*Standard Format and Content of Financial Assurance Mechanisms*	RG 3.66	1	May 2008	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: November 2013
RG	*Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities*	RG 3.67	1	April 2011	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Nuclear Criticality Safety Standards for Fuels and Material Facilities*	RG 3.71	2	December 2010	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations*	RG 3.73	N/A	October 2003	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: May 2011
RG	*Corrective Action Programs for Fuel Cycle Facilities*	RG 3.75	N/A	July 2014	U.S. Nuclear Regulatory Commission, Division 3, Fuels and Materials Facilities	N/A		Last NRC Staff Review: N/A
RG	*Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment*	RG 4.15	2	July 2007	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: April 2012
RG	*Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities*	RG 4.16	2	December 2010	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors*	RG 4.20	1	April 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A

Reference Category	Document Title	Document Number	Revision	Issued or Published Date	Issuer/Regulatory Entity	Listed Author(s)	NFST Reference	Coments/Notes
RG	*Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning*	RG 4.21	N/A	June 2008	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: July 2013
RG	*Decommissioning Planning During Operations*	RG 4.22	N/A	December 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Training, Equipping, and Qualifying of Guards and Watchmen*	RG 5.20	N/A	January 1974	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2011
RG	*Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities (for Comment)*	RG 5.55	N/A	March 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2011
RG	*Standard Format and Content of Safeguards Contingency Plans for Transportation (for Comment)*	RG 5.56	N/A	March 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: September 2011
RG	*Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas*	RG 5.7	1	May 1980	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: April 2015
RG	*Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material*	RG 7.10	2	March 2005	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011
RG	*Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)*	RG 7.11	N/A	June 1991	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011. Basis for NRC RG 7.11 is provided by NUREG-1815.
RG	*Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)*	RG 7.12	N/A	June 1991	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011
RG	*Transportation Security Plans for Classified Matter Shipments*	RG 7.13	?	Unknown	U.S. Nuclear Regulatory Commission	N/A		Document marked “Official Use Only – Security-Related Information.”
RG	*Leakage Tests on Packages for Shipment of Radioactive Materials*	RG 7.4	1	March 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*	RG 7.6	1	March 1978	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: June 2014
RG	*Administrative Guide for Verifying Compliance with Packaging Requirements for Shipments of Radioactive Materials (for Comment)*	RG 7.7	1	March 2012	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*	RG 7.8	1	March 1989	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: June 2011
RG	*Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*	RG 7.9	2	March 2005	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: March 2011
RG	*Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable*	RG 8.10	1-R	May 1997	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2014
RG	*Air Sampling in the Workplace*	RG 8.25	1	June 1992	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2011

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RG	*Applications of Bioassay for Fission and Activation Products*	RG 8.26	N/A	September 1980	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: December 2011
RG	*Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants*	RG 8.27	N/A	March 1981	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: October 2011
RG	*Instruction Concerning Risks from Occupational Radiation Exposure*	RG 8.29	1	February 1996	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: December 2011
RG	*Monitoring Criteria and Methods To Calculate Occupational Radiation Doses*	RG 8.34	N/A	July 1992	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*ALARA Levels for Effluents from Materials Facilities*	RG 8.37	N/A	July 1993	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: February 2012
RG	*Control of Access to High and Very High Radiation Areas of Nuclear Plants*	RG 8.38	1	May 2006	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Methods for Measuring Effective Dose Equivalent from External Exposure*	RG 8.40	N/A	July 2010	U.S. Nuclear Regulatory Commission	N/A		Last NRC Staff Review: N/A
RG	*Fuel Retrievability*	SFST-ISG-2	1	February 22, 2010	U.S. Nuclear Regulatory Commission, Division of Spent Fuel Storage and Transportation (SFST)	N/A		Interim Staff Guidance
RR	*Standards for Protection against Radiation*	10 CFR 20	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		
RR	*Domestic Licensing of Production and Utilization Facilities*	10 CFR 50	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		Potentially applicable, e.g., in any limited cases where 10 CFR 72 does not govern.
RR	*Packaging and Transportation of Radioactive Material*	10 CFR 71	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A	R 2.1.5	
RR	*Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*	10 CFR 72	N/A	January 1, 2006	U.S. Nuclear Regulatory Commission	N/A	R 2.0.4	
RR	*Physical Protection of Plants and Materials*	10 CFR 73	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		
RR	*Material Control and Accounting of Special Nuclear Material*	10 CFR 74	N/A	January 1, 2006	U.S. Nuclear Regulatory Commission	N/A		
RR	*Safeguards on Nuclear Materials-Implementation of U.S./IAEA Agreement*	10 CFR 75	N/A	N/A	U.S. Nuclear Regulatory Commission	N/A		
RR	*Worker Safety and Health Program*	10 CFR 851	N/A	N/A	U.S. Department of Energy	N/A		
RR	*Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste*	10 CFR 961	N/A	N/A	U.S. Department of Energy	N/A	R 0.3	
RR	*Occupational Safety and Health Standards*	29 CFR 1910	N/A	N/A	U.S. Department of Labor, Occupational Safety and Health Administration	N/A		
RR	*Identification and Listing of Hazardous Waste*	40 CFR 261	N/A	July 1, 2012	U.S. Environmental Protection Agency	N/A		

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RR	*National Primary and Secondary Ambient Air Quality Standards*	40 CFR 50	N/A	N/A	U.S. Environmental Protection Agency	N/A		
RR	*National Emission Standards for Hazardous Air Pollutants*	40 CFR 61	N/A	July 1, 2011	U.S. Environmental Protection Agency	N/A		
RR	*Designation of Areas for Air Quality Planning Purposes*	40 CFR 81	N/A		U.S. Environmental Protection Agency	N/A		
RR	*Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, Training Requirements, and Security Plans* , Subpart I, *Safety and Security Plans*	49 CFR 172, Subpart I	N/A	N/A	U.S. Department of Transportation, Pipeline and Hazardous Materials Safety Administration	N/A		
RR	*Transportation: Shippers—General Requirments for Shipments and Packings*	49 CFR 173	N/A	N/A	U.S. Department of Transportation, U.S. Department of Homeland Security	N/A	R 2.1.5	
RR, RD	*Regulations for the Safe Transport of Radioactive Material, 2012 Edition (Specific Safety Requirements)*	No. SSR-6	N/A	October 2012	International Atomic Energy Agency	N/A		IAEA Safety Standard, STI/PUB/1570, ISBN:978-92-0-133310-0

DR = Design Requirement (Design Standard?)
NR = NRC NUREG
PG = Program Guidance
RD = Resource Document
RG = Regulatory Guidance
RR = Regulatory Requirements