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Office of
Fissile Materials Disposition

United States Department of Energy

Supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement

April 1999

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Cover Sheet

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Title: *Supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement* (*Supplement*) (DOE/EIS-0283-DS)

Locations of Candidate Sites: Idaho, North Carolina, South Carolina, Texas, Virginia, and Washington

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Abstract: On May 22, 1997, DOE published a Notice of Intent in the Federal Register (62 Federal Register 28009) announcing its decision to prepare an environmental impact statement (EIS) that would tier from the analysis and decisions reached in connection with the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS (Storage and Disposition PEIS)*. The *Surplus Plutonium Disposition Draft Environmental Impact Statement* (SPD Draft EIS) (DOE/EIS-0283-D) was prepared in accordance with NEPA and issued in July 1998. It identified the potential environmental impacts of reasonable alternatives for the proposed siting, construction, and operation of three facilities for plutonium disposition. These three facilities would accomplish pit disassembly and conversion, immobilization, and MOX fuel fabrication. For the alternatives that included MOX fuel fabrication, the draft also described the potential environmental impacts of using from three to eight commercial nuclear reactors to irradiate MOX fuel. The potential impacts were based on a generic reactor analysis that used actual reactor data and a range of potential site conditions. In May 1998, DOE initiated a procurement process to obtain MOX fuel fabrication and reactor irradiation services. The request for proposals defined limited activities that may be performed prior to issuance of the SPD EIS Record of Decision (ROD) including non-site-specific work associated with the development of the initial design for the MOX fuel fabrication facility, and plans (paper studies) for outreach, long lead-time procurements, regulatory management, facility quality assurance, safeguards, security, fuel qualification, and deactivation. No construction on the proposed MOX facility would begin before an SPD EIS ROD is issued. In March 1999, DOE awarded a contract to Duke Engineering & Services; COGEMA, Inc.; and Stone & Webster (known as DCS) to provide the requested services. The procurement process included the environmental review specified in DOE's NEPA regulations in 10 CFR 1021.216. The six reactors selected are Catawba Nuclear Station Units 1 and 2 in South Carolina, McGuire Nuclear Station Units 1 and 2 in North Carolina, and North Anna Power Station Units 1 and 2 in Virginia. The *Supplement* describes the potential environmental impacts of using MOX fuel in these six specific reactors named in the DCS proposal as well as other program changes made since the SPD Draft EIS was published.

Public Involvement: Comments on the *Supplement* may be submitted by mail to DOE, Office of Fissile Materials Disposition, c/o Supplement to the SPD EIS, P.O. Box 23786, Washington, DC 20026–3786; by email at http://www.doe-md.com (Public Involvement, Comment Table, Send Us Email); by calling DOE at 1–800–820–5156; or by sending a facsimile (fax) message to DOE at 1–800–820–5156. To ensure consideration in the SPD Final EIS, these comments should be submitted within 45 days after the U.S. Environmental Protection Agency Notice of Availability is published in the Federal Register. Comments received after the end of the comment period will be considered to the extent possible. A public hearing will be held on the date and time specified in a DOE Federal Register notice and announced in local media. Comments on the SPD Draft EIS can also be submitted at this hearing. Preregistration for the public hearing is available by calling 1–800–820–5134 or by fax at 1–800–820–5156. Additional information can be obtained by calling the contacts listed above, or by visiting the Office of Fissile Materials Disposition Web site at http://www.doe-md.com.

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List of Acronyms

ALARA	as low as is reasonably achievable
APSF	Actinide Packaging and Storage Facility
AQCR	Air Quality Control Region
DCS	Duke Engineering & Services; COGEMA, Inc.; and Stone & Webster
DOE	U.S. Department of Energy
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
FMEF	Fuels and Materials Examination Facility
FR	Federal Register
HEPA	high-efficiency particulate air (filter)
INEEL	Idaho National Engineering and Environmental Laboratory
IPE	Individual Plant Examination
ISLOCA	interfacing systems loss-of-coolant accident
LANL	Los Alamos National Laboratory
LCF	latent cancer fatality
LEU	low-enriched uranium
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LOCA	loss-of-coolant accident
MACCS2	Melcor Accident Consequence Code System (computer code)
MEI	maximally exposed individual
MOX	mixed oxide
NAAQS	National Ambient Air Quality Standards
NEPA	National Environmental Policy Act of 1969
NOI	Notice of Intent
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PEIS	programmatic environmental impact statement
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RFETS	Rocky Flats Environmental Technology Site
ROD	Record of Decision
SALP	systematic assessment of licensee performance
SI	sealed insert
SPD EIS	*Surplus Plutonium Disposition Environmental Impact Statement*
SRS	Savannah River Site
SST	safe, secure trailer
UFSAR	Updated Final Safety Analysis Report

I. Introduction

The *Surplus Plutonium Disposition Draft Environmental Impact Statement* (SPD Draft EIS) (DOE/EIS-0283-D) was prepared in accordance with the National Environmental Policy Act (NEPA) and issued in July 1998. It identified the potential environmental impacts of reasonable alternatives for the proposed siting, construction, and operation of three facilities for plutonium disposition. These three facilities would accomplish pit disassembly and conversion, plutonium conversion and immobilization, and mixed oxide (MOX) fuel fabrication. For the alternatives that included MOX fuel fabrication, the draft also described the potential environmental impacts of using from three to eight commercial nuclear reactors to irradiate MOX fuel. The potential impacts were based on a generic reactor analysis that used actual reactor data and a range of potential site conditions. In May 1998, DOE initiated a procurement process to obtain MOX fuel fabrication and reactor irradiation services. The request for proposals defined limited activities that may be performed prior to issuance of the SPD EIS Record of Decision (ROD) including non-site-specific work associated with the development of the initial design for the MOX fuel fabrication facility, and plans (paper studies) for outreach, long lead-time procurements, regulatory management, facility quality assurance, safeguards, security, fuel qualification, and deactivation. No construction on the proposed MOX fuel fabrication facility would begin before an SPD EIS ROD is issued. In March 1999, DOE awarded a contract to Duke Engineering & Services; COGEMA, Inc.; and Stone & Webster (known as DCS) to provide the requested services. The procurement process included the environmental review specified in the U.S. Department of Energy's (DOE's) NEPA regulations in 10 CFR 1021.216. This *Supplement* describes the potential environmental impacts of using MOX fuel in the six specific reactors at three sites named in the DCS proposal, as well as other program changes made since the SPD Draft EIS was published.

This *Supplement* consists of six sections that (1) explain the purpose and context of this *Supplement*, (2) add new sections to the SPD Draft EIS, or (3) revise and replace portions of the SPD Draft EIS. The first part is this introduction. The second part includes background information extracted from the SPD Draft EIS that provides an overview of DOE's ongoing NEPA review process for this program. The third part discusses changes that have been made to the program since issuance of the SPD Draft EIS, as well as the environmental implications of these changes The fourth part describes the affected environment for the commercial reactor sites that are proposed to irradiate MOX fuel. The fifth part includes impacts analyzed for these reactor sites and replaces generic reactor information in the SPD Draft EIS.

The last part of this *Supplement* consists of three appendixes that either amend an existing appendix or add a new appendix to the SPD Draft EIS. Appendix A, *Federal Register Notices*, contains the Notice of Intent to publish this *Supplement*, which appeared in the Federal Register on April 6, 1999. Appendix K, *Facility Accidents*, and Appendix M, *Analysis of Environmental Justice*, include reactor-specific information that was not included in the SPD Draft EIS. This information, which is represented as stand-alone appendixes in this *Supplement*, will be appended to Appendixes A, K, and M in the SPD Final EIS. Appendix P, *Environmental Synopsis of Information Provided in Response to the Request for Proposals for MOX Fuel Fabrication and Reactor Irradiation Services*, is a new appendix that will be included in the SPD Final EIS.

During the public comment period on the SPD Draft EIS, DOE held five public meetings to solicit comments on the document. Comments were also received via fax, mail, phone answering machine, mail, and the MD Web site. DOE will present its responses to the comments as part of the SPD Final EIS. Comments presented both supporting and opposing views on the range of siting and technology alternatives being considered by DOE. Where specific, substantive technical issues were raised, DOE will make appropriate changes to the impact analysis in the SPD Final EIS. DOE is issuing this *Supplement* to provide an opportunity for public comment on sections that are being added to the SPD Draft EIS and sections that are being revised and replaced. DOE will respond to comments previously provided on the SPD Draft EIS, as well as comments provided on this *Supplement*, in the SPD Final EIS anticipated later this year.

II. Background Information Extracted From the Surplus Plutonium Disposition Draft Environmental Impact Statement

In December 1996, DOE published the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (*Storage and Disposition PEIS*) (DOE 1996). This programmatic environmental impact statement (PEIS) analyzes the potential environmental consequences of alternative strategies for the long-term storage of weapons-usable plutonium and highly enriched uranium and the disposition of weapons-usable plutonium that has been or may be declared surplus to national security needs. The Record of Decision (ROD) for the *Storage and Disposition PEIS,* issued on January 14, 1997 (DOE 1997a), outlines DOE's decision to pursue a hybrid approach to plutonium disposition that would make surplus weapons-usable plutonium inaccessible and unattractive for weapons use. DOE's disposition strategy, consistent with the preferred alternative analyzed in the *Storage and Disposition PEIS,* allows for both the immobilization of some (and potentially all) of the surplus plutonium and use of some of the surplus plutonium as mixed oxide (MOX) fuel in existing domestic, commercial reactors. The disposition of surplus plutonium would also involve disposal of both the immobilized plutonium and the MOX fuel (as spent fuel) in a potential geologic repository.

On May 22, 1997, DOE published a Notice of Intent (NOI) in the Federal Register (FR) (DOE 1997b) announcing its decision to prepare an environmental impact statement (EIS) that would tier from the analysis and decisions reached in connection with the *Storage and Disposition PEIS*. This EIS, the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (SPD Draft EIS) (DOE 1998), addresses the extent to which each of the two plutonium disposition approaches (immobilization and MOX) would be implemented and analyzes candidate sites for plutonium disposition facilities, as well as alternative technologies for immobilization.

The SPD EIS analyzes a nominal 50 t (55 tons) of surplus weapons-usable[1] plutonium, which is primarily in the form of pits (a nuclear weapons component), metal, and oxides. In addition to 38.2 t (42 tons) of weapons-grade plutonium[2] already declared by the President as surplus to national security needs, the 50 t (55 tons) of material analyzed includes weapons-grade plutonium that may be declared surplus in the future, as well as weapons-usable, reactor-grade plutonium that is surplus to the programmatic and national defense needs of DOE.

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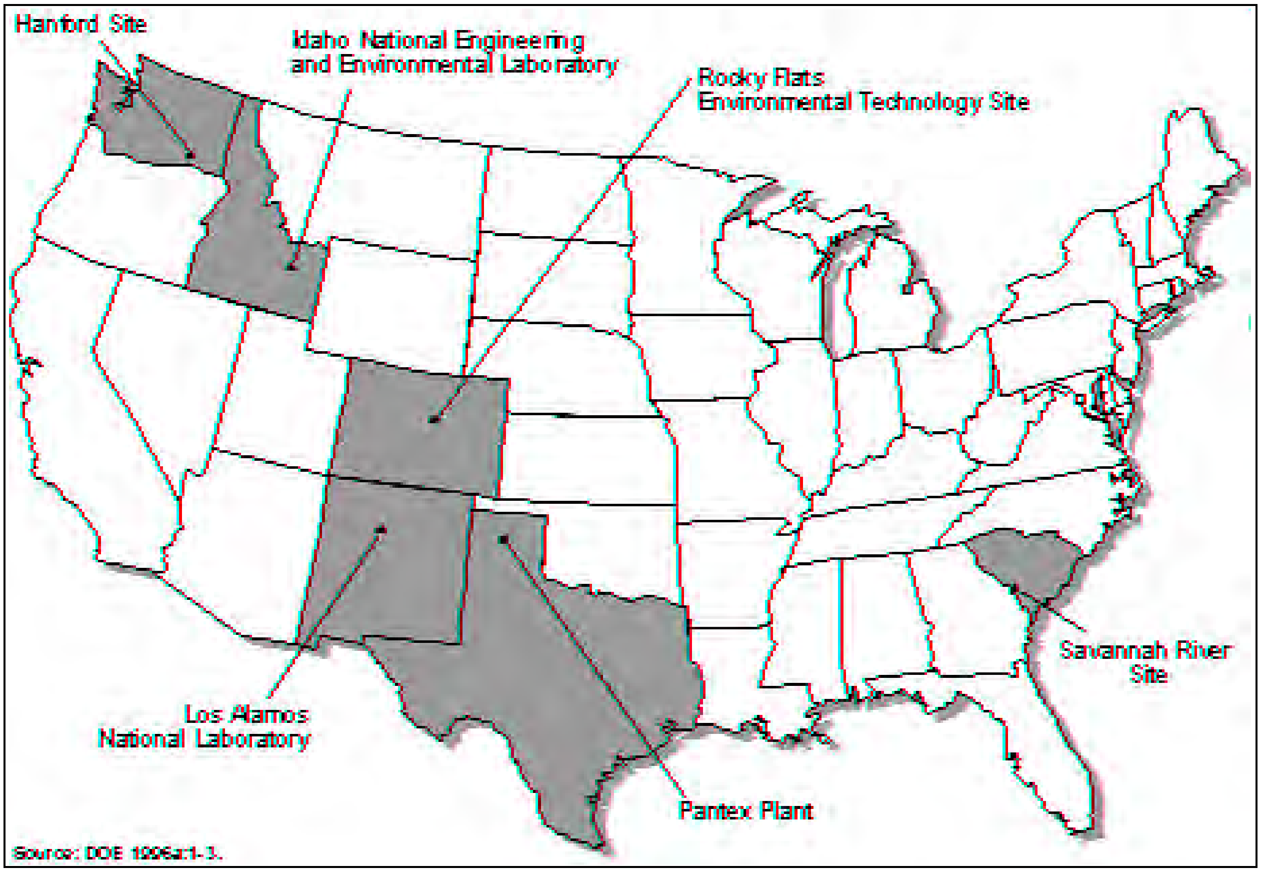
[1] Weapons-usable material includes plutonium or highly enriched uranium in forms (e.g., metals, oxides) that can be readily converted for use in nuclear weapons. Weapons-grade, fuel-grade, and power-reactor-grade plutonium are all weapons usable.

[2] Weapons-grade material includes plutonium or highly enriched uranium, in metallic form, that was manufactured for weapons application. Weapons-grade plutonium contains less than 7 percent plutonium 240.

As depicted in Figure II–1, surplus plutonium is stored at six locations within the DOE complex: the Hanford Site (Hanford) near Richland, Washington; Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho; Los Alamos National Laboratory (LANL) near Los Alamos, New Mexico; the Pantex Plant (Pantex) near Amarillo, Texas; the Rocky Flats Environmental Technology Site (RFETS) near Golden, Colorado; and the Savannah River Site (SRS) near Aiken, South Carolina.



Source: DOE 1996a:1-3.

The *Storage and Disposition PEIS* ROD determined that DOE would immobilize at least 8 t (9 tons) of the current surplus plutonium due to the technology, complexity, timing, and cost that would be involved in purifying the material to make it suitable for MOX fuel fabrication. Since issuance of the ROD, further consideration has indicated that 17 t (19 tons) of the 50 t (55 tons) of surplus plutonium is not suitable for use in MOX fuel and should be immobilized. Therefore, fabricating all 50 t (55 tons) of surplus plutonium into MOX fuel is not a reasonable alternative and is not analyzed. As a bounding case, the SPD EIS does, however, analyze the immobilization of all the surplus plutonium. Moreover, given the variability in purity of the surplus plutonium to be dispositioned, some of the plutonium currently considered for MOX fabrication may also need to be immobilized.

The purpose of and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by conducting disposition of surplus plutonium in the United States in an environmentally safe and timely manner. Comprehensive disposition actions are needed to ensure that surplus plutonium is converted to proliferation-resistant forms. In September 1993, President Clinton issued the *Nonproliferation and Export Control Policy* (White House 1993) in response to the growing threat of nuclear proliferation. Further, in January 1994, President Clinton and Russia's President Yeltsin issued a *Joint Statement by the President of the Russian Federation and the President of the United States of America on Non-proliferation of Weapons of*

Mass Destruction and the Means of Their Delivery (White House 1994). In accordance with these policies, the focus of the U.S. nonproliferation efforts includes ensuring the safe, secure, long-term storage, and disposition of surplus weapons-usable fissile plutonium. The disposition activities proposed in the SPD EIS will enhance U.S. credibility and flexibility in negotiations on bilateral and multilateral reductions of surplus weapons-usable fissile materials inventories. Actions undertaken by the United States would generally be coordinated with efforts to address surplus plutonium stocks in the Russian Federation. For example, the construction of new facilities for disposition of U.S. plutonium will likely depend on progress in Russia. However, the United States will retain the option to begin certain disposition activities, when appropriate, in order to encourage the Russians and set an international example.

The SPD Draft EIS addresses both the immobilization and MOX approaches to surplus plutonium disposition, which include siting, construction, operation, and ultimate decontamination and decommissioning of three types of facilities at one or two of four DOE candidate sites:

- A facility for disassembling pits (a weapons component) and converting the recovered plutonium, as well as plutonium metal from other sources, into plutonium dioxide suitable for disposition. This facility, the pit disassembly and conversion facility, is referred to in this document as the *pit conversion facility*. Candidate sites for this facility are Hanford, INEEL, Pantex, and SRS.[3]

- A facility for immobilizing surplus plutonium for eventual disposal in a potential geologic repository pursuant to the Nuclear Waste Policy Act. This facility, referred to as the *immobilization facility*, would include a collocated capability for converting nonpit plutonium materials into plutonium dioxide suitable for immobilization. The immobilization facility would be located at either Hanford or SRS. DOE identified SRS as the preferred site for an immobilization facility in its *Notice of Intent* to prepare the SPD EIS. Technologies for immobilization are also discussed in the SPD EIS.

- A facility for fabricating plutonium dioxide into MOX fuel, the MOX fuel fabrication facility, is referred to as the *MOX facility*. Candidate sites for this facility are Hanford, INEEL, Pantex, and SRS. SRS has been identified as the preferred site for this facility. Also included in the SPD Draft EIS is a separate analysis of MOX lead assembly activities at five DOE candidate sites: Argonne National Laboratory–West (ANL–W) at INEEL; Hanford; Lawrence Livermore National Laboratory (LLNL) in Livermore, California; LANL; and SRS. DOE would fabricate a limited number of MOX fuel assemblies, referred to as lead assemblies, for testing in reactors before commencing fuel irradiation under the proposed MOX fuel program.

The SPD Draft EIS also analyzes a No Action Alternative, as required by the National Environmental Policy Act. In the No Action Alternative, surplus weapons-usable plutonium in storage at various DOE sites would remain at those locations. The vast majority of pits and plutonium metal would continue to be stored at Pantex, and the remaining plutonium in various forms would continue to be stored at Hanford, INEEL, LANL, RFETS, and SRS.

REFERENCES

DOE (U.S. Department of Energy), 1996, *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement*, DOE/EIS-0229, Office of Fissile Materials Disposition, Washington, DC, December.

[3] As announced in a Secretarial Press Release on December 22, 1998 (R-98-200), SRS is the preferred site for the pit disassembly and conversion facility.

DOE (U.S. Department of Energy), 1997a, *Record of Decision for the Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement*, 62 FR 3014, Office of the Federal Register, Washington, DC, January.

DOE (U.S. Department of Energy), 1997b, *Surplus Plutonium Disposition Environmental Impact Statement*, Notice of Intent, 62 FR 28009, Office of the Federal Register, Washington, DC, May 22.

DOE (U.S. Department of Energy), 1998, *Surplus Plutonium Disposition Draft Environmental Impact Statement*, DOE/EIS-0283D, Office of Fissile Materials Disposition, Washington, DC, July.

White House, 1993, *Nonproliferation and Export Control Policy*, Office of the Press Secretary, Washington, DC, September 27.

White House, 1994, *Joint Statement by the President of the Russian Federation and the President of the United States of America on Nonproliferation of Weapons of Mass Destruction and the Means of Their Delivery*, Office of the Press Secretary, Washington, DC, January 14.

III. Summary of Changes Made to the Surplus Plutonium Disposition Program and New Information

Since the issuance of the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (SPD Draft EIS), DOE has made some minor technical changes to the program and has revised information or added new information in response to stakeholder comments and to reflect DOE's current planning. These changes and their effect on the environmental impacts of the proposed action are described below.

- **Further definition of the preferred alternative.** DOE has identified the Savannah River Site (SRS) as the preferred alternative for pit disassembly and conversion, Los Alamos National Laboratory (LANL) for lead assembly fabrication, and Oak Ridge National Laboratory for postirradiation examination.

- **Changes to the immobilization facility.** Since the issuance of the SPD Draft EIS, DOE has developed a more detailed conceptual design for the immobilization facility. Some of the design changes include lengthening the process gloveboxes by about 35 percent; doubling the material conveyor length; changing to a vertical ceramification stack that affected the configuration of the second level of the facility; increasing the heating, ventilating, and air conditioning systems and electrical support to correspond with the increased process space; enlarging the space required for maintenance activities; and increasing the size of the canister-loading area. To accommodate these design modifications, the proposed immobilization facility has approximately doubled in size, in terms of floor space; however, the change in required land area varies among the alternatives, depending on the configuration of the facilities. Similarly, the environmental impacts attributable to the larger facility size vary by specific resource area and by alternative. No changes have been made to the basic processes proposed in the SPD Draft EIS for immobilization, to the amount of material being considered for immobilization, or to the rate of throughput.

  For the alternatives that included immobilization at Hanford, the size of the immobilization facility varies depending on which of the other disposition facilities are also located at Hanford. The size ranges from 20,000 m^2 (215,000 ft^2) to 21,600 m^2 (233,000 ft^2); in the SPD Draft EIS, the facility varied in size from 6,698 m^2 (72,100 ft^2) to 13,694 m^2 (147,400 ft^2). The estimated land area required for construction and operation of the immobilization facility increased from 2.1 ha (5.2 acres) to as much as 8.3 ha (20 acres) for Alternative 4B where the immobilization facility is collocated with the MOX facility in the existing Fuels and Materials Examination Facility (FMEF); in order to accommodate the larger immobilization facility, a canister-loading facility would need to be constructed as a separate annex to FMEF. However, all new construction is in previously disturbed areas adjacent to existing facilities, so even with the larger facility, environmental impacts from construction are expected to be similar to those described in the SPD Draft EIS. Impacts from operation would be higher because of the approximately 24 percent increase in the number of workers and the correspondingly greater electricity, fuel, and water use requirements associated with the larger facility.

  At SRS, the eight alternatives that included using portions of Building 221–F for immobilization were eliminated (Alternatives 3B, 5B, 6C, 6D, 7B, 9B, 12B, and 12D), based on the increased space requirements. These alternatives are no longer reasonable because the amount of new construction required for the proposed immobilization facility is now nearly the same whether the facility is located entirely in a new building or uses a portion of Building 221–F. There is no longer any advantage associated with the use of Building 221–F at SRS in terms of reducing the local environmental impacts, reducing costs, or shortening the construction schedule for this facility. Therefore, DOE has determined that there is no longer a reasonable basis for carrying forward both the Building 221–F and the new facility options of the immobilization approach. Deletion of the Building 221–F option does not eliminate SRS from any of the immobilization alternatives under consideration. For all alternatives that originally considered both Building 221–F and a new facility at SRS as possible sites for the immobilization facility, DOE is still evaluating the new facility alternative.

For the remaining SRS alternatives, the size of the immobilization facility has increased from 13,000 m^2 (140,000 ft^2) to 25,000 m^2 (269,000 ft^2); however, the land area required for the immobilization facility is essentially the same as the amount analyzed in the SPD Draft EIS. Impacts from operation would be higher because of the approximately 33 percent increase in the number of workers and the correspondingly greater electricity, fuel, and water use requirements associated with the larger facility.

- **Changes resulting from the MOX procurement process.** Information provided as part of the MOX procurement process relating to the MOX facility, including the addition of a plutonium-polishing module to the front end of the MOX facility, was analyzed by DOE in an environmental critique prepared pursuant to the National Environmental Policy Act (NEPA) regulations in 10 CFR 1021.216 and summarized in an environmental synopsis. The synopsis is included in this *Supplement* (and will be included in the SPD Final EIS) as Appendix P. Information related to the affected environment for the domestic commercial reactors that would irradiate the MOX fuel is included in Section IV of this *Supplement* and will be added to the SPD Final EIS as Section 3.7. Environmental impacts analyzed for the actual reactor sites are presented in Section V of this *Supplement* and will be included as Section 4.28 of the SPD Final EIS.

  Appendix N, *Plutonium Polishing*, will be deleted from the SPD Final EIS because that information will be incorporated in Chapter 4 of the SPD Final EIS. Because the selected contractor, DCS, prefers to include the polishing step at the MOX facility, plutonium polishing is no longer considered as a contingency for the pit conversion facility.

  The impacts associated with the MOX facility (described in Appendix P of this *Supplement*) are essentially the same as those presented in Chapter 4 and Appendix N of the SPD Draft EIS. The size of the MOX facility has increased by approximately 4,200 m^2 (45,000 ft^2). The analysis in the SPD Draft EIS considered 11,000 m^2 (119,000 ft^2) for the MOX facility and 2,800 m^2 (30,000 ft^2) for the plutonium-polishing module for a total of about 13,800 m^2 (149,000 ft^2). In this *Supplement* and in the SPD Final EIS, the MOX facility is about 20,000 m^2 (215,000 ft^2), which includes additional space proposed by DCS as well as space for the plutonium-polishing capability and about 2,000 m^2 (21,000 ft^2) of administrative space that was located in separate support facilities in the SPD Draft EIS. The amount of land required for construction has not changed, and the amount required during operation has only increased slightly (approximately 5 percent). The number of workers and the projected worker doses, as proposed by DCS, are less than those estimated in the SPD Draft EIS and are also presented in Appendix P of this *Supplement*. No changes have been made in the amount of material proposed to be made into MOX fuel or in the overall process to be used to fabricate the fuel.

- **DOE's decision to delay the construction of the Actinide Packaging and Storage Facility (APSF) at SRS.** In the SPD Draft EIS, alternatives that considered locating the surplus plutonium disposition facilities in new construction at SRS (Alternatives 3A and 3B, 5A and 5B, 6A and 6B, 7A, 9A, and 12A and 12C) took into account the use of the adjacent proposed APSF as a receiving facility for safe, secure trailer shipments; as a storage vault for storing plutonium oxide and metal; and for the pit and immobilization facilities, as a nondestructive assay facility. Therefore, the SPD Draft EIS analyzed somewhat smaller disposition facilities for these alternatives. Because the schedule for APSF is uncertain at this time, the disposition facilities analyzed in the SPD Final EIS will be modified to disregard any benefit to the proposed facilities as a result of APSF being present at SRS. These facility changes are described in the following paragraphs and are expected to result in minor changes, if any, to the environmental impacts reported in the SPD Draft EIS.

The SPD Final EIS will present the environmental impacts that would be associated with the construction and operation of surplus plutonium disposition facilities at SRS that are stand alone and include no reliance on storage space or other functions at APSF. Throughout the SPD Final EIS, references to APSF will be qualified by the phrase "if built," and no credit will be taken in the environmental analyses for the presence of APSF. If DOE decides to collocate all three disposition facilities at SRS as indicated in the preferred alternative (see Section 1.6 of the SPD Draft EIS), the final design of these facilities would coordinate potential common functions among the facilities to the extent practical as a means to reduce space requirements and the associated environmental impacts.

The pit conversion facility that will be analyzed at SRS in the SPD Final EIS is identical to that proposed in the Pantex alternatives, where it has always been considered as a stand-alone facility. The MOX facility proposed for SRS has also been replaced with the larger stand-alone facility that is the same as the facility proposed at the other candidate sites. No longer relying on APSF results in minor adjustments in facility construction requirements and associated impacts that will be reflected in minor changes to Chapter 4 of the SPD Final EIS.

As discussed earlier, the proposed immobilization facility at SRS has been increased in size based on further analysis of the functional requirements for immobilization. Some space would be available in the current immobilization design to partially offset the use of APSF for functions such as storage or accountability measurements. However, without APSF, the construction of truck bays and other minor modifications (up to approximately 980 m^2 [10,500 ft^2]) would be necessary. These changes are not expected to substantially affect the environmental impacts associated with the larger immobilization facility that will be analyzed in the SPD Final EIS.

- **Pit repackaging requirements.** Based on estimates presented in the *Final EIS for the Continued Operation of Pantex and Associated Storage of Nuclear Weapons Components* (DOE 1996), 50 workers would be needed to repackage 12,000 pits from their current storage containers into containers that could also be used for shipping.[1] Work is currently underway to repackage pits from the AL–R8 container into the AL–R8 sealed insert (SI) container as discussed in the *Supplement Analysis for the Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapons Components—AL–R8 Sealed Insert Container* (DOE 1998). This effort would be completed over 10 years and the estimated annual dose received from repackaging activities would be about 73 mrem per worker. By locating the pit conversion facility at Pantex, it is expected that the additional dose associated with repackaging the surplus pits into shipping containers could be avoided. This would effectively reduce the total expected dose for these activities by 50 percent. If the pit conversion facility were sited at Pantex, the pits would be slowly moved from storage locations in storage containers on specially designed vehicles to the pit conversion facility instead of having to be put into offsite shipping containers. Over the 10-year operating life of the pit conversion facility, this would reduce the total estimated dose to involved Pantex transportation and staging workers

[1] In the analysis presented in the *Pantex EIS* (DOE 1996), pits are assumed to be repackaged in AT–400A containers. The amount of effort involved in repackaging a pit in an AT–400A container is more intense than the effort needed to repackage a pit in a FL-type container or equivalent; therefore, the doses would be expected to be higher. Since the *Pantex EIS* was completed, it has been decided that surplus pits would not be repackaged in AT–400A containers. As a result, the dose estimates associated with repackaging pits as presented in the *Pantex EIS* are conservatively high for the SPD EIS. No effort has been made to reestimate the doses associated with repackaging pits. The doses presented in the SPD EIS are based on using the AT–400A container and, therefore, represent upper bounds on the expected dose to involved workers.

from 74 person-rem to 37 person-rem. Under either scenario, the estimated number of excess cancer fatalities associated with repackaging activities would be 0.03 or less.

- **Changes to cumulative impacts.** New or revised NEPA documents, such as the *Savannah River Site Spent Nuclear Fuel Management Draft EIS* and the *Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Sites*, will result in changes to the discussion of cumulative impacts in the SPD Final EIS. In addition, cumulative impacts information will be added for Lawrence Livermore National Laboratory and LANL, two candidates sites for lead assembly fabrication. Because DOE has decided to use civilian light water reactors for the production of tritium rather than constructing a new linear accelerator at SRS, the impacts of construction and operation of that accelerator will no longer be included in the cumulative impacts section of the SPD Final EIS, thus reducing the overall cumulative impacts at that site.

REFERENCES

DOE (U.S. Department of Energy), 1996, *Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components*, DOE/EIS-0225, Albuquerque Operations Office, Albuquerque, NM, November.

DOE (U.S. Department of Energy), 1998, *Supplement Analysis for: Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components—AL–R8 Sealed Insert Container*, Albuquerque Operations Office, Amarillo Area Office, Amarillo, TX, August.

IV. Proposed MOX Fuel Irradiation Program

Under the mixed oxide (MOX) fuel approach being considered in the Surplus Plutonium Disposition Environmental Impact Statement *(SPD EIS), three reactor sites with a total of six reactors are now being considered for MOX fuel irradiation. The proposed action under the MOX approach is to use these reactors to irradiate MOX fuel. The cores of these reactors will be partially fueled (i.e., up to 40 percent) with MOX fuel and the MOX fuel will run through a normal fuel cycle. This section provides a description of the affected environment around the three proposed reactor sites. It will be included in the SPD Final EIS as Section 3.7.*

3.7 REACTOR SITES FOR MOX FUEL IRRADIATION

3.7.1 Catawba Units 1 and 2 Site Overview

The Catawba nuclear power plant occupies 158 ha (391 acres) in York County, South Carolina, 9.3 km (5.8 mi) north-northwest of Rock Hill, South Carolina, and 16.9 km (10.5 mi) west-southwest of Charlotte, North Carolina (see Figure 3.7–1). The site is on a peninsula bounded by Beaver Dam Creek to the north, Big Allison Creek to the south, Lake Wylie to the east, and private property to the west (Duke Power 1997:2-3). Lake Wylie has a surfaçe area of 5,040 ha (12,455 acres), a shoreline of approximately 523 km (325 mi), and a volume of 3.46×10^8 m^3 (281,900 acre-ft). The towns of Mount Holly and Belmont, North Carolina, take their raw water supplies from Lake Wylie. The communities of Chester, Fort Lawn, Fort Mill, Great Falls, Lancaster, Mitford, Riverview, and Rock Hill, South Carolina, obtain at least a portion of their municipal water supplies from the Catawba River within 80 km (50 mi) downstream from the site (Duke Power 1997:2-41, table 2-52).

In 1997, the plant employed 1,232 persons (DOE 1999). The Catawba reactors are operated by Duke Power Company. The operating licenses (Nos. NPF–35 and NPF–52) for Units 1 and 2 were granted in 1985 and 1986 and expire in 2024 and 2026, respectively (NRC 1991:vol. 1, 2-2; 1997). The population within an 80-km (50-mi) radius of these reactors is estimated to be 1,656,093 (Duke Power 1997:table 2-13).

Reactor cooling is accomplished using mechanical draft cooling towers, with water obtained from Lake Wylie (Duke Power 1997). During normal operations of Catawba, cooling water is pumped from the Beaver Dam Creek arm of Lake Wylie at a rate of 266,680 million l/yr (70,450 million gal/yr) and returned to Big Allison Creek at a rate of 172,902 million l/yr (45,676 million gal/yr). The net difference in water (93,779 million l/yr [24,774 million gal/yr]) is due to evaporation in the cooling towers (DOE 1999).

New (unirradiated) fuel assemblies are dry stored in racks located in the two New Fuel Storage Buildings. Each New Fuel Storage Building is designed to accommodate 98 fuel assemblies (a total of 196 assemblies). Spent (irradiated) fuel assemblies are stored in two spent fuel pools in the two fuel buildings. The spent fuel storage pools have a total capacity of 2,836 assemblies (Duke Power 1997:9-3–9-6). Security at the site is provided in accordance with U.S. Nuclear Regulatory Commission (NRC) regulations and includes security checkpoints, barbed wire fencing, surveillance cameras, and intruder detection. More information about these reactors can be found at the NRC Web site at http://www.nrc.gov/OPA/finder.htm (NRC 1999) and in NRC Docket Nos. 50–413 and 50–414.

3.7.1.1 Air Quality

Catawba is within the Metropolitan Charlotte, North Carolina, Air Quality Control Region (AQCR) #167. None of the areas within the site or York County are designated as nonattainment areas with respect to the National Ambient Air Quality Standards (NAAQS) for criteria air pollutants (EPA 1998a).

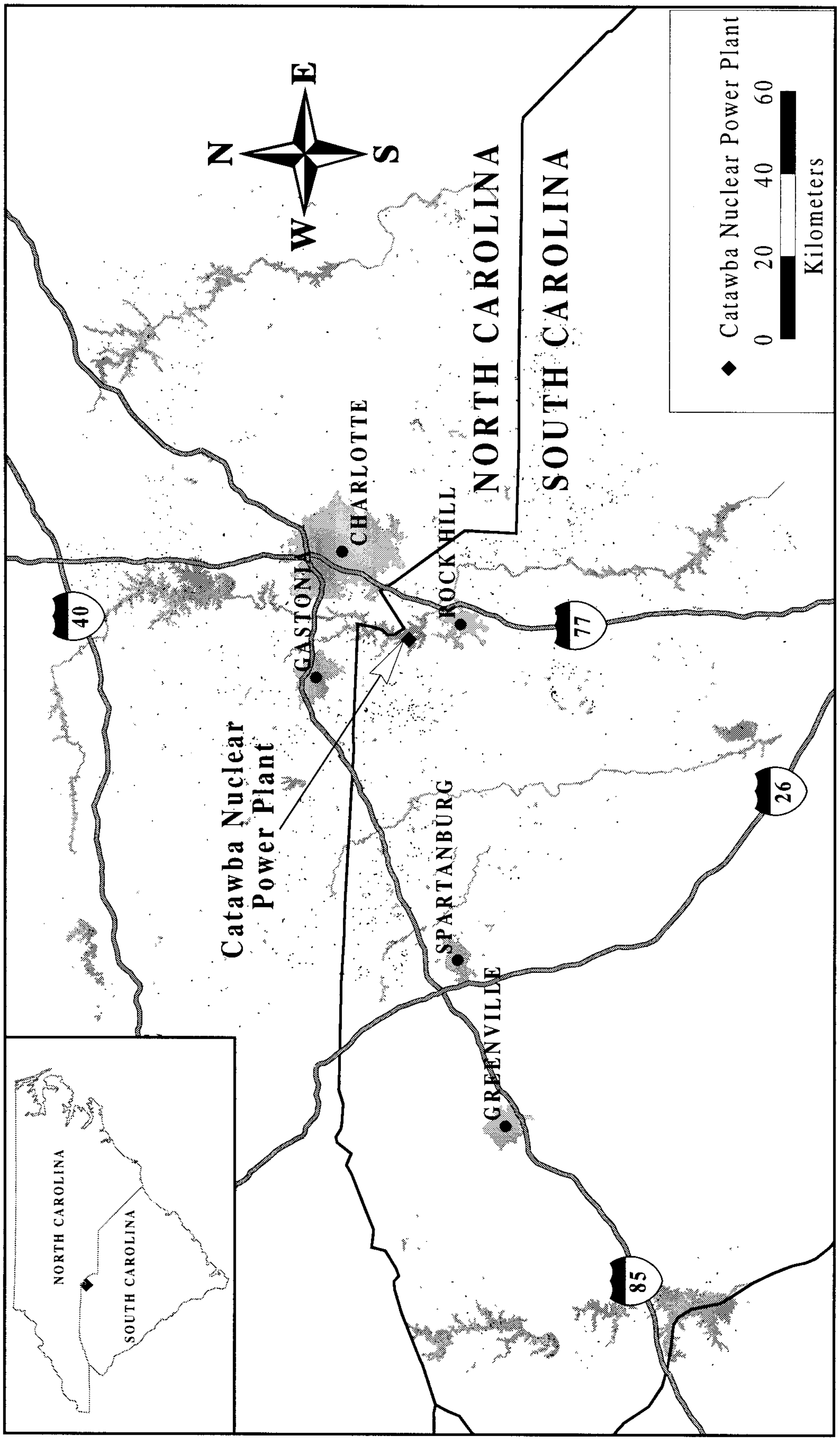


Figure 3.7–1. Catawba Nuclear Power Plant, South Carolina

Sources of criteria air pollutants from Catawba include five emergency diesel generators, a safe shutdown facility generator, and miscellaneous equipment such as trucks and forklifts. Table 3.7–1 provides a summary of criteria pollutant concentrations from operations of Catawba. The concentrations resulting from operations are well below the applicable ambient air quality standards even when background concentrations from other offsite sources are considered.

Table 3.7–1. Comparison of Contribution to Nonradiological Ambient Air Pollutant Concentrations From Catawba Sources With National Ambient Air Quality Standards

Pollutant	Averaging Period	NAAQS ($\mu g/m^3$)	Catawba ($\mu g/m^3$)
Carbon monoxide	8 hours	10,000	978
	1 hour	40,000	1,400
Nitrogen dioxide	Annual	100	3.26
PM_{10}	Annual	50	0.02
	24 hours	150	65.9
$PM_{2.5}$	3-year annual	15	(a)
	24 hours (98th percentile over 3 years)	65	(a)
Sulfur dioxide	Annual	80	0.0418
	24 hours	365	26.9
	3 hours	1,300	60.4

[a] No data is available with which to assess $PM_{2.5}$ concentrations.
Key: NAAQS, National Ambient Air Quality Standards.
Note: Based on 1994–1995 emissions data for diesel generators.
Source: Modeled concentrations based on DOE 1999; EPA 1997.

3.7.1.2 Waste Management

Table 3.7–2 presents the 5-year average annual waste generation rates for Catawba.

Table 3.7–2. Annual Waste Generation for Catawba (m^3)

Waste Type	Generation Rate
LLW	50
Mixed LLW	0.6[a]
Hazardous waste	29[a]
Nonhazardous waste	
Liquid	60,794[b]
Solid	909[a]

[a] Values converted from kilograms assuming a waste density such that 1 m^3 = 1,000 kg.
[b] Assuming sanitary wastewater is generated at the same rate 365 days per year.
Key: LLW, low-level waste.
Source: DOE 1999.

The waste disposal systems provide all equipment necessary to collect, process, store, and prepare for disposal of all radioactive liquid and solid wastes produced as a result of reactor operations. Potentially radioactive liquids may originate from a variety of sources, including the steam generator blowdown system, ventilation unit condensate system, drainage system sumps, laboratory drains, personnel decontamination area drains, decontamination system, sampling system, and laundry drains. Potentially radioactive liquid wastes are collected and characterized as to the level of contamination present. If contamination is below regulated levels, liquids may be discharged to the circulating water discharge outfall in accordance with the National Pollutant Discharge Elimination System (NPDES) permit. If liquids are determined to be radioactively contaminated, they are treated by filtration, evaporation, or mixing and settling, or are sent to the demineralizers, before being discharged. Continuous radiation monitoring is provided for treated liquid waste before its release to the circulating water discharge outfall. Liquid waste is analyzed and monitored to ensure that radionuclide concentrations are maintained as low as practical and well within the limits of applicable regulations and permits (Duke Power 1997:11-9–11-27).

The radioactive solid waste disposal system provides facilities for holdup, packaging, and storage of wastes before shipment to offsite licensed treatment and disposal facilities. Radioactive solid waste may include evaporator concentrates, spent demineralizer resins, spent filters, laboratory wastes, rags, gloves, boots, brooms, and other miscellaneous tools and apparel that become contaminated during normal plant operations and maintenance. Treatment on the site may include dewatering and compaction, or solidification using a contractor-supplied mobile unit. Materials that are compressible are placed in 208-l (55-gal) drums for compaction. Spent radioactive filter cartridges are packaged in either 114-l (30-gal) or 208-l (55-gal) drums. Packaged wastes are stored in the filter cartridge storage bunker, low-activity-waste storage room, high-activity-waste storage room, solidification area, and waste shipping area before being shipped to an offsite treatment or disposal facility (Duke Power 1997:11-53–11-61).

The small quantities of mixed low-level waste (LLW) and hazardous waste generated are accumulated on the site before being shipped for commercial treatment and disposal in offsite permitted facilities. Nonhazardous solid wastes are generated by typical industrial processes and housekeeping activities and are collected on the site and managed off the site at the local permitted sanitary landfill. Nonhazardous sanitary wastewater is treated in the onsite sanitary wastewater treatment facility and then discharged to Lake Wylie (SCDHEC 1997:6).

3.7.1.3 Existing Human Health Risk

Major sources and levels of background radiation exposure to individuals within the vicinity of Catawba are shown in Table 3.7–3. Annual background radiation doses to individuals are expected to remain constant over time. Total dose to the population changes as population size changes. Background radiation doses are unrelated to reactor operations.

Releases of radionuclides to the environment from normal reactor operations provide another source of radiation exposure to populations within the vicinity of the site. The doses to the public resulting from these releases are shown in Table 3.7–4. These doses fall within regulatory limits and are small when compared with background exposure.

Based on a risk estimator of 500 cancer deaths per 1 million person-rem (5×10^{-4} fatal cancers per person-rem) to the public, the latent cancer fatality (LCF) risk to the maximally exposed member of the public due to radiological releases from normal reactor operations in 1997 is estimated to be 1.7×10^{-8}. That is, the estimated probability of this person dying from cancer from radiation exposure from 1 year of normal reactor operations is about 1 chance in 60 million.

Table 3.7–3. Sources of Radiation Exposure to Individuals in the Vicinity Unrelated to Catawba Operations

Source	Effective Dose Equivalent (mrem/yr)
Natural background radiation	
Cosmic and external and internal terrestrial radiation[a]	125
Radon in homes (inhaled)[b]	200[c]
Other background radiation[b]	
Diagnostic x rays and nuclear medicine	53
Weapons test fallout	<1
Air travel	1
Consumer and industrial products	10
Total	390

[a] Virginia Power 1998:11B-3.
[b] NCRP 1987:11, 40, 53.
[c] An average for the United States.

Table 3.7–4. Radiological Impacts on the Public From Operations of Catawba in 1997 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard[a]	**Actual**	**Standard[a]**	**Actual**	**Standard[a]**	**Actual**
Maximally exposed individual (mrem)	5	0.045	3	0.11	25	0.16
Population within 80 km (person-rem)[b]	None	4.0	None	4.3	None	8.3

[a] The standards for individuals are given in 10 CFR 50, Appendix I. The standard for the maximally exposed offsite individual (25 mrem/yr total body from all pathways) is given in 40 CFR 190.
[b] Population used: 1,656,093; this population dose was estimated for the year 2000 and is assumed to be representative for the year 1997.
Source: DOE 1999; Duke Power 1997:tables 2-13, 11-12, and 11-15.

Based on the same risk estimator, 0.0042 excess LCFs are projected among the population living within 80 km (50 mi) of Catawba in 1997. For perspective, this number can be compared with the number of fatal cancers expected in this population from all causes. The 1990 mortality rate associated with cancer for the entire population was 0.2 percent per year. Based on this national rate, the number of fatal cancers from all causes expected during 1997 in the population living within 80 km (50 mi) of Catawba was about 3,300. This number of expected fatal cancers is much higher than the estimated 0.0042 LCFs that could result from normal reactor operations in 1997.

Workers at the reactors receive the same background radiation dose as the general public; however, they receive an additional dose from normal operations of the reactors. Table 3.7–5 includes average, maximally exposed, and total occupational doses to reactor workers from operations in 1997. Based on a risk estimator of 400 cancer deaths per 1 million person-rem (4×10^{-4} fatal cancers per person-rem) among workers, the number of LCFs to reactor workers from 1997 normal operations is estimated to be 0.11.

3.7.1.4 Environmental Justice

Environmental justice concerns the environmental impacts that proposed actions may have on minority and low-income populations, and whether such impacts are disproportionate to those on the population as a whole

Table 3.7–5. Radiological Impacts on Involved Workers From Operations of Catawba in 1997

Number of badged workers[a]	420
Total dose (person-rem/yr)	265
Annual latent fatal cancers	0.11
Average worker dose (mrem/yr)	78
Annual risk of latent fatal cancer	3.1×10^{-5}

[a] A badged worker is equipped with an individual dosimeter.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (10 CFR 20). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999.

in the potentially affected area. In the case of Catawba, the potentially affected area includes parts of North Carolina and South Carolina.

The potentially affected area around Catawba is defined by a circle with an 80-km (50-mi) radius centered at these reactors (lat. 35°03′05″ N, long. 81°04′10″ W). The total population residing within that area in 1990 was 1,519,392. The proportion of the population that was considered minority was 20.7 percent. The same census data show that the percentage of minorities for the contiguous United States was 24.1, and the percentages of the States of North Carolina and South Carolina were 25.0 and 31.5, respectively (DOC 1992).

At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting 19.0 percent of the total population. Asians and Hispanics contributed about 0.7 percent, and Native Americans made up about 0.3 percent of the population (DOC 1992).

A breakdown of incomes in the potentially affected area is also available from the 1990 census data (DOC 1992). At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 159,596 persons (10.5 percent of the total population) residing within the potentially affected area around Catawba reported incomes below that threshold. Data obtained during the 1990 census also show that of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold and that the figures for North Carolina and South Carolina were 13.0 and 15.4 percent, respectively (DOC 1992).

3.7.2 McGuire Units 1 and 2 Site Overview

The McGuire nuclear power plant occupies 12,000 ha (30,000 acres) in northwestern Mecklenburg County, North Carolina, 27.4 km (17 mi) northwest of Charlotte, North Carolina (see Figure 3.7–2). The site is bounded to the west by the Catawba River and to the north by Lake Norman. Surrounding land is generally rural nonfarmland. Lake Norman, with a surface area of 13,156 ha (32,510 acres), a volume of 1,349 million m^3 (1,093,600 acre-ft) and a shoreline of 837 km (520 mi), stretches 54.7 km (34 mi) from Cowans Ford Dam to the tailrace of Lookout Lake. The Charlotte municipal water intake is 18 km (11.2 mi) downstream from the site (Duke Power 1996:2-3, 2-27, 2-28). In addition, the communities of Belmont, Gastonia, and Mount Holly, North Carolina, and Chester, Fort Lawn, Fort Mill, Lancaster, Mitford, Riverview, and Rock Hill, South Carolina, obtain at least a portion of their municipal water supplies from the Catawba River within 80 km (50 mi) downstream from the site (Duke Power 1997:2-41, table 2-52).

In 1997, the plant employed 1,238 persons (DOE 1999). The McGuire reactors are operated by Duke Power Company. The operating licenses (Nos. NPF–9 and NPF–17) for these reactors were granted in 1981 and 1983, and expire in 2021 and 2023, respectively (NRC 1991:vol. 1, 2-2; 1997). The population within an 80-km

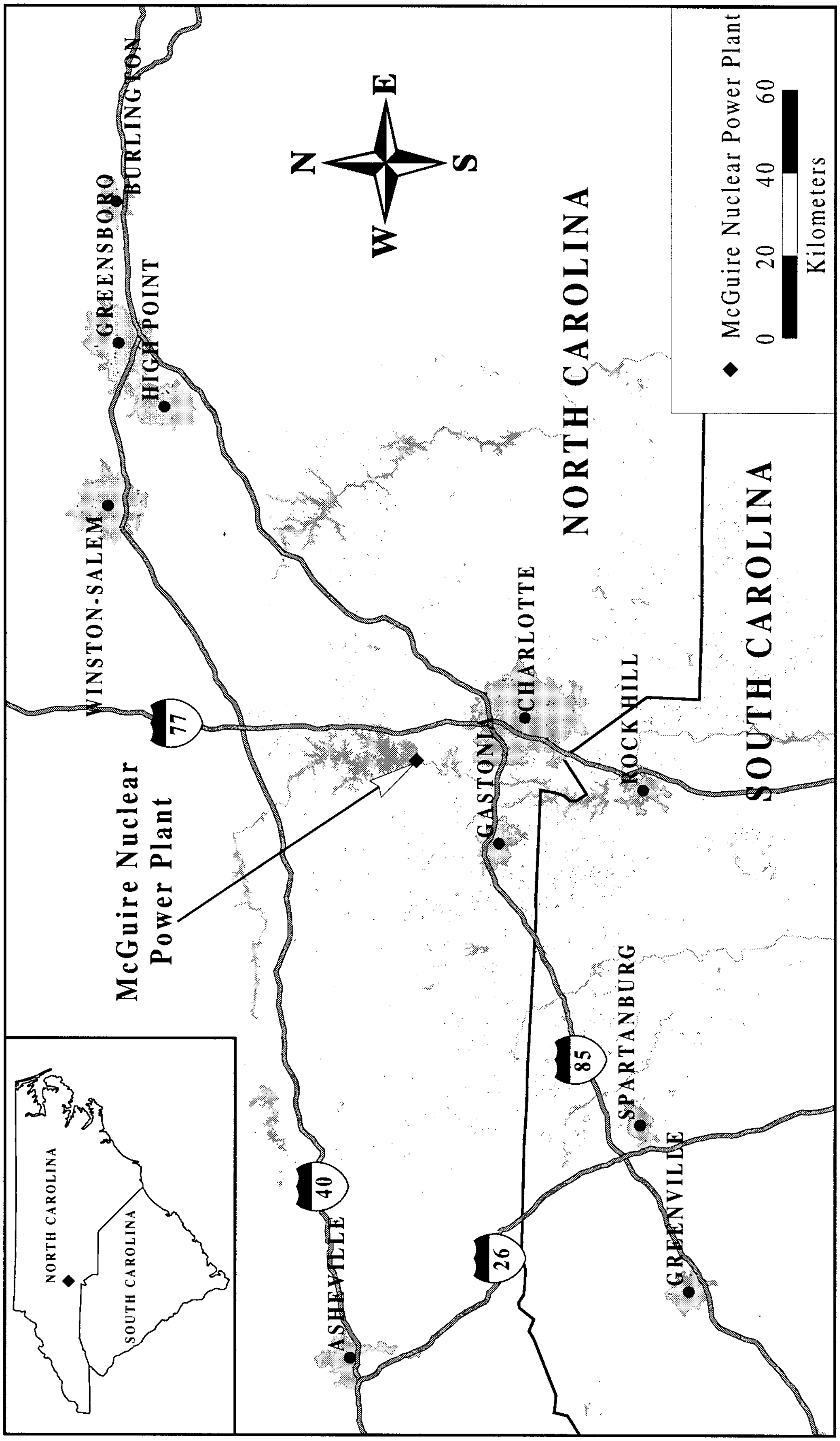


Figure 3.7–2. McGuire Nuclear Power Plant, North Carolina

(50-mi) radius of these reactors is estimated to be 2,140,720 (Duke Power 1996:table 2-1). Reactor cooling is accomplished using a once-through cooling system. Cooling water is withdrawn from Lake Norman at a rate of 3,512,969 million l/yr (928,031 million gal/yr) and discharged back into Lake Norman at a rate of 3,483,283 million l/yr (920,189 million gal/yr). The net difference in water (29,685 million l/yr [7,842 million gal/yr]) is due to evaporation (DOE 1999).

New (unirradiated) fuel assemblies are dry stored in racks located in the two New Fuel Storage Vaults. Each New Fuel Storage Vault is designed to accommodate 96 fuel assemblies (a total of 192 assemblies). Spent (irradiated) fuel assemblies are stored in two spent fuel pools in the two Auxiliary Buildings. The two spent fuel storage pools have a total capacity of 2,926 assemblies. New fuel can also be stored in the spent fuel pools (Duke Power 1996:9-3–9-8). Security at the site is provided in accordance with NRC regulations and includes security checkpoints, barbed wire fencing, surveillance cameras, and intruder detection. More information about these reactors can be found at the NRC Web site at http://www.nrc.gov/OPA/finder.htm (NRC 1999) and in NRC Docket Nos. 50–369 and 50–370.

3.7.2.1 Air Quality

McGuire is within the Metropolitan Charlotte AQCR #167. None of the areas within the site or Mecklenberg County are designated as nonattainment areas with respect to the NAAQS for criteria air pollutants (EPA 1998b).

Sources of criteria air pollutants from McGuire include five emergency diesel generators, a safe shutdown facility generator, and miscellaneous equipment such as trucks and forklifts. Table 3.7–6 provides a summary of criteria pollutant concentrations from operations of McGuire. The concentrations resulting from operations are well below the applicable ambient air quality standards even when background concentrations from other offsite sources are considered.

Table 3.7–6. Comparison of Contribution to Nonradiological Ambient Air Pollutant Concentrations From McGuire Sources With National Ambient Air Quality Standards

Pollutant	Averaging Period	NAAQS ($\mu g/m^3$)	McGuire ($\mu g/m^3$)
Carbon monoxide	8 hours	10,000	1,060
	1 hour	40,000	1,510
Nitrogen dioxide	Annual	100	2.55
PM_{10}	Annual	50	0.0799
	24 hours	150	71.2
$PM_{2.5}$	3-year annual	15	(a)
	24 hours (98th percentile over 3 years)	65	(a)
Sulfur dioxide	Annual	80	0.0336
	24 hours	365	29.9
	3 hours	1,300	67.4

[a] No data is available with which to assess $PM_{2.5}$ concentrations.
Key: NAAQS, National Ambient Air Quality Standards.
Note: Based on 1994–1997 emissions data for diesel generators.
Source: Modeled concentrations based on DOE 1999; EPA 1997.

3.7.2.2 Waste Management

Table 3.7–7 presents the 5-year average annual waste generation rates for McGuire.

Table 3.7–7. Annual Waste Generation for McGuire (m^3)

Waste Type	Generation Rate
LLW	42.2
Mixed LLW	0.18[a]
Hazardous waste	28.6[a]
Nonhazardous waste	
Liquid	49,740[b]
Solid	1,136[a]

[a] Values converted from kilograms assuming a waste density such that 1 m^3 = 1,000 kg.
[b] Assuming sanitary wastewater is generated at the same rate 365 days per year.
Key: LLW, low-level waste.
Source: DOE 1999.

The waste disposal systems provide all equipment necessary to collect, process, store, and prepare for disposal of all radioactive liquid and solid wastes produced as a result of reactor operations. Potentially radioactive liquids may originate from a variety of sources, including the steam generator blowdown system, ventilation unit condensate system, drainage system sumps, laboratory drains, personnel decontamination area drains, decontamination system, sampling system, and laundry drains. Potentially radioactive liquid wastes are collected and characterized as to the level of contamination present. If contamination is below regulated levels, liquids may be discharged to the circulating water discharge outfall in accordance with the NPDES permit. If liquids are determined to be radioactively contaminated, they are treated by filtration, evaporation, or mixing and settling, or are sent to the demineralizers, before being discharged. Continuous radiation monitoring is provided for treated waste before its release to the circulating water discharge outfall. Liquid waste is analyzed and monitored to ensure that radionuclide concentrations are maintained as low as practical and well within the limits of applicable regulations and permits (Duke Power 1996:11-9–11-26).

The radioactive solid waste disposal system provides facilities for holdup, packaging, and storage of wastes before shipment to offsite licensed treatment and disposal facilities. Radioactive solid waste may include evaporator concentrates, spent demineralizer resins, spent filters, laboratory wastes, contaminated oils, rags, gloves, boots, sweepings, brooms, and other miscellaneous tools and apparel that become contaminated during normal plant operations and maintenance. Treatment on the site may include dewatering, or solidification using a contractor-supplied mobile unit. Low-activity solid wastes, such as rags, clothing, and sweepings, are loaded directly into storage containers for shipment to an offsite treatment or disposal facility. Spent radioactive filter cartridges are packaged in drums or other waste containers, with spent resin solidified, if required. The disposal of slightly contaminated sludge from the wastewater treatment plant is carried out by landspreading the sludge on a site continguous to McGuire using a method approved by the State of North Carolina and NRC. Packaged wastes are stored in the filter storage bunker, solidified liner storage bunker, and the shielded storage bunker before being shipped to an offsite treatment or disposal facility (Duke Power 1996:11-49–11-56).

The small quantities of mixed LLW and hazardous waste generated are accumulated on the site before being shipped for commercial treatment and disposal in offsite permitted facilities. Nonhazardous solid wastes are generated by typical industrial processes and housekeeping activities and are collected on the site and managed off the site at the local permitted sanitary landfill. Nonhazardous sanitary wastewater is discharged to the Charlotte Mecklenburg Utility Department sanitary sewer system (Duke Power 1994).

3.7.2.3 Existing Human Health Risk

Major sources and levels of background radiation exposure to individuals within the vicinity of McGuire are shown in Table 3.7–8. Annual background radiation doses to individuals are expected to remain constant over time. Total dose to the population changes as population size changes. Background radiation doses are unrelated to reactor operations.

Table 3.7–8. Sources of Radiation Exposure to Individuals in the Vicinity Unrelated to McGuire Operations

Source	Effective Dose Equivalent (mrem/yr)
Natural background radiation	
Cosmic and external and internal terrestrial radiation[a]	125
Radon in homes (inhaled)[b]	200[c]
Other background radiation[b]	
Diagnostic x rays and nuclear medicine	53
Weapons test fallout	<1
Air travel	1
Consumer and industrial products	10
Total	390

[a] Virginia Power 1998:11B-3.
[b] NCRP 1987:11, 40, 53.
[c] An average for the United States.

Releases of radionuclides to the environment from normal reactor operations provide another source of radiation exposure to populations within the vicinity of the site. The doses to the public resulting from these releases are shown in Table 3.7–9. These doses fall within regulatory limits and are small when compared with background exposure.

Table 3.7–9. Radiological Impacts on the Public From Operations of McGuire in 1997 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard[a]	Actual	Standard[a]	Actual	Standard[a]	Actual
Maximally exposed individual (mrem)	5	0.033	3	0.065	25	0.098
Population within 80 km (person-rem)[b]	None	2.8	None	93	None	96

[a] The standards for individuals are given in 10 CFR 50, Appendix I. The standard for maximally exposed offsite individual (25 mrem/yr total body from all pathways) is given in 40 CFR 190.
[b] Population used: 2,140,720; this population dose was estimated for the year 2000 and is assumed to be representative for the year 1997.
Source: DOE 1999; Duke Power 1974:5.3-7, table 5.3.5-1; 1996:table 2-1.

Based on a risk estimator of 500 cancer deaths per 1 million person-rem (5×10^{-4} fatal cancers per person-rem) to the public, the LCF risk to the maximally exposed member of the public due to radiological releases from normal reactor operations in 1997 is estimated to be 4.9×10^{-8}. That is, the estimated probability of this person dying from cancer from radiation exposure from 1 year of normal reactor operations is about 1 chance in 20 million.

Based on the same risk estimator, 0.048 excess LCFs are projected among the population living within 80 km (50 mi) of McGuire in 1997. For perspective, this number can be compared with the number of fatal cancers expected in this population from all causes. The 1990 mortality rate associated with cancer for the entire population was 0.2 percent per year. Based on this national rate, the number of fatal cancers from all causes expected during 1997 in the population living within 80 km (50 mi) of McGuire was about 4,280. This number of expected fatal cancers is much higher than the estimated 0.048 LCFs that could result from normal reactor operations in 1997.

Workers at the reactors receive the same background radiation dose as the general public; however, they receive an additional dose from normal operations of the reactors. Table 3.7–10 includes average, maximally exposed, and total occupational doses to reactor workers from operations in 1997. Based on a risk estimator of 400 cancer deaths per 1 million person-rem (4×10^{-4} fatal cancers per person-rem) among workers, the number of LCFs to reactor workers from 1997 normal operations is estimated to be 0.20.

Table 3.7–10. Radiological Impacts on Involved Workers From Operations of McGuire in 1997

Number of badged workers[a]	3992
Total dose (person-rem/yr)	492
Annual latent fatal cancers	0.20
Average worker dose (mrem/yr)	123
Annual risk of latent fatal cancer	4.9×10^{-5}

[a] A badged worker is equipped with an individual dosimeter.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (10 CFR 20). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999.

3.7.2.4 Environmental Justice

Environmental justice concerns the environmental impacts that proposed actions may have on minority and low-income populations, and whether such impacts are disproportionate to those on the population as a whole in the potentially affected area. In the case of McGuire, the potentially affected area includes parts of North Carolina and South Carolina.

The potentially affected area around McGuire is defined by a circle with an 80-km (50-mi) radius centered at these reactors (lat. 35°25′59″ N, long. 80°56′55″ W). The total population residing within that area in 1990 was 1,738,966. The proportion of the population that was considered minority was 17.8 percent. The same census data show that the percentage of minorities for the contiguous United States was 24.1, and the percentages of the States of North and South Carolina were 25.0 and 31.5, respectively (DOC 1992).

At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting 15.9 percent of the total population. Hispanics and Asians contributed about 0.7 percent, and Native Americans made up about 0.3 percent of the population (DOC 1992).

A breakdown of incomes in the potentially affected area is also available from the 1990 census data (DOC 1992). At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 170,956 persons (9.8 percent of the total population) residing within the potentially affected area around McGuire reported incomes below that threshold. Data obtained during the 1990 census also show that of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold, and that the figures for North Carolina and South Carolina were 13.0 and 15.4 percent, respectively (DOC 1992).

3.7.3 North Anna Units 1 and 2 Site Overview

The North Anna nuclear power plant occupies 422 ha (1,043 acres) in Louisa County, Virginia, approximately 64.4 km (40 mi) north-northwest of Richmond, Virginia, and 113 km (70 mi) southwest of Washington, D.C. (see Figure 3.7–3). The largest community within 16 km (10 mi) of the site is the town of Mineral in Louisa County. The site is on a peninsula on the southern shore of Lake Anna. Lake Anna is approximately 27.4 km (17 mi) long, with a surface area of 5,260 ha (13,000 acres) and 322 km (200 mi) of shoreline. The reservoir contains approximately 380 billion l (100 billion gal) of water (Virginia Power 1998:2.1-1, 2.1-2).

In 1997, the plant employed 552 persons (DOE 1999). The North Anna reactors are operated by the Virginia Power Company. The operating licenses (Nos. NPF–4 and NPF–7) for these reactors were granted in 1978 and 1980, and expire in 2018 and 2020, respectively (NRC 1991:vol. 1, 2-2; 1997). It is estimated that the population within an 80-km (50-mi) radius of the reactor is 1,363,945 (Virginia Power 1998:2.1-21).

Reactor cooling is accomplished using a once-through cooling system with water obtained from Lake Anna (Virginia Power 1998:2.1-2). The rate of cooling water withdrawal is 5,565,000 million l/yr (1,470,000 million gal/yr), with all water returned to Lake Anna (DOE 1999). There are no known industrial users downstream from the site until some 97 km (60 mi) downstream at West Point, where a large pulp and paper manufacturing plant is located. There are no known potable water withdrawals along the entire stretch of the river downstream to West Point, where the river becomes brackish (Virginia Power 1998:2.4-3).

New (unirradiated) fuel assemblies are dry stored in the new fuel storage area of the fuel building. The new fuel storage area has a capacity of 126 fuel assemblies. Spent (irradiated) fuel assemblies are stored under water in the spent fuel pit in the fuel building. The spent fuel storage pit has a capacity of 1,737 fuel assemblies (Virginia Power 1998:9.1-1, 9.1-2). Dry cask storage is being developed and is expected to have a capacity of an additional 1,824 assemblies (NRC 1998). Security at the site is provided in accordance with NRC regulations and includes security checkpoints, barbed wire fencing, surveillance cameras, and intruder detection. More information about these reactors can be found at the NRC Web site at http://www.nrc.gov/OPA/finder.htm (NRC 1999) and in NRC Docket Nos. 50–338 and 50–339.

3.7.3.1 Air Quality

North Anna is within the Northeastern Virginia AQCR #224. None of the areas within the site or Louisa County are designated as nonattainment areas with respect to the NAAQS for criteria air pollutants (EPA 1998c).

Sources of criteria air pollutants from North Anna include two auxiliary boilers, four emergency diesel generators, a station blackout generator, and miscellaneous equipment such as trucks and forklifts. Table 3.7–11 provides a summary of criteria pollutant concentrations from operations of North Anna. The concentrations resulting from operationsare well below the applicable ambient air quality standards even when background concentrations from other offsite sources are considered.

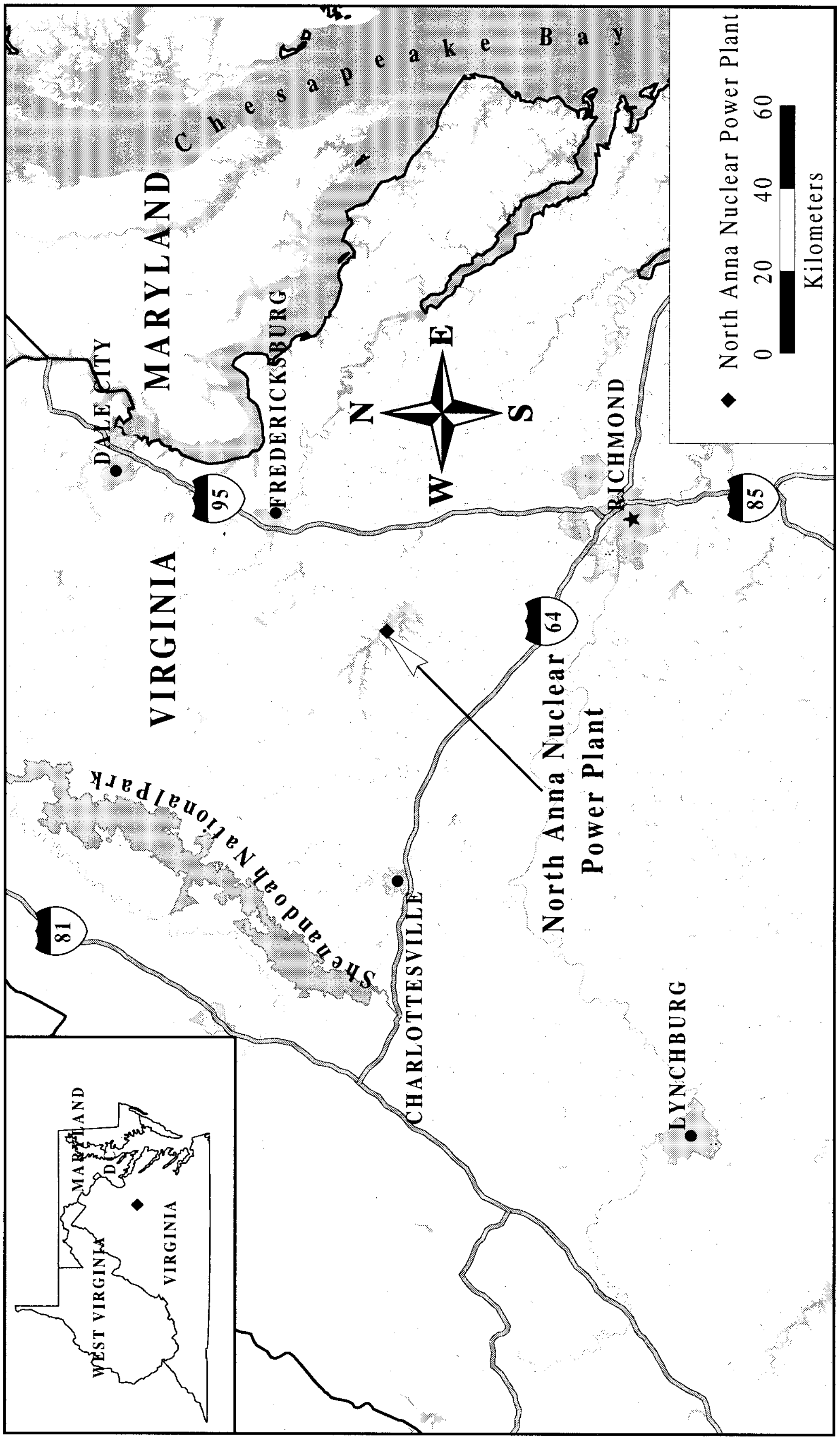


Figure 3.7–3. North Anna Nuclear Power Plant, Virginia

Table 3.7–11. Comparison of Contribution to Nonradiological Ambient Air Pollutant Concentrations From North Anna Sources With National Ambient Air Quality Standards

Pollutant	Averaging Period	NAAQS ($\mu g/m^3$)	North Anna ($\mu g/m^3$)
Carbon monoxide	8 hours	10,000	416
	1 hour	40,000	594
Nitrogen dioxide	Annual	100	0.00504
PM_{10}	Annual	50	0.00407
	24 hours	150	15.4
$PM_{2.5}$	3-year annual	15	(a)
	24 hours (98th percentile over 3 years)	65	(a)
Sulfur dioxide	Annual	80	0.0167
	24 hours	365	63
	3 hours	1,300	142

[a] No data is available with which to assess $PM_{2.5}$ concentrations.
Key: NAAQS, National Ambient Air Quality Standards.
Note: Based on 1997 emissions data for diesel generators.
Source: Modeled concentrations based on DOE 1999; EPA 1997.

3.7.3.2 Waste Management

Table 3.7–12 presents the 5-year average annual waste generation rates for North Anna.

Table 3.7–12. Annual Waste Generation for North Anna (m^3)

Waste Type	Generation Rate
LLW	236.6[a]
Mixed LLW	0
Hazardous waste	11.4
Nonhazardous waste	
Liquid	682
Solid	10,400

[a] Two-year average (1996–1997).
Key: LLW, low-level waste.
Source: DOE 1999.

The waste disposal systems provide all equipment necessary to collect, process, store, and prepare for disposal of all radioactive liquid and solid wastes produced as a result of reactor operations. Potentially radioactive liquids may originate from a variety of sources, including the boron recovery system, steam generator blowdown system, drainage system sumps, laboratory drains, personnel decontamination area drains, decontamination system, sampling system, laundry drains, and spent resin flush system. Potentially radioactive liquid wastes are collected and characterized as to the level of contamination present. If contamination is below regulated levels, liquids may be discharged to the circulating water discharge outfall in accordance with the NPDES permit. If liquids are determined to be radioactively contaminated, they are treated by the ion exchange filtration system or demineralizers to reduce contamination before being discharged. Continuous radiation monitoring is provided for treated liquid waste before its release to the circulating water discharge outfall. Liquid waste is analyzed and monitored to ensure that radionuclide concentrations are maintained as low as practical and well within the limits of applicable regulations and permits (Virginia Power 1998:11.2-1, 11.2-2).

The radioactive solid waste disposal system provides facilities for holdup, packaging, and storage of wastes before shipment to offsite treatment and disposal facilities. Radioactive solid waste may include spent resin slurries, spent filter cartridges, rags, gloves, boots, brooms, and other miscellaneous tools and apparel that become contaminated during normal plant operations and maintenance. Contaminated solid materials resulting from station maintenance are stored in specified areas of the auxiliary building and the decontamination building. Materials that are compressible are placed in 208-l (55-gal) drums for compaction at the bailing facility. Compressible materials and other contaminated solid materials that are not placed in drums are placed in 6.1-m (20-ft) seavans for shipment to offsite licensed treatment and disposal facilities. Contaminated metallic materials and highly contaminated solid objects are placed inside disposable containers for shipment to a disposal facility (Virginia Power 1998:11.5-1–11.5-3).

The small quantities of mixed LLW and hazardous waste generated are accumulated on the site before being shipped for commercial treatment and disposal in offsite permitted facilities. Nonhazardous solid wastes are generated by typical industrial processes and housekeeping activities and are collected on the site and managed off the site at the local permitted sanitary landfill. Nonhazardous sanitary wastewater is treated in the onsite sanitary wastewater treatment facility and then discharged to Lake Anna (VADEQ 1997:9, 28).

3.7.3.3 Existing Human Health Risk

Major sources and levels of background radiation exposure to individuals within the vicinity of North Anna are shown in Table 3.7–13. Annual background radiation doses to individuals are expected to remain constant over time. Total dose to the population changes as population size changes. Background radiation doses are unrelated to reactor operations.

Table 3.7–13. Sources of Radiation Exposure to Individuals in the Vicinity Unrelated to North Anna Operations

Source	Effective Dose Equivalent (mrem/yr)
Natural background radiation	
Cosmic and external and internal terrestrial radiation[a]	125
Radon in homes (inhaled)[b]	200[c]
Other background radiation[b]	
Diagnostic x rays and nuclear medicine	53
Weapons test fallout	<1
Air travel	1
Consumer and industrial products	10
Total	390

[a] Virginia Power 1998:11B-3.
[b] NCRP 1987:11, 40, 53.
[c] An average for the United States.

Releases of radionuclides to the environment from normal reactor operations provide another source of radiation exposure to populations within the vicinity of the site. The doses to the public resulting from these releases are shown in Table 3.7–14. These doses fall within regulatory limits and are small when compared with background exposure.

Based on a risk estimator of 500 cancer deaths per 1 million person-rem (5×10^{-4} fatal cancers per person-rem) to the public, the LCF risk to the maximally exposed member of the public due to radiological releases from normal reactor operations in 1997 is estimated to be 1.4×10^{-7}. That is, the estimated probability of this person

Table 3.7–14. Radiological Impacts on the Public From Operations of North Anna in 1997 (Total Effective Dose Equivalent)

	Atmospheric Releases		Liquid Releases		Total	
Members of the Public	**Standard[a]**	**Actual**	**Standard[a]**	**Actual**	**Standard[a]**	**Actual**
Maximally exposed individual (mrem)	5	6.1×10^{-4}	3	0.28	25	0.29
Population within 80 km (person-rem)[b]	None	6.0	None	9.0	None	15.0

[a] The standards for individuals are given in 10 CFR 50, Appendix I. The standard for the maximally exposed offsite individual (25 mrem/yr total body from all pathways) is given in 40 CFR 190.
[b] Population used: 1,614,983; this population dose was estimated for the year 2000 and is assumed to be representative for the year 1997. Population doses were ratioed to reflect latest census data projections.
Source: DOE 1999; Virginia Power 1998:2.1-21, 11B-3, 11.3-13.

dying from cancer from radiation exposure from 1 year of normal reactor operations is about one chance in seven million.

Based on the same risk estimator, 0.0075 excess LCFs are projected among the population living within 80 km (50 mi) of North Anna In 1997. For perspective, this number can be compared with the number of fatal cancers expected in this population from all causes. The 1990 mortality rate associated with cancer for the entire population was 0.2 percent per year. Based on this national rate, the number of fatal cancers from all causes expected during 1997 in the population living within 80 km (50 mi) of North Anna was about 3,230. This number of expected fatal cancers is much higher than the estimated 0.0075 LCFs that could result from normal reactor operations in 1997.

Workers at the reactors receive the same background radiation dose as the general public, however, they receive an additional dose from normal operations of the reactors. Table 3.7–15 includes average, maximally exposed, and total occupational doses to reactor workers from operations in 1997. Based on a risk estimator of 400 cancer deaths per 1 million person-rem (4×10^{-4} fatal cancers per person-rem) among workers, the number of LCFs to reactor workers from 1997 normal operations is estimated to be 0.041.

Table 3.7–15. Radiological Impacts on Involved Workers From Operations of North Anna in 1997

Number of badged workers[a]	2,243
Total dose (person-rem/yr)	103
Annual latent fatal cancers	0.041
Average worker dose (mrem/yr)	46
Annual risk of latent fatal cancer	1.8×10^{-5}

[a] A badged worker is equipped with an individual dosimeter.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (10 CFR 20). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999.

3.7.3.4 Environmental Justice

Environmental justice concerns the environmental impacts that proposed actions may have on minority and low-income populations, and whether such impacts are disproportionate to those on the population as a whole in the potentially affected area. In the case of North Anna, the potentially affected area includes parts of Maryland and Virginia.

The potentially affected area around North Anna is defined by a circle with an 80-km (50-mi) radius centered around these reactors (lat. 38°03′37″ N, long. 77°47′24″ W). The total population residing within that area in 1990 was 1,286,156. The proportion of the population that was considered minority was 21.9 percent. The same census data show that the percentages of minorities for the contiguous United States was 24.1, and the percentage of the States of Maryland and Virginia were 30.4 and 24.0, respectively (DOC 1992).

At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting 18.9 percent of the total population. Asians contributed about 1.5 percent, and Hispanics, about 1.4 percent. Native Americans made up about 0.3 percent of the population (DOC 1992).

A breakdown of incomes in the potentially affected area is also available from the 1990 census data (DOC 1992). At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 88,162 persons (6.9 percent of the total population) residing within the potentially affected area around North Anna reported incomes below that threshold. Data obtained during the 1990 census also show that of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold, and that the figures for Maryland and Virginia were 8.3 and 10.3 percent, respectively (DOC 1992).

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V. Environmental Impact Analysis

This section provides a description of the potential environmental impacts associated with operating the proposed reactors with mixed oxide (MOX) fuel. It replaces Section 4.28 in the Surplus Plutonium Disposition Draft Environmental Impact Statement *(SPD Draft EIS) and will be included in the SPD Final EIS under the same section number.*

The impacts associated with using mixed oxide (MOX) fuel during normal operations of the proposed reactors are not expected to be much different from those associated with the continued use of low-enriched uranium (LEU) fuel in these reactors. The radiation dose from normal operations to the surrounding population and workers at the reactors is not expected to change. Similarly, the amount of radioactive and hazardous waste generated during normal operation is expected to be the same regardless of fuel type. No changes are expected in the air or water quality surrounding the sites. If MOX fuel is used in these reactors, it is expected that about 5 percent more spent fuel would be generated by the reactors than if they continued to use LEU fuel. This increase in fuel is needed mainly during the transition from LEU fuel to a partial MOX core to maintain peaking in the reactors below design and regulatory limits and to compensate for greater end-of-cycle reactivity. Some additional assemblies are also expected to be needed by the North Anna reactors during equilibrium cycles. No other resource areas are expected to be impacted by the use of MOX fuel at any of these reactor sites. There are differences in the expected risk of reactor accidents from the use of MOX fuel. Some accidents would be expected to result in lower consequences to the surrounding population and, thus, lower risks, while others would be expected to result in higher consequences and higher risks. The largest estimated increase in risk to the surrounding population due to the use of MOX fuel is an estimated 15 percent increase in the risk of latent cancer fatalities (LCFs) associated with an interfacing systems loss-of-coolant accident (ISLOCA) at North Anna. The probability or frequency of this accident occurring at North Anna is estimated to be 2.4×10^{-7} or 1 chance in 4.2 million per year of reactor operation.

4.28 IMPACTS OF IRRADIATING MOX FUEL AT REACTOR SITES

The environmental impacts described in the following sections are based on using a partial MOX core (i.e., up to 40 percent MOX fuel) instead of an LEU core in existing, commercial light water reactors. As discussed in Section IV, the proposed sites are the Catawba Nuclear Station near York, South Carolina; the McGuire Nuclear Station near Huntersville, North Carolina; and the North Anna Power Station near Mineral, Virginia. Each of the proposed sites has two operating reactors that would be used to irradiate MOX fuel assemblies. All of these sites have been operating safely for a number of years. Table 4.28–1 indicates operating statistics for each of the proposed reactors.

Table 4.28–1. Reactor Operating Information

Reactor	Operator	Capacity (net MWe)	Date of First Operation (mo/yr)
Catawba 1	Duke Power	1,129	1/85
Catawba 2	Duke Power	1,129	5/86
McGuire 1	Duke Power	1,129	7/81
McGuire 2	Duke Power	1,129	5/83
North Anna 1	Virginia Power	900	4/78
North Anna 2	Virginia Power	887	8/80

Source: DOE 1996a.

Since 1978, the U.S. Nuclear Regulatory Commission (NRC) has conducted a systematic assessment of licensee performance (SALP) of each nuclear power plant in the United States. During a SALP, board members review inspection results; enforcement actions that may have been taken against a licensee; and results of the latest plant performance reviews, performance indicators, licensee self-assessments, third-party assessments, and indepth discussions with licensees. Regional managers used the SALP findings to identify those areas at a plant that required increased inspection. (In September 1998, NRC suspended the SALP program for an interim period while NRC reviews its nuclear power plant assessment process [NRC 1998].) Table 4.28–2 shows the results of the most recent SALP undertaken by NRC at each of the proposed reactor sites.

Table 4.28–2. Results of Systematic Assessment of Licensee Performance

	Catawba	McGuire	North Anna
Date of latest SALP	6/97	4/97	2/97
Operations	Superior	Superior	Superior
Maintenance	Good	Good	Superior
Engineering	Superior	Good	Good
Plant support	Superior	Superior	Superior

Source: NRC 1997a, 1997b, 1997c.

In accordance with the alternatives presented under the hybrid approach (i.e., Alternatives 2 through 10 in the SPD Draft EIS), all of these reactors would use MOX fuel to partially fuel their reactor cores. Up to 33 t (36 tons) of surplus plutonium could be used in MOX fuel at these reactors from 2007–2022. In March 1999, DOE awarded a contract to Duke Engineering & Services; COGEMA, Inc.; and Stone & Webster (known as DCS) to provide MOX fuel fabrication and reactor irradiation services contingent on the selection (in the SPD EIS Record of Decision) of the hybrid approach described in Chapter 2 of the SPD Draft EIS.

The analyses prepared for this section are based on information provided by DCS. Data was also developed independently to support these analyses. This included projecting the population around the proposed reactor sites to 2015[1] and compiling information related to the topography surrounding the proposed reactor sites for evaluating air dispersal patterns. Information to support accident analysis was also provided by Oak Ridge National Laboratory (ORNL). Based on information provided by DCS, ORNL developed expected ratios of radionuclide activities in MOX fuel versus that in LEU fuel as it would be used in the reactors. Standard models for estimating radiation doses from normal operations and accident scenarios, and estimating air pollutant concentrations at the proposed reactor sites were run using this new information. Human health risk and accident analyses were performed for a maximum use of a 40 percent MOX core, which is a conservative estimate of the amount of MOX fuel that would be used in each of the reactors.

Under the MOX approach, both MOX and LEU fuel assemblies would be loaded into the reactor. The MOX assemblies would remain in the core for two 18-month cycles and the LEU assemblies for either two or three 18-month cycles, in accordance with the plant's current operating schedule. When the MOX fuel completes a normal cycle, it would be withdrawn from the reactor in accordance with the plant's standard refueling procedures and placed in the plant's spent fuel pool for cooling alongside other spent fuel. No changes are expected in the plant's spent fuel storage plans to accommodate the spent MOX fuel. Eventually the spent fuel would be shipped to a potential geologic repository for permanent disposal.

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[1] Population projections for the area encompassed in a 80-km (50-mi) radius around the proposed reactor sites were projected to 2015 to approximate the midpoint of the irradiation services program. By 2015, the MOX program would be firmly established at all of the proposed reactor sites and would be expected to remain stable through the end of the program. Using 1990 census data as the base year and state-provided population increase factors for all counties included in this analysis, the population around the sites was projected for 2015. Baseline projections were needed for two of the reactor sites because the population information provided in the proposal was based on 1970 census data. Recent (i.e., 1990) census data were provided for the other proposed site and projected by the offeror to the years 2010 and 2020. From these data points, 2015 projections were interpolated.

4.28.1 Construction Impacts

The proposed reactor sites have indicated that little or no new construction would be needed to support the irradiation of MOX fuel at the sites. Any new construction would be inconsequential. As a result, land use; visual, cultural, and paleontological resources; geology and soils; and site infrastructure would not be affected by any new construction or other activities related to MOX fuel use. Nor would there be any effect on air quality and noise, ecological and water resources, or socioeconomics.

4.28.2 Operational Impacts

4.28.2.1 Air Quality and Noise

Continued operation of the proposed reactor sites would result in a small amount of nonradiological air pollutants being released to the atmosphere mainly due to the requirement to periodically test diesel generators. As shown in Section IV, all of the proposed reactors are operated within Federal, State, and local air quality regulations or guidelines. The estimated air pollutants resulting from operation of the proposed reactors would not be expected to increase due to the use of MOX fuel in these reactors. (See Tables 3.7–1, 3.7–6, and 3.7–11 in Section IV for projected concentrations at the proposed reactor sites.)

There would also not be any increase in the noise levels expected from the operation of these reactors due to the use of MOX fuel.

4.28.2.2 Waste Management

The proposed reactors would be expected to continue to produce low-level waste (LLW), mixed LLW, hazardous waste, and nonhazardous waste as part of their normal operations. The volume of waste generated is not expected to increase as a result of the reactors using MOX fuel. This is consistent with information presented in the *Storage and Disposition PEIS* that stated the use of MOX fuel is not expected to increase the amount or change the content of the waste being generated (DOE 1996b:4-734). (The amount of spent fuel generated would increase somewhat, as discussed in Section 4.28.2.8.)

As shown in Section IV, the estimated LLW generation for each of the proposed reactors is less than the amount estimated in the *Storage and Disposition PEIS* (DOE 1996b:4-734). (See Tables 3.7–2, 3.7–7, and 3.7–12 in Section IV.) None of these waste estimates are expected to impact the proposed reactor sites in terms of their ability to handle these wastes. The wastes would continue to be handled in the same manner as they are today with no change required due to the use of MOX fuel at the reactors.

4.28.2.3 Socioeconomics

The proposed reactor sites would not need to employ any additional workers to support the use of MOX fuel in the reactors. This is consistent with information presented in the *Storage and Disposition PEIS* which concluded that the use of MOX fuel could result in small increases in the worker population at the reactor sites (between 40 and 105), but that any increase would be filled from the area's existing workforce (DOE 1996b:4-727).

4.28.2.4 Human Health Risk From Normal Operation

There should be no change in the radiation dose to the general public from normal operation of the reactors with a partial MOX fuel core versus a full LEU fuel core. This is consistent with findings in the *Storage and Disposition PEIS* that showed a very small range in the expected difference: -1.1×10^{-2} to 2×10^{-2} person-rem

(DOE 1996b:4-729). Therefore, the doses would be approximately the same for either core. The annual estimated radiological releases from normal operation of the proposed reactors to the environment are shown in Table 4.28–3.

Table 4.28–3. Expected Radiological Releases From Continued Operation of the Proposed Reactors (Ci)

Reactor	Atmospheric Releases	Liquid Release	Total Estimated Release
Catawba	349.6	591.4	941.0
McGuire	165.2	626.1	791.3
North Anna	132.5	1,036.0	1,168.5

Table 4.28–4 shows the projected radiological doses that would be received by the maximally exposed offsite individual (MEI) and the general population based on the releases shown in Table 4.28–3. As shown in Table 4.28–4, the average individual living within 80 km (50 mi) of one of the proposed reactor sites could expect to receive an annual dose of between 2.5×10^{-3} to 9.9×10^{-3} mrem/yr from normal operation of these reactors regardless of whether the reactors were using MOX fuel or LEU fuel. This is a small dose compared with the average annual dose an individual would receive from natural background radiation near these sites (about 325 mrem).

Table 4.28–4. Estimated Dose to the Public From Continued Operation of the Proposed Reactors in the Year 2015 (Partial MOX or LEU Core)

Impact	Catawba[a]	McGuire[b]	North Anna[c]	S&D PEIS
Population within 80 km for year 2015				
Dose (person-rem)	5.7	10.7	20.3	2.0
Percent of natural background	7.7×10^{-4}	1.3×10^{-3}	3.0×10^{-3}	2.6×10^{-4}
Latent fatal cancers	2.9×10^{-3}	5.4×10^{-3}	1.0×10^{-2}	1.0×10^{-3}
Maximally exposed individual (mrem/yr)				
Annual dose (mrem)	0.73	0.31	0.37	0.17
Percent of natural background	0.22	0.095	0.11	0.052
Latent fatal cancer risk	3.7×10^{-7}	1.6×10^{-7}	1.9×10^{-7}	8.5×10^{-8}
Average exposed individual within 80 km				
Annual dose (mrem)	2.5×10^{-3}	4.2×10^{-3}	9.9×10^{-3}	7.8×10^{-4}
Latent fatal cancer risk	1.3×10^{-9}	2.1×10^{-9}	4.9×10^{-9}	3.9×10^{-10}

[a] The population for the year 2015 is estimated to be 2,265,000.
[b] The population for the year 2015 is estimated to be 2,575,000.
[c] The population for the year 2015 is estimated to be 2,042,000.
Key: LEU, low-enriched uranium; S&D PEIS, *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement.*

The average radiation worker at the proposed reactor sites could expect to receive an annual dose of between 46 and 123 mrem/yr from normal operations with a partial MOX core. (See Tables 3.7–5, 3.7–10, and 3.7–15 in Section IV.) As discussed in Section IV and Section VI (Appendix P), this is the same amount of radiation dose that would be received if the reactors continued to use only LEU fuel. This is because the MOX fuel would be shipped in safe, secure trailers (SSTs) and moved remotely or in shielded vehicles to the reactor's fuel staging area and finally into and out of the reactor core. The projection that the use of MOX fuel would not change the estimated worker dose is consistent with data presented in the *Storage and Disposition PEIS*, which showed an incremental increase in worker dose of less than 0.1 percent due to the use of MOX fuel (DOE 1996b:4-730).

4.28.2.5 Reactor Accident Analysis

The reactor accident analysis includes an assessment of postulated design basis and beyond-design-basis accidents at each reactor site. The accidents presented were selected because of their potential to release substantial amounts of radioactive material to the environment. A detailed discussion of the accident analysis methodology is provided in Section VI (Appendix K).

There are differences in the expected risk of reactor accidents from the use of MOX fuel. Risk is determined by multiplying two factors. The first factor is the probability or frequency of the accident occurring. In the case of the reactor accidents evaluated in this *Supplement*, no change has been made in the estimated frequency of the accident based on the presence of MOX fuel. The frequencies used in the analysis are the same as those used in each reactor's probabilistic risk assessment (PRA), which was prepared for NRC for the reactor's current LEU core. Although it has been suggested that the frequency of these accidents would be higher with MOX fuel present, no empirical data is available to support this. Further, the National Academy of Sciences has stated that "We believe, further, that under these circumstances no important overall adverse impact of MOX use on the accident probabilities of the LWRs involved will occur; if there are adequate reactivity and thermal margins in the fuel, as licensing review should ensure, the main remaining determinants of accident probabilities will involve factors not related to fuel composition and hence unaffected by the use of MOX rather than LEU fuel" (NAS 1995). The second factor in the risk equation is an estimate of what the consequences would be should the accident occur. Depending on the accident being analyzed, the presence of MOX fuel would decrease or increase the consequences of the accident because it would result in a different amount of radiation being released during the accident due to different isotopics and amounts of radioactive isotopes and noble gases being generated.

The change in consequences to the surrounding population due to the use of MOX fuel is estimated to range from 9.5×10^{-4} fewer to 5.5×10^{-2} additional LCFs for design basis accidents evaluated in this *Supplement*, to 7.5 fewer to 1,600 additional LCFs for beyond-design-basis accidents (14,800 versus 13,200 LCFs in the worst accident). Also, some of the beyond-design-basis accidents could result in prompt fatalities should they occur. The estimated increase in prompt fatalities due to MOX fuel being used during one of these accidents would range from no change to 28 additional fatalities (843 versus 815 prompt fatalities in the worst accident). As a result of these changes in projected consequences, there would be a change in the risk to the public associated with these accidents. The change in risk (in terms of an LCF or prompt fatality) to the surrounding population within 80 km (50 mi) of the proposed reactors is projected to range from a decrease of 6 percent to an increase of 3 percent in the risk of additional LCFs from design basis accidents, and from a decrease of 4 percent to an increase of 15 percent in the risk of additional prompt fatalities and LCFs from beyond-design-basis accidents.

The risk to the MEI would also change with the use of MOX fuel. The change in risk to the MEI of an LCF as a result of using MOX fuel during one of the design basis accidents evaluated is expected to range from a decrease of 10 percent to an increase of 3 percent. The change in risk to the MEI of a prompt fatality or LCF as a result of using MOX fuel during one of the beyond-design-basis accidents evaluated is expected to range from a 1 percent increase to a 22 percent increase. In the most severe accident evaluated, an ISLOCA, it is projected that the MEI would receive a fatal dose of radiation regardless of whether the reactor was using MOX fuel or LEU fuel at all of the proposed sites. It should be noted that the probability or estimated frequency of this accident occurring is very low; an average of 1 chance in 3.2 million per year of reactor operation.

Beyond-design-basis accidents, if they were to occur, would be expected to result in major impacts to the reactors and the surrounding communities and environment regardless of whether the reactor were using an LEU or partial MOX core. However, the probability of a beyond-design-basis accident actually happening is extremely unlikely, so the risk to an individual living within 80 km (50 mi) of the proposed reactors from these accidents is estimated to be low.

The following comments were received on the reactor analysis presented in the SPD Draft EIS and represent different or opposing views. Several comments indicated that the generic reactor analysis, presented in the *Storage and Disposition PEIS* and summarized in the SPD Draft EIS, was inadequate for a decision on the use of MOX fuel in specific reactors. Commentors, including the Blue Ridge Environmental Defense League, the Institute for Energy and Environmental Research, Serious Texans Against Nuclear Dumping, the Nuclear Control Institute, the Nuclear Information and Resource Service, and several individuals, while acknowledging that DOE committed to perform a site-specific reactor analysis in the SPD Final EIS, were concerned that such analysis should be available for public review prior to finalizing the document. Accordingly, the new analysis presented in this *Supplement* was performed using site-specific information and operating characteristics from the six reactors proposed for irradiation services and updated MOX fuel-loading estimates. NRC-accepted models were used to estimate impacts associated with normal operations, design basis, and beyond-design-basis accidents. The methodology used is consistent with DOE and industry practice, as well as the approach advocated by the commentors who requested additional analysis. The results are determined by the methodology and the assumptions. As indicated in this section, DOE's assumptions are based on its current planning, for example, 40 percent MOX cores rather than full cores as used in the *Storage and Disposition PEIS,* as well as site-specific meteorology and population data—all factors that influence the results.

4.28.2.5.1 Design Basis Accident Analysis

Design basis events are not expected to take place, but are postulated because their consequences would include the potential for the release of substantial amounts of radioactive material. They are the most drastic events that must be designed against and represent limiting design cases. The design basis accidents evaluated in this *Supplement* include a large-break loss-of-coolant accident (LOCA) and a fuel-handling accident.

The large-break LOCA is defined as a break equivalent in size to a double-ended rupture of the largest pipe of the reactor coolant system. Following this rupture of a reactor coolant pipe, the emergency core cooling system keeps cladding temperatures well below melting, ensuring that the core remains intact and in a coolable geometry. The increase in cladding temperature and rapid depressurization of the core, however, may cause some cladding failure in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the containment. Although no core melting would occur during this LOCA, a gross release of fission products is evaluated consistent with NRC methodology. For a gross release of fission products to occur, a number of simultaneous and extended failures in the engineered safety feature systems would be required.

The fuel-handling accident is defined as dropping of a spent fuel assembly resulting in breaching of the fuel rod cladding. This breach would release a portion of the volatile fission gases from the damaged fuel rods. Although this fuel-handling accident would realistically result in only a fraction of the fuel rods being damaged, all the fuel rods in the assembly are assumed to be damaged consistent with NRC methodology.

No major increase in estimated impacts would be expected from design basis accidents at the proposed reactor sites due to the use of MOX fuel. In fact, the risk from the postulated fuel-handling accident at all three sites would slightly decrease as a result of using MOX fuel. The fuel-handling accident doses are driven by the noble gases, primarily krypton. The percentage of the dose attributable to krypton is 58 percent at Catawba, 56 percent at McGuire, and 54 percent at North Anna. With the 40 percent MOX core, the MOX/LEU ratios for the krypton isotopes range from 0.78–0.89 indicating that there is less krypton present in a partial MOX core. The combination of the low MOX/LEU ratio and the large percentage of dose contribution associated with krypton results in a lower dose for this accident with a 40 percent MOX core.

The doses to the surrounding population within 80 km (50 mi) from a LOCA are expected to be about 3 percent higher for a partial MOX core versus a full LEU core. The LOCA doses are driven by radioactive isotopes of iodine. The percentage of dose attributable to iodine in a LOCA is approximately 97 percent at each reactor site. Because the iodine MOX/LEU ratios average slightly over one, indicating that there is more iodine present in a partial MOX core, the dose also rises slightly for this accident.

Catawba Design Basis Accident Analysis

Table 4.28–5 presents the results of this analysis for design basis accidents at Catawba. (To derive the increase or decrease in risk associated with the use of MOX fuel at any of the proposed reactors, subtract the risk associated with the full LEU core from the same risk for a partial MOX core for any of the accidents presented in Tables 4.28–5 through 4.28–7 and 4.28–10 through 4.28–12. For example, the risk to the MEI from a LOCA at Catawba, as shown in Table 4.28–5, is calculated by subtracting 8.64×10^{-8} from 8.88×10^{-8} for an increase in risk of 2.4×10^{-9}. All risks have been rounded to two significant figures, so, in cases where the difference is only one digit, the numbers have been extended to two significant figures using model results.)

The results indicate that the highest risk increase to the surrounding population for a design basis accident with a partial MOX core configuration instead of a full LEU core is 3.3 percent from the LOCA. The increased risk from the use of MOX fuel to the noninvolved worker[2] is one fatality every 210 million years (4.8×10^{-9} per 16-year campaign[3]); the MEI, one fatality every 420 million years (2.4×10^{-9} per 16-year campaign); and the general population, one fatality every 100,000 years (6.4×10^{-6} per 16-year campaign).

McGuire Design Basis Accident Analysis

Table 4.28–6 presents the results of this analysis for design basis accidents at McGuire.

The results indicate that the highest risk increase to the surrounding population for a design basis accident with a partial MOX core configuration instead of a full LEU core is approximately 3.0 percent from the LOCA. The increased risk from the use of MOX fuel to the noninvolved worker is one fatality every 69 million years (1.4×10^{-8} per 16-year campaign); the MEI, one fatality every 120 million years (8.0×10^{-9} per 16-year campaign); and the general population, one fatality every 78,000 years (1.3×10^{-5} per 16-year campaign).

[2] During a design-basis accident at a commercial reactor the involved workers are defined for the purposes of this *Supplement* as control room operators. Control rooms at commercial reactors are designed so that during a design basis accident, the doses to control room operators are mitigated by emergency systems. These systems include isolation dampers, emergency ventilation systems, bottled air supplies, and high-efficiency particulate air (HEPA) filtration to lower the doses to control room operators. Control room operator doses are predominantly from noble gases and iodine because the HEPA filtration removes almost all of the particulates. Therefore, the assumption is made that an unprotected noninvolved worker (i.e., all workers except those in the control room at the time of the accident) would most likely receive a larger dose. Because the objective of the analysis is to determine the maximum increased risk from a partial MOX core versus an LEU core, the noninvolved worker was chosen as the onsite receptor.

[3] If MOX fuel is used in the proposed reactors, it is estimated that it will take approximately 16 years to irradiate all of the surplus plutonium currently considered for use in MOX fuel.

NORTH ANNA DESIGN BASIS ACCIDENT ANALYSIS

Table 4.28–7 presents the results of this analysis for design basis accidents at North Anna.

The results indicate that the highest risk increase to the surrounding population for a design basis accident with a partial MOX core configuration instead of a full LEU core is approximately 2.5 percent from the LOCA. The increased risk from the use of MOX fuel to the noninvolved worker is one fatality every 7.8 billion years (1.3×10^{-10} per 16-year campaign); the MEI, one fatality every 31 billion years (3.2×10^{-11} per 16-year campaign); and the general population, one fatality every 6.2 million years (1.6×10^{-7} per 16-year campaign).

Table 4.28–5. Design Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Loss-of-coolant accident	7.50×10^{-6}	LEU	3.78	1.51×10^{-3}	1.81×10^{-7}	1.44	7.20×10^{-4}	8.64×10^{-8}	3.64×10^{3}	1.82	2.19×10^{-4}
		MOX	3.85	1.54×10^{-3}	1.86×10^{-7}	1.48	7.40×10^{-4}	8.88×10^{-8}	3.75×10^{3}	1.88	2.26×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.27	1.10×10^{-4}	1.78×10^{-7}	0.13	6.90×10^{-5}	1.10×10^{-7}	1.12×10^{2}	5.61×10^{-2}	8.98×10^{-5}
		MOX	0.26	1.05×10^{-4}	1.68×10^{-7}	0.13	6.55×10^{-5}	1.05×10^{-7}	1.10×10^{2}	5.48×10^{-2}	8.77×10^{-5}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table 4.28–6. Design Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	1.50×10^{-5}	LEU	5.31	2.12×10^{-3}	5.10×10^{-7}	2.28	1.14×10^{-3}	2.74×10^{-7}	3.37×10^{3}	1.68	4.03×10^{-4}
		MOX	5.46	2.18×10^{-3}	5.25×10^{-7}	2.34	1.17×10^{-3}	2.82×10^{-7}	3.47×10^{3}	1.73	4.16×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.392	1.57×10^{-4}	2.51×10^{-7}	0.212	1.06×10^{-4}	1.70×10^{-7}	99.1	4.96×10^{-2}	7.94×10^{-5}
		MOX	0.373	1.49×10^{-4}	2.38×10^{-7}	0.201	1.01×10^{-4}	1.62×10^{-7}	97.3	4.87×10^{-2}	7.79×10^{-5}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table 4.28–7. Design Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	2.10×10^{-5}	LEU	0.114	4.56×10^{-5}	1.53×10^{-8}	3.18×10^{-2}	1.59×10^{-5}	5.34×10^{-9}	39.4	1.97×10^{-2}	6.62×10^{-6}
		MOX	0.115	4.60×10^{-5}	1.55×10^{-8}	3.20×10^{-2}	1.60×10^{-5}	5.38×10^{-9}	40.3	2.02×10^{-2}	6.78×10^{-6}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.261	1.04×10^{-4}	1.66×10^{-7}	9.54×10^{-2}	4.77×10^{-5}	7.63×10^{-8}	29.4	1.47×10^{-2}	2.35×10^{-5}
		MOX	0.239	9.56×10^{-5}	1.53×10^{-7}	8.61×10^{-2}	4.31×10^{-5}	6.90×10^{-8}	27.5	1.38×10^{-2}	2.21×10^{-5}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

4.28.2.5.2 Beyond-Design-Basis Accident Analysis

Only beyond-design-basis accident scenarios that lead to containment bypass or failure were evaluated because these are the accidents with the greatest potential consequences. The public health and environmental consequences would be significantly less for accident scenarios that do not lead to containment bypass or failure. A steam generator tube rupture, early containment failure, late containment failure, and an ISLOCA were chosen as the representative set of beyond-design-basis accidents.

Commercial reactors, licensed by NRC, are required to complete Individual Plant Examinations (IPEs) to assess plant vulnerabilities to severe accidents. An acceptable method of completing the IPEs is to perform a PRA. A PRA evaluates, in full detail (quantitatively), the consequences of all potential events caused by the operating disturbances (known as internal initiating events) within each plant. The PRA uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident. The PRAs for the proposed reactors provided the required data to evaluate beyond-design-basis accidents.

A beyond-design-basis steam generator tube rupture induced by high temperatures represents a containment bypass event. Analyses have indicated a potential for very high gas temperatures in the reactor coolant system during accidents involving core damage with the primary system at high pressure. The high temperature could fail the steam generator tubes long before the core begins to relocate. As a result of the tube rupture, the secondary (nonradioactive) side may be exposed to high pressure. This pressure would likely cause relief valves to open. If these valves failed to reclose, an open pathway from the vessel to the environment would result.

An early containment failure is defined as the failure of containment prior to or very soon (within a few hours) after breach of the reactor vessel. A variety of mechanisms can cause failure such as direct contact of core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, and fuel-coolant interactions. Early containment failure can be important because it tends to result in shorter warning times for initiating public protective measures and because radionuclide releases would generally be more severe than if the containment were to fail late.

A late containment failure involves failure of the containment several hours after breach of the reactor vessel. A variety of mechanisms can cause late containment failure such as gradual pressure and temperature increase, hydrogen combustion, and basemat melt-through by core debris.

An ISLOCA refers to a class of accidents in which the reactor coolant system pressure boundary interfacing with a supporting system of lower design pressure is breached. If this occurs, the low-pressure system would be overpressurized and could rupture outside the containment. This failure would establish a flow path directly to the environment or, sometimes, to another building of small-pressure capacity.

Each of these accidents has a warning time and a release time associated with it. The warning time is the time at which notification is given to offsite emergency response officials to initiate protective measures for the surrounding population. The release time is when the release to the environment begins. The minimum time between the warning time and the release time is one-half hour; enough time to evacuate onsite personnel. This also conservatively assumes that an onsite emergency has not been declared prior to initiating an offsite notification. Intact containment severe accident scenarios, which were not analyzed because of their insubstantial offsite consequences, take place on an even longer timeframe.

For severe accident scenarios that postulate large abrupt releases, there exists a possibility for prompt fatalities. Prompt fatalities may occur if the radiation dose is sufficiently high. Table 4.28–8 shows the number of prompt fatalities in the offsite population estimated from a postulated beyond-design-basis steam generator tube rupture and ISLOCA. None of the other accidents evaluated in the SPD EIS is expected to result in prompt fatalities.

Table 4.28–8. Estimated Prompt Fatalities in the Public From Beyond-Design-Basis Reactor Accidents

Reactor	LEU Core	Partial MOX Core
Steam generator tube rupture		
Catawba	1	1
McGuire	1	1
North Anna	0	0
Interfacing systems loss-of-coolant accident		
Catawba	815	843
McGuire	398	421
North Anna	54	60

Table 4.28–9 shows the difference in accident consequences for reactors using MOX fuel versus LEU fuel. For beyond-design-basis accidents, the consequences would be expected to be higher, with the largest increase associated with an ISLOCA. This is because the MOX fuel would release a higher actinide inventory in a severe accident. The increased impacts of an ISLOCA range from 10 to 15 percent and are estimated, on average, to be about 13 percent greater to the general population living within 80 km (50 mi) of the reactor with a partial MOX core instead of an LEU core. It should be noted that this accident has a very low estimated frequency of occurrence, an average of 1 chance in 3.2 million per year of reactor operation for the reactors being proposed to irradiate MOX fuel.

Table 4.28–9. Ratio of Accident Impacts for MOX-Fueled and Uranium-Fueled Reactors (MOX Impacts/LEU Impacts)

	Catawba		McGuire		North Anna		S&D PEIS	
Accident	MEI	Population	MEI	Population	MEI	Population	MEI	Population
Design basis accidents								
LOCA[a]	1.03	1.03	1.03	1.03	1.01	1.03	NA	NA
Fuel-handling accident[a]	0.95	0.98	0.95	0.98	0.90	0.94	NA	NA
Beyond-design-basis accidents								
SG tube rupture	1.06	1.04	1.06	1.04	1.16	1.07	0.94	0.94
Early containment failure	1.01	1.05	1.03	1.02	1.10	1.01	0.96	0.97
Late containment failure	1.07	0.96	1.01	0.97	1.03	1.09	1.07	1.08
ISLOCA	1.14	1.12	1.12	1.10	1.22	1.15	0.92	0.93

[a] No design basis accidents were analyzed in the *Storage and Disposition PEIS.*

Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; LOCA, loss-of-coolant accident; MEI, maximally exposed individual; NA, not applicable; S&D PEIS, *Storage and Disposition of Weapons-Usable Fissile Material Final Programmatic Environmental Impact Statement*; SG, steam generator.

CATAWBA BEYOND-DESIGN-BASIS ACCIDENTS

Table 4.28–10 shows the risks of LCFs associated with all of the evaluated Catawba beyond-design-basis accidents.

Table 4.28–10. Beyond-Design-Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
SG tube rupture[e]	6.31×10^{-10}	LEU	3.46×10^{2}	0.346	3.49×10^{-9}	5.71×10^{6}	2.86×10^{3}	2.88×10^{-5}
		MOX	3.67×10^{2}	0.367	3.71×10^{-9}	5.93×10^{6}	2.96×10^{3}	2.99×10^{-5}
Early containment failure	3.42×10^{-8}	LEU	5.97	2.99×10^{-3}	1.63×10^{-9}	7.70×10^{5}	3.85×10^{2}	2.11×10^{-5}
		MOX	6.01	3.01×10^{-3}	1.65×10^{-9}	8.07×10^{5}	4.04×10^{2}	2.21×10^{-4}
Late containment failure	1.21×10^{-5}	LEU	3.25	1.63×10^{-3}	3.15×10^{-7}	3.93×10^{5}	1.96×10^{2}	3.79×10^{-2}
		MOX	3.48	1.74×10^{-3}	3.38×10^{-7}	3.78×10^{5}	1.89×10^{2}	3.66×10^{-2}
ISLOCA	6.90×10^{-8}	LEU	1.40×10^{4}	1	1.10×10^{-6}	2.64×10^{7}	1.32×10^{4}	1.46×10^{-2}
		MOX	1.60×10^{4}	1	1.10×10^{-6}	2.96×10^{7}	1.48×10^{4}	1.63×10^{-2}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire timing and release fractions were used to compare like scenarios.
Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; SG, steam generator.

At Catawba, the greatest increase in risk of LCFs from the use of a partial MOX core to the surrounding population within 80 km (50 mi) for a beyond-design-basis accident is from an ISLOCA. If this accident were to occur, the consequences, in terms of LCFs and prompt fatalities in the general population within 80 km (50 mi), would be approximately 12 percent greater than those from an ISLOCA with an LEU core. It would be expected to result in approximately 14,000 fatalities with an LEU core and 15,600 fatalities with a partial MOX core. The increased risk in terms of an LCF in the surrounding population associated with the use of MOX fuel would be one additional LCF every 570 years or 1.7×10^{-3} per 16-year campaign. The increased risk in terms of a prompt fatality is one additional fatality every 32,000 years or 3.1×10^{-5} per 16-year campaign. No increase in risk to the MEI would be expected due to the severity of this accident. The MEI would be expected to receive a fatal dose regardless of whether the core was partially fueled with MOX fuel or not, so the risk of a fatality is estimated to be the same in either case; 1 in 900,000 years or 1.1×10^{-6} per 16-year campaign.

At Catawba, the highest risk from a beyond-design-basis accident to the surrounding population within 80 km (50 mi) is from a late containment failure regardless of core type. If this accident were to occur with a partial MOX core, the consequences, in terms of LCFs, would be approximately 3.6 percent lower than those from the same accident with an LEU core. This accident would be expected to result in 196 LCFs with an LEU core and 189 LCFs with a partial MOX core. The decreased risk to the population associated with the use of MOX fuel would be one less LCF every 780 years or 1.3×10^{-3} per 16-year campaign. No prompt fatalities would be expected to result from this accident. However, the risk to the MEI would be expected to increase by approximately 6.7 percent if a partial MOX core were being used.[4] The increased risk of an LCF to the MEI from this accident with a partial MOX core is estimated to be one in 45 million years or 2.2×10^{-8} per 16-year campaign.

McGuire Beyond-Design-Basis Accidents

Table 4.28–11 shows the risks of LCFs associated with all of the evaluated McGuire beyond-design-basis accidents.

Table 4.28–11. Beyond-Design-Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
SG tube rupture[e]	5.81×10^{-9}	LEU	6.10×10^{2}	0.610	5.66×10^{-8}	5.08×10^{6}	2.54×10^{3}	$2.37\times10_{-4}$
		MOX	6.47×10^{2}	0.647	6.02×10^{-8}	5.28×10^{6}	2.64×10^{3}	2.45×10^{-4}
Early containment failure	9.89×10^{-8}	LEU	12.2	6.10×10^{-3}	9.65×10^{-9}	7.90×10^{5}	3.95×10^{2}	6.26×10^{-4}
		MOX	12.6	6.30×10^{-3}	9.97×10^{-9}	8.04×10^{5}	4.02×10^{2}	6.37×10^{-4}
Late containment failure	7.21×10^{-6}	LEU	2.18	1.09×10^{-3}	1.26×10^{-7}	3.04×10^{5}	1.52×10^{2}	1.76×10^{-2}
		MOX	2.21	1.11×10^{-3}	1.28×10^{-7}	2.96×10^{5}	1.48×10^{2}	1.71×10^{-2}
ISLOCA	6.35×10^{-7}	LEU	1.95×10^{4}	1	1.02×10^{-5}	1.79×10^{7}	8.93×10^{3}	0.091
		MOX	2.19×10^{4}	1	1.02×10^{-5}	1.97×10^{7}	9.85×10^{3}	0.10

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire timing and release fractions were used to compare like scenarios.
Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; SG, steam generator.

At McGuire, the greatest increase in risk from the use of a partial MOX core and the highest risk regardless of core type to the surrounding population within 80 km (50 mi) for a beyond-design-basis accident is from an ISLOCA. If this accident were to occur, the consequences, in terms of LCFs and prompt fatalities, in the general population within 80 km (50 mi) would be approximately 10 percent greater than those from an ISLOCA with an LEU core. It would be expected to result in approximately 9,300 fatalities with an LEU core

[4] For the late containment failure scenario at Catawba and McGuire, the MEI dose increases while the population dose decreases. The MEI dose increases because 96 percent of the MEI dose is from direct exposure during the initial plume passage. With a 40 percent MOX core, there is approximately double the actinide inventory. Because the actinide isotopes contribute greatly to the inhalation dose, the MEI dose increases. The majority of the population dose (78 percent) is from long-term effects, primarily groundshine. With a 40 percent MOX core, the majority of the fission products decrease, resulting in a lower groundshine dose. Therefore, the population dose decreases.

and 10,300 fatalities with a partial MOX core. The increased risk, in terms of an LCF, in the surrounding population would be one additional LCF every 110 years or 9.3×10^{-3} per 16-year campaign. The increased risk in terms of a prompt fatality would be one additional fatality every 4,300 years or 2.3×10^{-4} per 16-year campaign. For the same reasons as discussed above for Catawba, no increase in risk to the MEI would be expected due to the severity of this accident. The risk to the MEI of a fatality is estimated to be the same in either case, one fatality every 98,000 years or 1.0×10^{-5} per 16-year campaign.

NORTH ANNA BEYOND-DESIGN-BASIS ACCIDENTS

Table 4.28–12 shows the risks of LCFs associated with all of the evaluated North Anna beyond-design-basis accidents.

Table 4.28–12. Beyond-Design-Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts on Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
SG tube rupture[e]	7.38×10^{-6}	LEU	2.09×10^{2}	0.209	2.46×10^{-5}	1.73×10^{6}	8.63×10^{2}	0.102
		MOX	2.43×10^{2}	0.243	2.86×10^{-5}	1.84×10^{6}	9.20×10^{2}	0.109
Early containment failure[e]	1.60×10^{-7}	LEU	19.6	1.96×10^{-2}	5.02×10^{-8}	8.33×10^{5}	4.17×10^{2}	1.07×10^{-3}
		MOX	21.6	2.16×10^{-2}	5.54×10^{-8}	8.42×10^{5}	4.21×10^{2}	1.08×10^{-3}
Late containment failure[e]	2.46×10^{-6}	LEU	1.12	5.60×10^{-4}	2.21×10^{-8}	4.04×10^{4}	20.2	7.95×10^{-4}
		MOX	1.15	5.75×10^{-4}	2.26×10^{-8}	4.43×10^{4}	22.1	8.70×10^{-4}
ISLOCA[e]	2.40×10^{-7}	LEU	1.00×10^{4}	1	3.84×10^{-6}	4.68×10^{6}	2.34×10^{3}	8.99×10^{-3}
		MOX	1.22×10^{4}	1	3.84×10^{-6}	5.41×10^{6}	2.70×10^{3}	1.04×10^{-2}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire release durations and warning times were used in lieu of site specific data.
Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; SG, steam generator.

At North Anna, the greatest increase in risk from the use of a partial MOX core to the surrounding population within 80 km (50 mi) for a beyond-design-basis accident is from an ISLOCA. If this accident were to occur, the consequences, in terms of LCFs and prompt fatalities in the general population within 80 km (50 mi) would be approximately 15 percent greater than those from an ISLOCA with an LEU core. It would be expected to result in approximately 2,400 fatalities with an LEU core and 2,800 fatalities with a partial MOX core. The increased risk, in terms of an LCF, to the surrounding population, would be one additional fatality every 730 years or 1.4×10^{-3} per 16-year campaign. The increased risk in terms of a prompt fatality is one additional fatality every 43,000 years or 2.3×10^{-5} per 16-year campaign. For the same reasons as discussed above for Catawba, no increase in risk to the MEI would be expected due to the severity of this accident. The risk to the

MEI of a fatality is estimated to be the same in either case, one fatality every 260,000 years or 3.8×10^{-6} per 16-year campaign.

At North Anna, the highest risk from a beyond-design-basis accident to the surrounding population within 80 km (50 mi) is from a steam generator tube rupture regardless of core type. If this accident were to occur with a partial MOX core, the consequences, in terms of LCFs, would be approximately 6.6 percent greater than those from the same accident with an LEU core. It would be expected to result in approximately 860 LCFs with an LEU core and 920 LCFs with a partial MOX core. The increased risk, in terms of an LCF, to the surrounding population would be one additional LCF every 150 years or 6.7×10^{-3} per 16-year campaign. No prompt fatalities would be expected to result from this accident. The risk to the MEI would be expected to increase by approximately 16 percent if a partial MOX core were being used. The increased risk to the MEI of a fatal dose from this accident with a partial MOX core is estimated to be 1 in 250,000 years or 4.0×10^{-6} per 16-year campaign.

4.28.2.6 Transportation

Transportation required under the MOX approach would include shipments of MOX fuel from the proposed MOX facility to the proposed reactor sites for irradiation. It is estimated that approximately 830 shipments of fresh MOX fuel would be shipped to the proposed reactor sites in DOE-provided SSTs. While these shipments would likely replace similar shipments of fresh LEU fuel to the reactor sites, thereby reducing the transportation risks associated with this fuel, this *Supplement* analyzes the shipments on a stand-alone basis to estimate the maximum risk to the public. (The shipment of spent fuel is being considered in DOE's EIS for a potential geologic repository that includes in its inventory the MOX fuel that would be generated from the surplus plutonium disposition program.)

The highest dose for these transportation activities would be associated with those alternatives that include locating the MOX facility at Hanford because it is the candidate site farthest from the proposed reactor sites. Similarly, the lowest dose would be associated with alternatives considering placing the MOX facility at SRS because this is the candidate site closest to the proposed reactors.

The estimated dose to the transportation crew from the incident-free transportation activities of fresh MOX fuel to the proposed reactors is estimated to range from 0.036 rem to 0.19 rem depending on the location of the MOX facility. In terms of the number of LCFs in the crew from this transportation, the number would range from 1.4×10^{-5} to 7.8×10^{-5}. The estimated dose to the public from the incident-free transportation of this material is estimated to range from 0.019 rem to 0.092 rem. In terms of the number of LCFs in the public from this transportation, the number would range from 9.3×10^{-6} to 4.6×10^{-5}. The estimated number of LCFs from emissions associated with this transportation would range from 9.0×10^{-4} to 1.4×10^{-2}. Thus, no fatalities would be expected as a result of incident-free transportation of this material.

The number of LCFs expected from transportation accidents is also projected to be small. The estimated dose from accidents involving this MOX fuel is projected to range from 0.15 rem to 0.46 rem. These doses range from 7.5×10^{-5} to 2.3×10^{-4} LCFs in the public. In terms of a traffic fatality from accidents, it is estimated that this transportation would result in between 5.6×10^{-3} and 3.0×10^{-2} traffic fatalities. Thus, no fatalities would be expected as a result of accidents associated with this transportation.

4.28.2.7 Environmental Justice

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse health or environmental effects of their programs, policies, and activities on minority and low-income populations. The Council on Environmental Quality has oversight responsibility for documentation prepared in compliance with the National Environmental Policy Act (NEPA). In December 1997, the Council released guidance on environmental justice under NEPA (CEQ 1997). The Council's guidance was adopted as the basis for the analysis of environmental justice contained in the SPD EIS. This section provides an assessment of the potential for disproportionately high and adverse human health or environmental effects on minority and low-income populations that could result from implementation of the alternatives for the proposed action.

As demonstrated throughout the analyses presented in Section 4.28, normal irradiation of MOX fuel in existing, commercial reactors would pose no significant health risks to the public. As shown in Section 4.28.2.4, the expected number of LCFs would not increase as a result of radiation released during normal operations for the irradiation of this fuel because there would be essentially no increase in radiation received by the general population from the use of MOX fuel.

Some of the reactor accidents would be expected to result in LCFs and prompt fatalities among the general public regardless of whether the reactor was fueled with MOX fuel or LEU fuel. However, it is unlikely that any of these accidents would occur. The consequences associated with use of MOX fuel would range from 7 less fatalities expected from a late containment failure at Catawba to 1,628 additional fatalities from an ISLOCA at Catawba. However, because these accidents have a very small frequency, the risk to the general population only changes by a small amount. The greatest increase in risk to the general population of an LCF from a severe reactor accident using MOX fuel is an increase of 9.3×10^{-3} over the 16-year MOX campaign; the equivalent of one additional fatality every 110 years. The increased risk of a prompt fatality from this accident due to the use of MOX fuel would be 2.3×10^{-4} over the 16-year MOX campaign; the equivalent of one additional fatality every 4,300 years. Thus, the use of MOX fuel in the proposed reactors would pose no significant risks to the general population residing within the area potentially affected by radiological contamination.

As shown in Section 4.28.2.6, no radiological or nonradiological fatalities would be expected to result from the incident-free transportation of MOX fuel to the proposed reactors. Nor would radiological or nonradiological fatalities be expected to result from transportation accidents.

The implementation of the MOX fuel irradiation program at any of the proposed reactor sites would pose no significant risks to the public, nor would implementation of this program pose significant risks to groups within the general public, including the risk of disproportionately high and adverse effects on minority and low-income populations. The population surrounding the North Anna site is projected to have a larger minority population then the national average by 2015 (35.8 percent versus 34 percent) (See Appendix M). However, the increased risk associated with the use of MOX fuel at this site is low. The greatest increase in risk of LCFs is 1.4×10^{-3} over the 16-year MOX campaign for an ISLOCA accident. If this accident were to occur, the increased number of fatalities due to the use of MOX fuel in the general population within 80 km (50 mi) of the North Anna site would be 366, of which 131 would be expected to be from minority populations; approximately 7 fatalities would be considered to be disproportionate versus the national average.

4.28.2.8 Spent Fuel

As shown in Table 4.28–13, it is likely that some additional spent LEU fuel would be generated by using a partial MOX core in the mission reactors. The amount of additional spent nuclear fuel generated is estimated to range from approximately 2 to 16 percent of the total amount of spent fuel that would be generated by the proposed reactors during the time period MOX fuel would be used. The reactor sites intend to manage the spent MOX fuel the same as spent LEU fuel, by storing it in the reactor's spent fuel pool or placing it in dry storage. The amount of additional spent fuel is not expected to impact spent fuel management at the reactor sites.

Table 4.28–13. Total Additional Spent Fuel Assemblies Generated by MOX Fuel Irradiation

Reactor	Number of Spent Fuel Assemblies Generated With No MOX Fuel	Number of Additional Spent Fuel Assemblies With MOX Fuel	Percent Increase
Catawba 1	672	12	1.8
Catawba 2	672	12	1.8
McGuire 1	756	12	1.6
McGuire 2	672	12	1.8
North Anna 1	420	67	16.0
North Anna 2	540	84	15.6
Total	3,732	199	5.3

For the four units at Catawba and McGuire, all of the additional spent nuclear fuel assemblies would be generated during the transition cycles from LEU to MOX fuel. Additional assemblies help to maintain peaking below design and regulatory limits, and compensate for the greater end-of-cycle reactivity. For Catawba and McGuire, once equilibrium is reached in the partial MOX core, additional fuel assemblies would not be required.

Like McGuire and Catawba, the North Anna units are expected to require additional LEU assemblies during the first transition cores. However, additional assemblies will also be required during equilibrium cycles because of operational considerations of the smaller North Anna cores (157 fuel assemblies compared to 193 each for the McGuire and Catawba units).

As core designs are finalized and optimized for MOX fuel, it may be possible to reduce MOX fuel assembly peaking and thereby reduce the number of additional assemblies required (and spent fuel generated) at the proposed reactors. As it currently stands, the North Anna site could generate approximately 16 percent more spent fuel by using MOX fuel than if the plants continued to use LEU fuel. The total amount of additional spent fuel generated by all six proposed reactors is estimated to be approximately 92 t (101 tons) of heavy metal. However, such MOX fuel is included in the inventory for the potential Nuclear Waste Policy Act geologic repository being studied by DOE. As discussed earlier, DOE is in the process of completing an EIS for a potential geologic repository.

4.28.2.9 Geology and Soils

No ground-disturbing activities related exclusively to the use of MOX fuel are proposed at any of the reactor sites. Therefore, there would be no impact on the reactor site's geology or soils resulting from the use of MOX fuel.

4.28.2.10 Water Resources

There would be no change in water usage or discharge of nonradiological pollutants resulting from use of MOX fuel in the proposed reactors. Each of the reactor sites discharges nonradiological wastewater in accordance with a National Pollutant Discharge Elimination System permit, or an analogous State-issued permit. Permitted outfalls discharge conventional and priority pollutants from the reactor and ancillary processes that are similar to discharges from most reactor sites. Monitoring, analyses, and toxicity testing are also consistent with the types of discharges. Discharge Monitoring Reports for North Anna (May 1994 through April 1998) and Catawba (calendar years 1995 through 1997) showed that, for the most part, there were only occasional noncompliances with permit limitations, only one of which occurred at an outfall receiving reactor process discharges. The effluent from outfall 001 at Catawba failed a quarterly chronic toxicity test in March 1996. However, a followup sample collected after receiving these results passed the test. During the period reviewed, Catawba experienced four noncompliances, two in 1995 and two in early 1996. North Anna exceeded the chlorine limitation at its sewage treatment facility, but this would neither affect, nor be affected by, the use of MOX fuel.

4.28.2.11 Ecological Resources

The use of MOX fuel in existing reactors would not be expected to result in any impacts on ecological resources at the proposed sites. There would be no new construction, and emissions of effluents from the reactors would not change significantly.

4.28.2.12 Cultural and Paleontological Resources

No ground-disturbing activities are proposed at the sites related exclusively to the use of MOX fuel. Therefore, the use of MOX fuel in existing reactors is not expected to affect cultural and paleontological resources at the proposed sites. Similarly, no impacts on Native American resources in the areas surrounding the reactor sites are expected.

4.28.2.13 Land Use

The proposed reactor sites would not require any additional land to support the use of MOX fuel in their reactors. This statement is consistent with information presented in the *Storage and Disposition PEIS* (DOE 1996b:4-720). Nor would the use of MOX fuel in an existing reactor affect the use of other onsite lands (e.g., buffer zones and undeveloped land areas would not be impacted). Prime farmland would not be affected and, because the use of MOX fuel would not result in an in-migration of workers, as discussed in Section 4.28.2.3, no indirect impacts on offsite lands would be expected.

4.28.2.14 Infrastructure

Existing site infrastructure would continue to serve the sites proposed to irradiate MOX fuel. Each site is equipped with water and an existing power distribution system that would adequately support the demands of the reactors should MOX fuel be used. Therefore, the proposed reactor sites would not require any additional infrastructure to support the use of MOX fuel in the reactors. This is consistent with information presented in the *Storage and Disposition PEIS* (DOE 1996b:4-721).

REFERENCES

CEQ (Council on Environmental Quality), 1997, *Environmental Justice Guidance Under the National Environmental Policy Act*, Executive Office of the President, Washington, DC, December 10.

DOE (U.S. Department of Energy), 1996a, *Nuclear Power Generation and Fuel Cycle Report 1996*, DOE/EIA-0436(96), Energy Information Administration, Washington, DC, October.

DOE (U.S. Department of Energy), 1996b, *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement*, DOE/EIS-0229, Office of Fissile Materials Disposition, Washington, DC, December.

DOE (U.S. Department of Energy), 1998, *Surplus Plutonium Disposition Draft Environmental Impact Statement*, DOE/EIS-0283-D, Office of Fissile Materials Disposition, Washington, DC, July.

NAS (National Academy of Sciences and National Research Council), 1995, *Management and Disposition of Excess Weapons Plutonium, Reactor-Related Options*, National Academy Press, Washington, DC.

NRC (U.S. Nuclear Regulatory Commission), 1997a, *SALP Report on Catawba 1 & 2*, http://www.nrc.gov/RIII/rjs2/salp/413salp.htm, Washington, DC, June 4.

NRC (U.S. Nuclear Regulatory Commission), 1997b, *SALP Report on McGuire*, http://www.nrc.gov/RIII/rjs2/salp/salp/369salp.htm, Washington, DC, April 18.

NRC (U.S. Nuclear Regulatory Commission), 1997c, *SALP Report on North Anna*, http://www.nrc.gov/RIII/rjs2/salp/338salp.htm, Washington, DC, February 21.

NRC (U.S. Nuclear Regulatory Commission), 1998, *Evaluating Reactor Licensees*, http://www.nrc.gov/OPA/gmo/tip/process.htm. Washington, DC, September.

VI. Appendixes

The following appendixes are additions to existing appendixes in the Surplus Plutonium Disposition Draft Environmental Impact Statement *(SPD Draft EIS), or in the case of Appendix P,* Environmental Synopsis of Information Provided in Response to the Request for Proposals for MOX Fuel Fabrication and Reactor Irradiation Services, *an entirely new appendix that will be added to the SPD Final EIS.*

Appendix A of this Supplement, Federal Register Notices, *includes a copy of the Notice of Intent that was published in the Federal Register regarding the decision to prepare this Supplement. It will be added to the other Federal Register Notices included in the SPD Draft EIS and reprinted in the SPD Final EIS later this year.*

Appendix K of this Supplement, Reactor Accidents, *will be appended to the existing Appendix K from the SPD Draft EIS and reprinted in the SPD Final EIS. It discusses the methodology used to analyze the reactor accidents evaluated in this Supplement for the Catawba, McGuire, and North Anna reactor sites.*

Appendix M of this Supplement, Analysis of Environmental Justice, *will be appended to the existing Appendix M from the SPD Draft EIS and reprinted in the SPD Final EIS. It discusses the minority and low-income populations surrounding the reactor sites being evaluated in this Supplement.*

Appendix P of this Supplement, Environmental Synopsis of Information Provided in Response to the Request for Proposals for MOX Fuel Fabrication and Reactor Irradiation Services, *is an entirely new appendix that will be added to the SPD Final EIS. It was developed in accordance with DOE's National Environmental Policy Act guidance in 10 CFR 1021.216, and is based on information provided by the winning bidder for operating the proposed mixed oxide (MOX) fuel fabrication facility and irradiating MOX fuel should the decision be made in the SPD EIS Record of Decision to go forward with the MOX approach.*

ajpa.wpd

Appendix A
Federal Register Notices

A.1 NOTICE OF INTENT - SUPPLEMENT TO THE DRAFT SURPLUS PLUTONIUM DISPOSITION ENVIRONMENTAL IMPACT STATEMENT

Dated: March 30, 1999.

Judith Johnson,

Acting Assistant Secretary, Elementary and Secondary Education.

[FR Doc. 99-8394 Filed 4-5-99; 8:45 am]

BILLING CODE 4000-01-P

DEPARTMENT OF ENERGY

Office of Arms Control and Nonproliferation Policy; Proposed Subsequent Arrangement

AGENCY: Department of Energy.

ACTION: Subsequent arrangement.

SUMMARY: This notice is being issued under the authority of Section 131 of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2160). The Department is providing notice of a "subsequent arrangement" under the Agreement for Cooperation in the Peaceful Uses of Nuclear Energy Between the United States of America and the European Atomic Energy Community (EURATOM) and the Agreement for Cooperation Between the Government of the United States of America and the Government of Canada Concerning the Civil Uses of Atomic Energy.

This subsequent arrangement concerns the transfer of 90,552,300 grams of natural uranium in the form of hexafluoride from Cameco Corporation in Canada to Urenco Limited in the United Kingdom for toll enrichment. The enrichment will not exceed 20%. The material will then be transferred to Northern States Power in Minneapolis, MN for use in their commercial power reactor.

In accordance with Section 131 of the Atomic Energy Act of 1954, as amended, we have determined that this subsequent arrangement will not be inimical to the common defense and security.

This subsequent arrangement will take effect no sooner than fifteen days after the date of publication of this notice.

Dated: March 30, 1999.

For the Department of Energy.

Edward T. Fei,

Deputy Director, International Policy and Analysis Division, Office of Arms Control and Nonproliferation.

[FR Doc. 99-8451 Filed 4-5-99; 8:45 am]

BILLING CODE 6450-01-P

DEPARTMENT OF ENERGY

Office of Arms Control and Nonproliferation Policy; Proposed Subsequent Arrangement

AGENCY: Department of Energy.

ACTION: Subsequent Arrangement.

SUMMARY: This notice is being issued under the authority of Section 131 of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2160). The Department is providing notice of a "subsequent arrangement" under the Agreement for Cooperation in the Peaceful Uses of Nuclear Energy Between the United States of America and the European Atomic Energy Community (EURATOM) and the Agreement for Cooperation Between the Government of the United States of America and the Government of Canada Concerning the Civil Uses of Atomic Energy.

This subsequent arrangement concerns the transfer of 3,078,600 grams of natural uranium in the form of hexafluoride from Cameco Corporation in Canada to Urenco Limited in the United Kingdom for toll enrichment. The enrichment will not exceed 20%. The material will then be transferred to Wolf Creek Nulcear Operation Corporation in Burlington, KS for use in their commercial power reactor.

In accordance with Section 131 of the Atomic Energy Act of 1954, as amended, we have determined that this subsequent arrangement will not be inimical to the common defense and security.

This subsequent arrangement will take effect no sooner than fifteen days after the date of publication of this notice.

Dated: March 30, 1999.

For the Department of Energy.

Edward T. Fei,

Deputy Director, International Policy and Analysis Division Office of Arms Control and Nonproliferation.

[FR Doc. 99-8452 Filed 4-5-99; 8:45 am]

BILLING CODE 6450-01-P

DEPARTMENT OF ENERGY

Supplement to the Draft Surplus Plutonium Disposition Environmental Impact Statement

AGENCY: Department of Energy.

ACTION: Notice of Intent.

SUMMARY: The Department of Energy (DOE) announces its intent to prepare a supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement (SPD EIS) pursuant to the National Environmental Policy Act (NEPA). The SPD Draft EIS (DOE/EIS-0283D) was issued for public comment in July 1998. The Supplement will update the SPD Draft EIS by examining the potential environmental impacts of using mixed oxide (MOX) fuel in six specific commercial nuclear reactors at three sites for the disposition of surplus weapons-grade plutonium. DOE identified these reactors through a competitive procurement process. The Department is planning to issue the Supplement to the SPD Draft EIS in April 1999. DOE will publish a separate Notice of Availability in the **Federal Register** at that time. This Notice of Intent describes the content of the Supplement to the SPD Draft EIS, solicits public comment on the Supplement, and announces DOE's intention to conduct a public hearing. Consistent with 40 CFR 1502.9(c)(4) and 10 CFR 1021.314(d), DOE has determined not to conduct scoping for the Supplement.

ADDRESSES: Requests for information concerning the plutonium disposition program can be submitted by calling (answering machine) or faxing them to the toll free number 1-800-820-5156, or by mailing them to: Bert Stevenson, NEPA Compliance Officer, Office of Fissile Materials Disposition, U.S. Department of Energy, Post Office Box 23786, Washington, DC 20026-3786.

FOR FURTHER INFORMATION CONTACT: For general information on the DOE NEPA process, please contact: Carol Borgstrom, Director, Office of NEPA Policy and Assistance, U.S. Department of Energy, 1000 Independence Avenue, S.W., Washington, DC 20585, 202-586-4600 or leave a message at 1-800-472-2756.

Additional information regarding the DOE NEPA process and activities is available on the Internet through the NEPA Home Page at http://www.eh.doe.gov/nepa.

SUPPLEMENTARY INFORMATION:

Background

In October 1994, the Secretary of Energy and the Congress created the Office of Fissile Materials Disposition (MD) within the Department of Energy (DOE) to focus on the elimination of surplus highly enriched uranium (HEU) and plutonium surplus to national defense needs. As one of its major responsibilities, MD is tasked with determining how to disposition surplus weapons—usable plutonium. In January 1997, DOE issued a Record of Decision (ROD) for the Storage and Disposition of Weapons—Usable Fissile Materials Final Programmatic Environmental Impact Statement (S&D PEIS) (DOE/EIS-

0229; December 1996). In that ROD, DOE decided to pursue a strategy that would allow for the possibility of both the immobilization of surplus plutonium and the use of surplus plutonium as mixed oxide (MOX) fuel in existing domestic, commercial reactors. DOE is in the process of completing the Surplus Plutonium Disposition Environmental Impact Statement (SPD Draft EIS) (DOE/EIS–0283D; July 1998) to choose a site(s) for plutonium disposition activities and to determine the technology(ies) that will be used to support this effort.

Related Procurement Action

To support the timely undertaking of the surplus plutonium disposition program, DOE initiated a procurement action to contract for MOX fuel fabrication and reactor irradiation services. The services requested in this procurement process include design, licensing, construction, operation, and eventual deactivation of a MOX facility, as well as irradiation of the MOX fuel in three to eight existing domestic, commercial reactors, should the decision be made by DOE to go forward with the MOX program.

On May 19, 1998, DOE issued a Request for Proposal (RFP) (Solicitation Number DE–RP02–98CH10888) that defined limited activities that may be performed prior to issuance of the SPD EIS ROD. These activities include non-site-specific work primarily associated with the development of the initial conceptual design for the fuel fabrication facility, and plans (paper studies) for outreach, long lead-time procurements, regulatory management, facility quality assurance, safeguards, security, fuel qualifications, and deactivation. No construction would be started on a MOX fuel fabrication facility until the SPD EIS ROD is issued. The MOX facility, if built, would be DOE-owned, licensed by the Nuclear Regulatory Commission, and located at one of four candidate DOE sites. DOE has designated the Savannah River Site as the preferred alternative for the MOX fuel fabrication facility.

Based on a review of proposals received in response to the RFP, DOE determined in January 1999 that one proposal was in the competitive range. Under this proposal, MOX fuel would be fabricated at a DOE site and then irradiated in one of six domestic commercial nuclear reactors.

Environmental Review During Procurement Action

An environmental critique was prepared in accordance with DOE's National Environmental Policy Act (NEPA) regulations at 10 CFR 1021.216. Because an EIS is in progress on this action, DOE required offerors to submit reasonably available environmental data and analyses as a part of their proposals. DOE independently evaluated and verified the accuracy of the data provided by the offeror in the competitive range, and prepared an environmental critique for consideration before the selection was made. The Environmental Critique was used by DOE to determine:

(1) if there are any important environmental issues in the offeror's proposal that may affect the selection process; and

(2) if the potential environmental impacts of the offeror's proposal were bounded by impacts presented in the S&D PEIS and SPD Draft EIS or whether additional analysis was required in the SPD Final EIS.

As required by Section 216, the Environmental Critique included a discussion of the purpose of the procurement; the salient characteristics of the offeror's proposal; any licenses, permits or approvals needed to support the program; and an evaluation of the potential environmental impacts of the offer. The Environmental Critique is a procurement-sensitive document and subject to all associated restrictions. DOE then prepared a synopsis, which summarizes the Environmental Critique and reduces business-sensitive information to a level that will not compromise the procurement process. The Synopsis will be filed with the Environmental Protection Agency and made available to the public.

Contract Award

As a result of the procurement process described above, in March 1999, the Department of Energy contracted with Duke Engineering & Services, COGEMA, Inc., and Stone & Webster to provide mixed oxide fuel fabrication and reactor irradiation services. The team, known as DUKE COGEMA STONE & WEBSTER or DCS, has its corporate headquarters in Charlotte, NC. Subcontractors to DCS include Duke Power Company, Charlotte, NC and Virginia Power Company, Richmond, VA, who will provide the reactor facilities in which mixed oxide fuel will be used upon receipt of Nuclear Regulatory Commission license amendments. Other major subcontractors include Nuclear Fuel Services, Inc., Erwin, TN; Belgonucleaire, Brussels, Belgium; and Framatome Cogema Fuels of Lynchburg, VA. Under the contract, the team will also modify six existing U.S. commercial light water reactors at three sites to irradiate mixed oxide fuel assemblies. These reactors sites are Catawba in York, SC; McGuire in Huntersville, NC; and North Anna in Mineral, VA. The team will be responsible for obtaining a license to operate the fuel fabrication facility and the license modifications for the reactors from the Nuclear Regulatory Commission. Full execution of this contract is contingent on DOE's completion of the SPD EIS, as provided by 40 CFR 1021.216(i).

Supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement

The purpose of the Supplement to the SPD Draft EIS is to update the Draft by including specific information available as a result of the award of the DCS contract. The Supplement to the SPD Draft EIS will contain background information on the SPD Draft EIS; changes made to the SPD Draft EIS (Section 1.7.2); a description of the reactor sites (Section 3.7); impacts of irradiating mixed oxide fuel in existing light water reactors (Section 4.28); Facility Accidents (Appendix K); Analysis of Environmental Justice (Appendix M); and the Environmental Synopsis (Appendix O).

DOE anticipates that the Supplement to the SPD Draft EIS will be available in April. DOE intends to hold an interactive hearing in Washington, DC in May 1999 to discuss issues and receive oral and written comments on the Supplement to the Draft SPD EIS. The Notice of Availability will provide specific information concerning the date, time and location for the public hearing.

Issued in Washington, DC this 31st day of March 1999, for the United States Department of Energy.

David Michaels,

Assistant Secretary, Environment, Safety and Health.

[FR Doc. 99–8455 Filed 4–5–99; 8:45 am]

BILLING CODE 6450–01–P

DEPARTMENT OF ENERGY

Office of Science; Biological and Environmental Research Advisory Committee

AGENCY: Department of Energy.

ACTION: Notice of open meeting.

SUMMARY: This notice announces a meeting of the Biological and Environmental Research Advisory Committee. Federal Advisory Committee Act (Public Law 92–463, 86 Stat. 770) requires that public notice of

Appendix K
Facility Accidents

K.1 COMMERCIAL REACTOR ACCIDENT ANALYSIS

K.1.1 Introduction

Postulated design basis and beyond-design-basis accidents were analyzed using the Melcor Accident Consequence Code System (MACCS2) computer code (NRC 1990, SNL 1997) for each of the three proposed reactor sites, Catawba Nuclear Station, McGuire Nuclear Station, and North Anna Power Station. Only those accidents with the potential for substantial radiological releases to the environment were evaluated. Two design basis accidents, a loss-of-coolant accident (LOCA) and a fuel-handling accident; and four beyond-design-basis accidents, a steam generator tube rupture, an early containment failure, a late containment failure, and an interfacing systems loss-of-coolant accident (ISLOCA) meet this criteria. Each of these accidents was analyzed twice, once using the current low-enriched uranium (LEU) core, and again, assuming a partial (40 percent) mixed oxide (MOX) core. Doses (consequences) and risks to a noninvolved worker, the offsite maximally exposed individual (MEI), and the general public within 80 km (50 mi) of each plant from each accident scenario were calculated. These results were then compared, by plant, for each postulated accident.

The MEI dose is calculated at the exclusion area boundary of each plant. The exclusion area boundary is that area surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided any one of these is not so close to the facility that it interferes with normal operations of the facility and appropriate and effective arrangements are made to control traffic and protect public health and safety on the highway, railroad, or waterway in an emergency. There are generally no residences within an exclusion area. However, if there were residents, they would be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety would result.

K.1.2 Reactor Accident Identification and Quantification

Catawba and McGuire are similar plants, both with two 3,411 MWt Westinghouse pressurized water reactors (PWRs) with ice condenser containments. Because of these similarities, the release paths and mitigating mechanisms for the two plants are almost identical. The conservative assumptions of the U.S. Nuclear Regulatory Commission (NRC) regulatory guidance produce identical radiological releases to the environment (source terms) for the two plants. However, site-specific population and meteorological inputs result in different consequences from the two plants. The North Anna site has two 2,893 MWt Westinghouse PWRs with subatmospheric containments.

Both the design basis and beyond-design-basis accidents were identified from plant documents. Design basis accidents were selected by reviewing the Updated Final Safety Analysis Report (UFSAR) for each plant (DPC 1996, 1997; VPC 1998). Beyond-design-basis accidents were identified from the submittals (DPC 1991, 1992; VPC 1992) in response to the NRC's Generic Letter 88-20 (NRC 1988), which required reactor licensees to perform Individual Plant Examinations (IPEs) for severe accident vulnerabilities. Source terms for each accident for LEU-only cores were identified from these documents, source terms for partial MOX cores were developed based on these LEU source terms, and analyses were performed assuming both the current LEU-only cores and partial MOX cores containing 40 percent MOX fuel and 60 percent LEU fuel. After the source term is developed, the consequences (in terms of latent cancer fatalities [LCFs] and prompt fatalities) can be determined. To determine the risk, however, the frequency (probability) of occurrence of the accident must be determined. Then the consequences are multiplied by the frequency to determine the risk.

For this analysis, the frequencies of occurrence for the accidents with a 40 percent MOX core are assumed to be the same as those with an LEU core. The National Academy of Sciences reported (NAS 1995) that "any approach to the use of MOX fuel in U.S. power reactors must and will receive a thorough, formal safety review before it is licensed. While we are not in a position to predict what if any modifications to existing reactor types will be required as a result of such licensing reviews, we expect that the final outcome will be certification that whatever LWR type is chosen will be able, with modifications if appropriate, to operate within prevailing reactivity and thermal margins using sufficient plutonium loadings to accomplish the disposition mission in a small number of reactors. We believe, further, that under these circumstances no important overall adverse impact of MOX use ion the accident probabilities of the LWRs involved will occur; if there are adequate reactivity and thermal margins in the fuel, as licensing review should ensure, the main remaining determinants of accident probabilities will involve factors not related to fuel composition and hence unaffected by the use of MOX rather than LEU fuel." Considering the National Academy of Sciences statements, the lack of empirical data, and the degree of uncertainty associated with accident frequencies, this analysis assumes that the accident frequencies are the same for a 40 percent MOX core as those for a 100 percent LEU core.

K.1.2.1 MOX Source Term Development

MOX source terms were developed by applying the calculated ratio for individual radioisotopes present in both the MOX and LEU cores to the source term for each of the LEU accidents. MOX source term development required several steps. The analysis assumes that the initial isotopic composition of the plutonium is that delivered to the MOX facility for fabrication into MOX fuel. The MOX facility includes a polishing step that removes impurities, including americium 241, a major contributor to the dose from plutonium 235. This analysis conservatively assumes that the polishing step reduces the americium 241 to 1 part per million (ppm), then ages the plutonium for 1 year after polishing prior to being loaded into a reactor. Table K–1 provides the assumed isotopic composition for the plutonium source material.

Table K–1. Isotopic Breakdown of Plutonium

Isotope	Prior to Polishing (wt %)	After Polishing and Aging (wt %)
Plutonium 236	<1 ppb	1 ppb
Plutonium 238	0.03	0.03
Plutonium 239	92.2	93.28
Plutonium 240	6.46	6.54
Plutonium 241	0.05	0.05
Plutonium 242	0.1	0.1
Americium 241	0.9	25 ppm

Key: ppb, parts per billion; ppm, parts per million; wt %, weight percent.

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) assumes that MOX fuel would be fabricated using depleted uranium (0.25 weight percent uranium 235) (White 1997). The MOX assemblies are assumed to be 4.37 percent plutonium/americium and the LEU assemblies are assumed to be 4.37 percent uranium 235. To simulate a normal plant refueling cycle, the MOX portion was assumed to be 50 percent once-burned and 50 percent twice-burned assemblies. The LEU portion of the MOX was assumed to be 33.3 percent once-burned, 33.3 percent twice-burned, and 33.3 percent thrice-burned assemblies. The LEU-only cores were assumed to be equally divided between once-, twice-, and thrice-burned assemblies. All analyses assumed end-of-cycle inventories to produce the highest consequences. Fuel cycles were based on an 18-month refueling schedule with a 40-day downtime between cycles. The source terms for the LEU-only accident analyses were those identified in plant documents. Source terms for the partial MOX cores were developed using the isotopic ratios in Table K–2 provided by Oak Ridge National Laboratory (ORNL 1999). The MOX core inventory for

Table K–2. MOX/LEU Core Inventory Isotopic Ratios

Isotope	Ratio	Isotope	Ratio	Isotope	Ratio
Americium 241	2.06	Krypton 85m	0.86	Strontium 91	0.86
Antimony 127	1.15	Krypton 87	0.85	Strontium 92	0.89
Antimony 129	1.07	Krypton 88	0.84	Technetium 99m	0.99
Barium 139	0.97	Lanthanum 140	0.97	Tellurium 127	1.16
Barium 140	0.98	Lanthanum 141	0.97	Tellurium 127m	1.20
Cerium 141	0.98	Lanthanum 142	0.97	Tellurium 129	1.08
Cerium 143	0.95	Molybdenum 99	0.99	Tellurium 129m	1.09
Cerium 144	0.91	Neodymium 147	0.98	Tellurium 131m	1.11
Cesium 134	0.85	Neptunium 239	0.99	Tellurium 132	1.01
Cesium 136	1.09	Niobium 95	0.94	Xenon 131m	1.02
Cesium 137	0.91	Plutonium 238	0.76	Xenon 133	1.00
Cobalt 58	0.86	Plutonium 239	2.06	Xenon 133m	1.01
Cobalt 60	0.72	Plutonium 240	2.20	Xenon 135	1.28
Curium 242	1.43	Plutonium 241	1.79	Xenon 135m	1.04
Curium 244	0.94	Praseodymium 143	0.95	Xenon 138	0.96
Iodine 131	1.03	Rhodium 105	1.19	Yttrium 90	0.76
Iodine 132	1.02	Rubidium 86	0.77	Yttrium 91	0.85
Iodine 133	1.00	Ruthenium 103	1.11	Yttrium 92	0.89
Iodine 134	0.98	Ruthenium 105	1.18	Yttrium 93	0.91
Iodine 135	1.00	Ruthenium 106	1.28	Zirconium 95	0.94
Krypton 83m	0.89	Strontium 89	0.83	Zirconium 97	0.98
Krypton 85	0.78	Strontium 90	0.75		

each isotope was divided by the LEU core inventory for that isotope to provide a MOX/LEU ratio for each isotope. These ratios were then applied to LEU releases for each accident to estimate the MOX releases.

The NRC licensing process will thoroughly review precise enrichments and fuel management schemes. The enrichments and fuel management schemes analyzed in the SPD EIS were chosen as realistic upper bounds. The accidents also assumed a maximum 40 percent MOX core. Taken together, these assumptions are sufficiently conservative to account for uncertainties associated with the MOX/LEU ratios.

K.1.2.2 Meteorological Data

Meteorological data for each specific reactor site were used. The meteorological data characteristic of the site region are described by 1 year of hourly data (8,760 measurements). This data includes wind speed, wind direction, atmospheric stability, and rainfall (DOE 1999).

K.1.2.3 Population Data

The population distribution around each plant was determined using 1990 Census data extrapolated to the year 2015. The population was then split into segments which correspond to the chosen polar coordinate grid. The polar coordinate grid for this analysis consists of 12 radial intervals aligned with the 16 compass directions. For Catawba and McGuire, the distances (in kilometers) of the 12 radial intervals are: 0.64, 0.762, 1.61, 3.22, 4.83, 6.44, 8.05, 16.09, 32.18, 48.27, 64.36, 80.45. For North Anna, these distances (in kilometers) are: 0.64, 1.350, 1.61, 3.22, 4.83, 6.44, 8.05, 16.09, 32.18, 48.27, 64.36, 80.45. The first of the 12 segments represents the location of the noninvolved worker and the second is the location of the site boundary. Projected population data for the year 2015 corresponding to the grid segments at Catawba, McGuire, and North Anna are presented in Tables K–3, K–4, and K–5, respectively.

Table K–3. Projected Catawba Population for Year 2015

Direction	Distance in Kilometers From Release Point											
	0.64	**0.762**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	6	14	73	469	800	2,642	51,540	31,112	49,551	33,306
NNE	0	0	6	112	250	334	362	9,394	173,036	135,229	102,558	66,298
NE	0	0	7	119	239	394	595	6,442	212,814	143,650	22,571	20,108
ENE	0	0	11	81	504	1,409	1,042	5,842	72,488	52,784	32,588	10,919
E	0	0	21	5	863	1,059	570	7,959	12,144	27,800	22,844	10,995
ESE	0	0	23	47	295	388	679	7,449	8,607	18,196	12,293	9,290
SE	0	0	20	25	284	893	1,060	37,300	14,279	14,657	12,776	3,692
SSE	0	0	6	80	278	706	891	16,458	10,249	4,190	1,599	11,376
S	0	0	24	165	275	606	819	4,529	4,457	15,062	1,579	1,874
SSW	0	0	17	137	245	238	346	2,268	3,563	2,093	12,970	4,245
SW	0	0	20	114	162	208	267	5,538	9,559	2,040	11,272	12,302
WSW	0	0	21	84	159	205	257	2,493	4,756	8,947	31,712	80,518
W	0	0	23	113	202	272	345	4,979	6,978	17,182	26,070	35,091
WNW	0	0	23	103	199	283	363	3,011	17,814	32,751	29,031	8,706
NW	0	0	23	96	165	274	363	3,099	65,856	28,474	33,819	45,793
NNW	0	0	21	85	125	1,153	1,296	3,404	48,431	24,219	32,537	52,530

Table K–4. Projected McGuire Population for Year 2015

Direction	Distance in Kilometers From Release Point											
	0.64	**0.762**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	44	0	269	110	203	3,153	14,870	28,254	12,987	15,726
NNE	0	0	28	0	124	569	1,728	9,493	21,903	12,317	24,826	43,937
NE	0	0	30	0	5	832	1,016	6,944	30,939	44,064	55,186	44,691
ENE	0	0	184	144	405	684	591	4,289	51,928	37,373	13,039	28,160
E	0	0	217	180	448	381	493	7,575	26,495	21,992	16,957	14,635
ESE	0	0	65	69	271	381	507	7,423	119,345	79,039	36,221	26,552
SE	0	0	15	59	130	244	273	8,387	219,183	204,614	46,100	24,527
SSE	0	0	15	59	99	138	100	9,530	90,900	95,688	79,859	15,954
S	0	0	14	83	165	182	165	6,429	35,178	21,241	41,638	9,071
SSW	0	0	18	101	169	240	221	3,261	61,514	29,814	10,774	9,327
SW	0	0	26	101	169	236	305	5,338	20,195	31,064	47,641	43,067
WSW	0	0	19	101	169	236	296	2,741	20,873	17,334	15,815	15,077
W	6	0	14	112	184	252	312	2,048	24,932	11,715	12,705	43,357
WNW	0	0	3	101	444	811	338	2,187	14,985	57,262	74,708	60,953
NW	0	0	0	224	200	1,005	793	4,260	8,528	22,380	26,093	12,511
NNW	0	0	0	0	4	0	36	1,989	8,570	40,993	13,101	10,686

Table K–5. Projected North Anna Population for Year 2015

Direction	Distance in Kilometers From Release Point											
	0.64	**1.35**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	0	39	98	122	153	576	7,816	5,149	17,803	42,233
NNE	0	0	2	37	58	160	206	1,236	7,634	10,765	25,976	172,658
NE	0	0	2	30	43	94	100	1,122	38,833	90,820	34,429	77,097
ENE	0	0	0	15	103	40	64	1,373	5,822	6,693	11,426	17,324
E	0	0	0	17	112	42	34	1,183	6,128	5,175	1,839	4,296
ESE	0	0	2	7	17	97	135	950	5,595	5,454	5,161	7,909
SE	0	0	1	18	77	9	12	575	2,989	19,343	59,057	76,396
SSE	0	0	3	50	29	27	40	919	5,051	15,259	443,326	392,420
S	0	0	0	42	20	30	40	669	4,413	11,763	20,254	34,375
SSW	0	0	0	10	12	54	65	554	3,098	5,803	5,616	6,222
SW	0	0	0	4	14	54	86	1,186	2,678	2,845	5,482	4,576
WSW	0	0	0	19	42	31	63	1,381	4,402	6,729	8,905	8,094
W	0	0	0	31	24	24	29	466	2,883	4,529	109,205	21,748
WNW	0	0	0	30	79	52	29	606	2,725	8,371	17,931	9,934
NW	0	0	1	35	52	92	81	662	3,327	11,604	11,816	3,090
NNW	0	0	0	28	64	13	25	771	4,725	9,040	25,534	10,041

K.1.2.4 Design Basis Events

Design basis events are defined by the American Nuclear Society as Condition IV occurrences or limiting faults. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of substantial radioactive material. These are the most serious events which must be designed against and represent limiting design cases.

The accident analyses presented in the UFSARs are conservative design basis analyses and therefore the dose consequences are bounding (i.e., a realistically based analysis would result in lower doses). The results, however, provide a comparison of the potential consequences resulting from design basis accidents. The consequences also provide insight into which design basis accidents should be analyzed in an environmental impact statement, such as the SPD EIS. After reviewing the UFSAR accident analyses, the design basis accidents chosen for evaluation in the SPD EIS are a large-break LOCA and a fuel-handling accident.

LOCA. A design basis large-break LOCA was chosen for evaluation because it is the limiting reactor design basis accident at each of the three plants. The analysis was performed in accordance with the methodology and assumptions in Regulatory Guide 1.4 (NRC 1974). The large-break LOCA is defined as a break equivalent in size to a double-ended rupture of the largest pipe of the reactor coolant system. Following a postulated double-ended rupture of a reactor coolant pipe, the emergency core cooling system keeps cladding temperatures well below melting, ensuring that the core remains intact and in a coolable geometry. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the containment. Although no core melting would occur for the design basis LOCA, a gross release of fission products is evaluated. The only postulated mechanism for such a release would require a number of simultaneous and extended failures to occur in the engineered safety feature systems, producing severe physical degradation of core geometry and partial melting of the fuel.

Development of the LOCA source term is based on the conservative assumptions specified in Regulatory Guide 1.4. Consistent with this Regulatory Guide, 100 percent of the noble gas inventory and 25 percent of the iodine inventory in the core are assumed to be immediately available for leakage from the primary containment. However, all of this radioactivity is not released directly to the environment because there are a number of mitigating mechanisms which can delay or retain radioisotopes. The principal mechanism, the primary containment, substantially restricts the release rate of the radioisotopes. Following a postulated LOCA, another potential source of fission product release to the environment is the leakage of radioactive water from engineered safety feature equipment located outside containment. The fission products could then be released from the water into the atmosphere, resulting in offsite radiological consequences that contribute to the total dose from the LOCA.

The LOCA radiological consequence analysis for the LEU cores was performed assuming a ground-level release based on offeror-supplied plant-specific radioisotope release data. All possible leak paths (containment, bypass, and the emergency core cooling system) were included. Were a LOCA to occur, a substantial percentage of the releases would be expected to be elevated, which would be expected to reduce the consequences from those calculated in this analysis. To analyze the accident for a partial MOX core, the LEU isotopic activity was multiplied by the MOX/LEU ratios (from Table K–2) to provide a MOX core activity for each isotope. The LEU and MOX LOCA releases for Catawba and McGuire are provided in Table K–6 and for North Anna in Table K–7.

Table K–6. Catawba and McGuire LOCA Source Term

Isotope	LEU LOCA Release (Ci)	MOX/LEU Ratio	40% MOX Core Release (Ci)
Iodine 131	2.42×10^4	1.03	2.49×10^4
Iodine 132	7.76×10^2	1.02	7.92×10^2
Iodine 133	3.22×10^3	1.00	3.22×10^3
Iodine 134	6.55×10^2	0.98	6.42×10^2
Iodine 135	2.51×10^3	1.00	2.51×10^3
Krypton 83m	3.62×10^3	0.89	3.22×10^3
Krypton 85	1.96×10^4	0.78	1.53×10^4
Krypton 85m	1.96×10^4	0.86	1.68×10^4
Krypton 87	1.04×10^4	0.85	8.82×10^3
Krypton 88	3.23×10^4	0.84	2.72×10^4
Xenon 131m	2.79×10^4	1.02	2.84×10^4
Xenon 133	2.33×10^6	1.00	2.33×10^6
Xenon 133m	3.45×10^4	1.01	3.49×10^4
Xenon 135	2.90×10^5	1.28	3.71×10^5
Xenon 135m	1.40×10^3	1.04	1.46×10^3
Xenon 138	7.21×10^3	0.96	6.92×10^3

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Fuel-Handling Accident. The fuel-handling accident analysis was performed in a conservative manner, in accordance with Regulatory Guide 1.25 methodology (NRC 1972). In the fuel-handling accident scenario, a spent fuel assembly is dropped. The drop results in a breach of the fuel rod cladding, and a portion of the volatile fission gases from the damaged fuel rods is released. A fuel-handling accident would realistically result in only a fraction of the fuel rods being damaged. However, consistent with NRC methodology, all the fuel rods in the assembly are assumed to be damaged.

Table K–7. North Anna LOCA Source Term

Isotope	LEU LOCA Release (Ci)	MOX/LEU Ratio	40% MOX Core Release (Ci)
Iodine 131	3.68×10^{2}	1.03	3.79×10^{2}
Iodine 132	3.45×10^{2}	1.02	3.52×10^{2}
Iodine 133	5.87×10^{2}	1.00	5.87×10^{2}
Iodine 134	5.10×10^{2}	0.98	5.00×10^{2}
Iodine 135	5.01×10^{2}	1.00	5.01×10^{2}
Krypton 83m	4.26×10^{2}	0.89	3.79×10^{2}
Krypton 85	5.06×10^{1}	0.78	3.95×10^{1}
Krypton 85m	1.48×10^{3}	0.86	1.27×10^{3}
Krypton 87	2.22×10^{3}	0.85	1.89×10^{3}
Krypton 88	3.50×10^{3}	0.84	2.94×10^{3}
Xenon 131m	3.20×10^{1}	1.02	3.26×10^{1}
Xenon 133	6.91×10^{3}	1.00	6.91×10^{3}
Xenon 133m	1.70×10^{2}	1.01	1.72×10^{2}
Xenon 135	6.37×10^{3}	1.28	8.15×10^{3}
Xenon 135m	6.72×10^{2}	1.04	6.99×10^{2}
Xenon 138	1.90×10^{3}	0.96	1.82×10^{3}

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

The accident is assumed to occur at the earliest time fuel-handling operations may begin after shutdown as identified in each plant's Technical Specifications.[1] The assumed accident time is 72 hr after shutdown at Catawba and McGuire. North Anna Technical Specifications require a minimum of 150 hr between shutdown and the initiation of fuel movement, but assumed an accident time of 100 hr.

As assumed in Regulatory Guide 1.25, the damaged assembly is the highest powered assembly being removed from the reactor. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. All of the gap activity in the damaged rods is assumed to be released to the spent fuel pool. Noble gases released to the spent fuel pool are immediately released at ground level to the environment, but the water in the spent fuel pool greatly reduces the iodine available for release to the environment. It is assumed that all of the iodine escaping from the spent fuel pool is released to the environment at ground level over a 2-hr time period through the fuel-handling building ventilation system. The Catawba and McGuire UFSARs assume iodine filter efficiencies of 95 percent for both the inorganic and organic species. The North Anna UFSAR assumes a filter efficiency of 90 percent for the inorganic iodine and 70 percent for the organic iodine. The LEU and MOX source terms for Catawba and McGuire are provided in Table K–8 and the source terms for North Anna are provided in Table K–9.

The frequencies for the design basis LOCAs, obtained from the IPEs, are Catawba, 7.50×10^{-6}; McGuire, 1.50×10^{-5}; and North Anna, 2.10×10^{-5}. The frequencies of the fuel-handling accidents were estimated in lieu of plant-specific data. For conservatism, a frequency of 1×10^{-4} was chosen for the analysis.

[1] Technical Specifications are plant-specific operating conditions that control safety-related parameters of plant operation. Technical Specifications are part of the operating license and require an operating license amendment to change.

Table K–8. Catawba and McGuire Fuel-Handling Accident Source Term

Nuclide	LEU Release (Ci)	MOX/LEU Ratio	40% MOX Core Release
Iodine 131	3.83×10^{1}	1.03	3.94×10^{1}
Iodine 132	5.55×10^{1}	1.02	5.66×10^{1}
Iodine 133	8.00×10^{1}	1.00	8.00×10^{1}
Iodine 134	8.80×10^{1}	0.98	8.62×10^{1}
Iodine 135	7.55×10^{1}	1.00	7.55×10^{1}
Krypton 83m	9.47×10^{3}	0.89	8.43×10^{3}
Krypton 85	1.11×10^{3}	0.78	8.66×10^{2}
Krypton 85m	2.16×10^{4}	0.86	1.86×10^{4}
Krypton 87	4.04×10^{4}	0.85	3.43×10^{4}
Krypton 88	5.58×10^{4}	0.84	4.69×10^{4}
Xenon 133	1.60×10^{5}	1.00	1.60×10^{5}
Xenon 133m	4.81×10^{3}	1.01	4.86×10^{3}
Xenon 135	1.65×10^{5}	1.28	2.11×10^{5}
Xenon 135m	2.96×10^{4}	1.04	3.08×10^{4}
Xenon 138	1.34×10^{5}	0.96	1.29×10^{5}

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–9. North Anna Fuel-Handling Accident Source Term

Nuclide	LEU Release (Ci)	MOX/LEU Ratio	40% MOX Core Release
Iodine 131	9.05×10^{1}	1.03	9.32×10^{1}
Iodine 132	1.37×10^{2}	1.02	1.40×10^{2}
Iodine 133	2.01×10^{2}	1.00	2.01×10^{2}
Iodine 134	2.36×10^{2}	0.98	2.31×10^{2}
Iodine 135	1.82×10^{2}	1.00	1.82×10^{2}
Krypton 85	2.60×10^{3}	0.78	2.03×10^{3}
Krypton 85m	2.65×10^{4}	0.86	2.28×10^{4}
Krypton 87	5.10×10^{4}	0.85	4.34×10^{4}
Krypton 88	7.25×10^{4}	0.84	6.09×10^{4}
Xenon 131m	4.56×10^{2}	1.02	4.65×10^{2}
Xenon 133	1.36×10^{5}	1.00	1.36×10^{5}
Xenon 133m	3.46×10^{3}	1.01	3.49×10^{3}
Xenon 135	3.70×10^{4}	1.28	4.74×10^{4}
Xenon 135m	3.74×10^{4}	1.04	3.89×10^{4}
Xenon 138	1.22×10^{5}	0.96	1.17×10^{5}

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

K.1.2.5 Beyond-Design-Basis Events

Beyond-design-basis accidents (severe reactor accidents) are less likely to occur than reactor design basis accidents. In the reactor design basis accidents, the mitigating systems are assumed to be available. In the severe reactor accidents, even though the initiating event could be a design basis event (e.g., large-break LOCA), additional failures of mitigating systems would cause some degree of physical deterioration of the fuel in the reactor core and a possible breach of the containment structure leading to the direct release of radioactive materials to the environment.

The beyond-design-basis accident evaluation in the SPD EIS included a review of each plant's IPE. In 1988, the NRC required all licensees of operating plants to perform IPEs for severe accident vulnerabilities (Generic Letter 88-20) (NRC 1988), and indicated that a Probabalistic Risk Assessment (PRA) would be an acceptable approach to performing the IPE. A PRA evaluates, in full detail (quantitatively), the consequences of all potential events caused by the operating disturbances (known as internal initiating events) within each plant. The state-of-the-art PRA uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident.

A plant-specific PRA for severe accident vulnerabilities starts with identification of initiating events (i.e., challenges to normal plant operation or accidents) that require successful mitigation to prevent core damage. These events are grouped into initiating event classes that have similar characteristics and require the same overall plant response.

Event trees are developed for each initiating event class. These event trees depict the possible sequence of events that could occur during the plant's response to each initiating event class. The trees delineate the possible combinations (sequences) of functional and/or system successes and failures that lead to either successful mitigation of the initiating event or core damage. Functional and/or system success criteria are developed based on the plant response to the class of accident sequences. Failure modes of systems that are functionally important to preventing core damage are modeled. This modeling process is usually done with fault trees that define the combinations of equipment failures, equipment outages, and human errors that could cause the failure of systems to perform the desired functions.

Quantification of the event trees leads to hundreds, or even thousands, of different end states representing various accident sequences that are either mitigated or lead to core damage. Each accident sequence and its associated end state has a unique "signature" because of the particular combination of system successes and failures. These end states are grouped together into plant damage states, each of which collects sequences for which the progression of core damage, the release of fission products from the fuel, the status of containment and its systems, and the potential for mitigating source terms are similar. The sum of all core damage accident sequences will then represent an estimate of plant core damage frequency. The analysis of core damage frequency calculations is called a Level 1 PRA, or front-end analysis.

Next, an analysis of accident progression, containment loading[2] resulting from the accident, and the structural response to the accident loading is performed. The primary objective of this analysis, which is called a Level 2 PRA, is to characterize the potential for, and magnitude of, a release of radioactive material from the reactor fuel to the environment, given the occurrence of an accident that damages the core. The analysis includes an assessment of containment performance in response to a series of severe accidents. Analysis of the progression of an accident (an accident sequence within a plant damage state) generates a time history of loads imposed on the containment pressure boundary. These loads would then be compared against the containment's structural performance limits. If the loads exceed the performance limits, the containment would be expected to fail; conversely, if the containment performance limits exceed the calculated loads, the containment would be expected to survive. Four modes of containment failure are defined: containment isolation failure, containment bypass, early containment failure, and late containment failure.

The magnitude of the radioactive release to the atmosphere in an accident is dependent on the timing of the reactor vessel failure and the containment failure. To determine the magnitude of the release, a containment

[2] Challenges to containment integrity such as elevated temperature or pressure are referred to as containment loading.

event tree representing the time sequence of major phenomenological events that could occur during the formation and relocation of core debris (after core melt), availability of the containment heat removal system, and the expected mode of containment failures (i.e., bypass, early, and late), is developed. A reduced set of plant damage states is defined by culling the lower frequency plant damage states into higher frequency ones that have relatively similar severity and consequence potential. This condensed set is known as the key plant damage states. These key plant damage states would then become the initiating events for the containment event tree. The outcome of each sequence in this event tree represents a specific release category. Release categories that can be represented by similar source terms are grouped. Source terms associated with various release categories describe the fractional releases for representative radionuclide groups, as well as the timing, duration, and energy of release.

Beyond-design-basis accidents evaluated in the SPD EIS included only those scenarios that lead to containment bypass or failure because the public and environmental consequences would be significantly less for accident scenarios that do not lead to containment bypass or failure. The accidents evaluated consisted of a steam generator tube rupture, an early containment failure, a late containment failure, and an ISLOCA.

Steam Generator Tube Rupture. A beyond-design-basis steam generator tube rupture induced by high temperatures represents a containment bypass event. Analyses have indicated a potential for very high gas temperatures in the reactor coolant system during accidents involving core damage when the primary system is at high pressure. The high temperature could fail the steam generator tubes. As a result of the tube rupture, the secondary side may be exposed to full Reactor Coolant System pressures. These pressures are likely to cause relief valves to lift on the secondary side as they are designed to do. If these valves fail to close after venting, an open pathway from the reactor vessel to the environment can result.

Early Containment Failure. This accident is defined as the failure of containment prior to or very soon (within a few hours) after breach of the reactor vessel. A variety of mechanisms such as direct contact of core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, and fuel-coolant interactions can cause structural failure of the containment. Early containment failure can be important because it tends to result in shorter warning times for initiating public protective measures, and because radionuclide releases would generally be more severe than if the containment fails late.

Late Containment Failure. A late containment failure involves structural failure of the containment several hours after breach of the reactor vessel. A variety of mechanisms such as gradual pressure and temperature increase, hydrogen combustion, and basemat melt-through by core debris can cause late containment failure.

ISLOCA. An ISLOCA refers to a class of accidents in which the reactor coolant system pressure boundary interfacing with a supporting system of lower design pressure is breached. If this occurs, the lower pressure system will be overpressurized and could rupture outside the containment. This failure would establish a flow path directly to the environment or, sometimes, to another building of small-pressure capacity.

For each of the proposed reactors, an assessment was made of the pre-accident inventories of each radioactive species in the reactor fuel, using information on the thermal power and refueling cycles. For the source term and offsite consequence analysis, the radioactive species were collected into groups that exhibit similar chemical behavior. The following groups represent the radionuclides considered to be most important to offsite consequences: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium.

The LEU end-of-cycle isotopic activities (inventories) were multiplied by the MOX/LEU ratio to provide a MOX end-of-cycle activity for each isotope. The LEU and MOX core activities for Catawba and McGuire are provided in Table K–10. The activities for North Anna are provided in Table K–11.

Table K–10. Catawba and McGuire End-of-Cycle Core Activities

Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)	Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)
Americium 241	3.13×10^{3}	2.06	6.45×10^{3}	Niobium 95	1.41×10^{8}	0.94	1.33×10^{8}
Antimony 127	7.53×10^{6}	1.15	8.66×10^{6}	Plutonium 238	9.90×10^{4}	0.76	7.53×10^{4}
Antimony 129	2.67×10^{7}	1.07	2.85×10^{7}	Plutonium 239	2.23×10^{4}	2.06	4.60×10^{4}
Barium 139	1.70×10^{8}	0.97	1.65×10^{8}	Plutonium 240	2.82×10^{4}	2.20	6.20×10^{4}
Barium 140	1.68×10^{8}	0.98	1.65×10^{8}	Plutonium 241	4.74×10^{6}	1.79	8.49×10^{6}
Cerium 141	1.53×10^{8}	0.98	1.50×10^{8}	Praseodymium 143	1.46×10^{8}	0.95	1.39×10^{8}
Cerium 143	1.48×10^{8}	0.95	1.41×10^{8}	Rhodium 105	5.53×10^{7}	1.19	6.58×10^{7}
Cerium 144	9.20×10^{7}	0.91	8.37×10^{7}	Rubidium 86	5.10×10^{4}	0.77	3.93×10^{4}
Cesium 134	1.17×10^{7}	0.85	9.93×10^{6}	Ruthenium 103	1.23×10^{8}	1.11	1.36×10^{8}
Cesium 136	3.56×10^{6}	1.09	3.88×10^{6}	Ruthenium 105	7.98×10^{7}	1.18	9.42×10^{7}
Cesium 137	6.53×10^{6}	0.91	5.94×10^{6}	Ruthenium 106	2.79×10^{7}	1.28	3.57×10^{7}
Cobolt 58	8.71×10^{5}	0.86	7.49×10^{5}	Strontium 89	9.70×10^{7}	0.83	8.05×10^{7}
Cobolt 60	6.66×10^{5}	0.72	4.80×10^{5}	Strontium 90	5.24×10^{6}	0.75	3.93×10^{6}
Curium 242	1.20×10^{6}	1.43	1.71×10^{6}	Strontium 91	1.25×10^{8}	0.86	1.07×10^{8}
Curium 244	7.02×10^{4}	0.94	6.60×10^{4}	Strontium 92	1.30×10^{8}	0.89	1.16×10^{8}
Iodine 131	8.66×10^{7}	1.03	8.92×10^{7}	Technetium 99m	1.42×10^{8}	0.99	1.41×10^{8}
Iodine 132	1.28×10^{8}	1.02	1.30×10^{8}	Tellurium 127	7.28×10^{6}	1.16	8.44×10^{6}
Iodine 133	1.83×10^{8}	1.00	1.83×10^{8}	Tellurium 127m	9.63×10^{5}	1.20	1.16×10^{6}
Iodine 134	2.01×10^{8}	0.98	1.97×10^{8}	Tellurium 129	2.50×10^{7}	1.08	2.70×10^{7}
Iodine 135	1.73×10^{8}	1.00	1.73×10^{8}	Tellurium 129m	6.60×10^{6}	1.09	7.20×10^{6}
Krypton 85	6.69×10^{5}	0.78	5.22×10^{5}	Tellurium 131m	1.26×10^{7}	1.11	1.40×10^{7}
Krypton 85m	3.13×10^{7}	0.86	2.69×10^{7}	Tellurium 132	1.26×10^{8}	1.01	1.27×10^{8}
Krypton 87	5.72×10^{7}	0.85	4.87×10^{7}	Xenon 133	1.83×10^{8}	1.00	1.83×10^{8}
Krypton 88	7.74×10^{7}	0.84	6.50×10^{7}	Xenon 135	3.44×10^{7}	1.28	4.40×10^{7}
Lanthanum 140	1.72×10^{8}	0.97	1.67×10^{8}	Yttrium 90	5.62×10^{6}	0.76	4.27×10^{6}
Lanthanum 141	1.57×10^{8}	0.97	1.53×10^{8}	Yttrium 91	1.18×10^{8}	0.85	1.00×10^{8}
Lanthanum 142	1.52×10^{8}	0.97	1.47×10^{8}	Yttrium 92	1.30×10^{8}	0.89	1.16×10^{8}
Molybdenum 99	1.65×10^{8}	0.99	1.63×10^{8}	Yttrium 93	1.47×10^{8}	0.91	1.34×10^{8}
Neodymium 147	6.52×10^{7}	0.98	6.39×10^{7}	Zirconium 95	1.49×10^{8}	0.94	1.40×10^{8}
Neptunium 239	1.75×10^{9}	0.99	1.73×10^{9}	Zirconium 97	1.56×10^{8}	0.98	1.53×10^{8}

Key: LEU, low-enriched uranium.

Table K–11. North Anna End-of-Cycle Core Activities

Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)	Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)
Americium 241	1.03×10^{4}	2.06	2.13×10^{4}	Plutonium 238	1.99×10^{5}	0.76	1.51×10^{5}
Antimony 127	6.36×10^{6}	1.15	7.31×10^{6}	Plutonium 239	2.70×10^{4}	2.06	5.57×10^{4}
Antimony 129	2.41×10^{7}	1.07	2.58×10^{7}	Plutonium 240	3.43×10^{4}	2.20	7.54×10^{4}
Barium 139	1.39×10^{8}	0.97	1.35×10^{8}	Plutonium 241	9.82×10^{6}	1.79	1.76×10^{7}
Barium 140	1.37×10^{8}	0.98	1.34×10^{8}	Praseodymium 143	1.17×10^{8}	0.95	1.11×10^{8}
Cerium 141	1.25×10^{8}	0.98	1.22×10^{8}	Rhodium 105	7.22×10^{7}	1.19	8.59×10^{7}
Cerium 143	1.18×10^{8}	0.95	1.12×10^{8}	Rubidium 86	1.45×10^{4}	0.77	1.12×10^{4}
Cerium 144	9.70×10^{7}	0.91	8.82×10^{7}	Rubidium 103	1.16×10^{8}	1.11	1.28×10^{8}
Cesium 134	1.28×10^{7}	0.85	1.09×10^{7}	Rubidium 105	7.84×10^{7}	1.18	9.25×10^{7}
Cesium 136	3.42×10^{6}	1.09	3.72×10^{6}	Rubidium 106	3.83×10^{7}	1.28	4.90×10^{7}
Cesium 137	8.41×10^{6}	0.91	7.66×10^{6}	Strontium 89	7.48×10^{7}	0.83	6.21×10^{7}
Curium 242	2.72×10^{6}	1.43	3.88×10^{6}	Strontium 90	6.22×10^{6}	0.75	4.66×10^{6}
Curium 244	2.75×10^{5}	0.94	2.58×10^{5}	Strontium 91	9.36×10^{7}	0.86	8.05×10^{7}
Iodine 131	7.33×10^{7}	1.03	7.55×10^{7}	Strontium 92	1.04×10^{8}	0.89	9.23×10^{7}
Iodine 132	1.07×10^{8}	1.02	1.09×10^{8}	Technetium 99m	1.26×10^{8}	0.99	1.25×10^{8}
Iodine 133	1.52×10^{8}	1.00	1.52×10^{8}	Tellurium 127	6.21×10^{6}	1.16	7.21×10^{6}
Iodine 134	1.75×10^{8}	0.98	1.71×10^{8}	Tellurium 127m	9.87×10^{5}	1.20	1.18×10^{6}
Iodine 135	1.49×10^{8}	1.00	1.49×10^{8}	Tellurium 129	2.29×10^{7}	1.08	2.47×10^{7}
Krypton 85	3.51×10^{6}	0.78	2.74×10^{6}	Tellurium 129m	4.20×10^{6}	1.09	4.58×10^{6}
Krypton 85m	8.69×10^{5}	0.86	7.48×10^{5}	Tellurium 132	1.07×10^{8}	1.01	1.08×10^{8}
Krypton 87	3.86×10^{7}	0.85	3.28×10^{7}	Xenon 133	1.59×10^{8}	1.00	1.59×10^{8}
Krypton 88	5.46×10^{7}	0.84	4.59×10^{7}	Xenon 133m	4.69×10^{6}	1.01	4.73×10^{6}
Lanthanum 140	1.42×10^{8}	0.97	1.37×10^{8}	Xenon 135	4.47×10^{7}	1.28	5.72×10^{7}
Lanthanum 141	1.28×10^{8}	0.97	1.24×10^{8}	Yttrium 90	6.21×10^{6}	0.76	4.72×10^{6}
Lanthanum 142	1.24×10^{8}	0.97	1.21×10^{8}	Yttrium 91	9.93×10^{7}	0.85	8.44×10^{7}
Molybdenum 99	1.43×10^{8}	0.99	1.42×10^{8}	Yttrium 92	1.01×10^{8}	0.89	8.97×10^{7}
Neodymium 147	5.12×10^{7}	0.98	5.02×10^{7}	Yttrium 93	1.16×10^{8}	0.91	1.05×10^{8}
Neptunium 239	1.51×10^{9}	0.99	1.50x109	Zirconium 95	1.27×10^{8}	0.94	1.20×10^{8}
Niobium 95	1.31×10^{8}	0.94	1.23×10^{8}	Zirconium 97	1.28×10^{8}	0.98	1.26×10^{8}

Key: LEU, low-enriched uranium.

The source term for each accident, taken from each plant's PRA, is described by the release height, timing, duration, and heat content of the plume, the fraction of each isotope group released, and the warning time (time when offsite officials are warned that an emergency response should be initiated). The PRAs included several release categories for each bypass and failure scenario. These release categories were screened for each accident scenario to determine which release category resulted in the highest risk. The risk was determined by multiplying the consequences by the frequency for each release category. The release category with the highest risk for each scenario was used in the SPD EIS analysis. The highest risk release category source terms for Catawba, McGuire, and North Anna are presented in Table K–12. Also included in each release category characterization is the frequency of occurrence.

The overall risk from beyond-design-basis accidents can be described by the sum of risks from all beyond-design-basis accidents. The group of accidents derived from the screening process results in the highest risks from the containment bypass and failure scenarios. The screened-out accidents in these categories not only result in lower consequences, but also have much lower probabilities, often resulting in risks several orders of magnitude lower. The other type of severe accident scenario for these reactors results in an intact containment. The risks from these events are several orders of magnitude lower than the risks from the bypass and failure scenarios. Therefore, a summation of the severe accident risks presented in the SPD EIS is a good indicator of overall risk.

Table K–12. Beyond-Design-Basis Accident Source Terms

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
CATAWBA												
SG tube rupture[a]	Time: 20 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 7.5 hr	1.04	6.31×10^{-10}	1.0	7.7×10^{-1}	7.9×10^{-1}	7.3×10^{-1}	5.0×10^{-3}	9.4×10^{-2}	1.3×10^{-4}	NA	4.0×10^{-2}
Early containment failure	Time: 6.0 hr Duration: 0.5 hr Energy: 2.0×10^{7} cal/sec (8.37×10^{7} W) Elevation: 10.0 m Warning time: 5.5 hr	5.01	3.42×10^{-8}	1.0	5.5×10^{-2}	4.8×10^{-2}	3.0×10^{-2}	2.5×10^{-4}	2.2×10^{-3}	1.2×10^{-4}	NA	1.7×10^{-3}
Late containment failure	Time: 18.5 hr Duration: 0.5 hr Energy: 1.0×10^{7} cal/sec (4.2×10^{7} W) Elevation: 10.0 m Warning time: 18.0 hr	6.01	1.21×10^{-5}	1.0	3.6×10^{-3}	3.9×10^{-3}	1.8×10^{-3}	5.2×10^{-5}	3.8×10^{-4}	2.6×10^{-5}	NA	1.6×10^{-4}
Interfacing systems LOCA	Time: 6.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 5.5 hr	2.04	6.9×10^{-8}	1.0	8.2×10^{-1}	8.2×10^{-1}	7.9×10^{-1}	5.8×10^{-2}	2.1×10^{-1}	3.1×10^{-2}	NA	1.4×10^{-1}

Table K–12. Beyond-Design-Basis Accident Source Terms (Continued)

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
				McGUIRE								
SG tube rupture	Time: 20.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 7.5 hr	1.04	5.81×10^{-9}	1.0	7.7×10^{-1}	7.9×10^{-1}	7.3×10^{-1}	5.0×10^{-3}	9.4×10^{-2}	1.3×10^{-4}	NA	4.0×10^{-2}
Early containment failure	Time: 6.0 hr Duration: 0.5 hr Energy: 2.0×10^{7} cal/sec (8.37×10^{7} W) Elevation: 10.0 m Warning time: 5.5 hr	5.01	9.89×10^{-8}	1.0	4.4×10^{-2}	3.5×10^{-2}	2.1×10^{-2}	1.4×10^{-4}	4.3×10^{-3}	2.0×10^{-5}	NA	1.4×10^{-3}
Late containment failure	Time: 32.0 hr Duration: 0.5 hr Energy: 1.0×10^{7} cal/sec (4.2×10^{7} W) Elevation: 10.0 m Warning time: 31.5 hr	6.01	7.21×10^{-6}	1.0	3.2×10^{-3}	2.4×10^{-3}	3.3×10^{-3}	1.0×10^{-8}	5.8×10^{-8}	1.0×10^{-9}	NA	1.8×10^{-7}
Interfacing systems LOCA	Time: 3.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 2.0 hr	2.04	6.35×10^{-7}	1.0	7.5×10^{-1}	7.5×10^{-1}	6.6×10^{-1}	4.2×10^{-2}	1.5×10^{-1}	2.0×10^{-2}	NA	9.8×10^{-2}

Table K–12. Beyond-Design-Basis Accident Source Terms (Continued)

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
				NORTH ANNA								
SG tube rupture	Time: 20.3 hr Duration: 1.0 hr Energy: 8.48×10^{3} cal/sec (3.55×10^{4} W) Elevation: 10.0 m Warning time: 7.8 hr	24	7.38×10^{-6}	9.96×10^{-1}	5.2×10^{-1}	5.4×10^{-1}	2.6×10^{-3}/ 6.8×10^{-1}	3.4×10^{-2}	1.4×10^{-1}	5.5×10^{-5}	5.2×10^{-3}	2.1×10^{-2}
Early containment failure	Time: 3.056 hr Duration: 0.5 hr Energy: 1.696×10^{7} cal/sec (7.1×10^{7} W) Elevation: 10.0 m Warning time: 2.556 hr	7	1.60×10^{-7}	9.0×10^{-1}	7.4×10^{-2}	9.7×10^{-2}	1.4×10^{-2}/ 1.3×10^{-1}	1.5×10^{-2}	2.5×10^{-2}	8.1×10^{-6}	9.7×10^{-5}	8.7×10^{-3}
Late containment failure	Time: 8.33 hr Duration: 0.5 hr Energy: 8.48×10^{6} cal/sec (3.55×10^{7} W) Elevation: 10.0 m Warning time: 7.83 hr	9	2.46×10^{-6}	8.2×10^{-1}	2.3×10^{-6}	1.4×10^{-5}	1.6×10^{-5}/ 1.2×10^{-4}	3.2×10^{-4}	3.9×10^{-4}	1.8×10^{-11}	1.4×10^{-11}	1.3×10^{-5}
Interfacing systems LOCA[b]	Time: 5.56 hr Duration: 1.0 hr Energy: 8.48×10^{3} cal/sec (3.55×10^{4} W) Elevation: 10.0 m Warning time: 4.56 hr	23	2.40×10^{-7}	9.4×10^{-1}	2.9×10^{-1}	3.1×10^{-1}	1.6×10^{-5}/ 5.0×10^{-1}	2.3×10^{-1}	2.8×10^{-1}	3.6×10^{-4}	3.7×10^{-2}	1.5×10^{-1}

[a] McGuire data was used for the Catawba steam generator tube rupture event to compare similar scenarios.

[b] McGuire release duration, elevation, and warning time span were used for North Anna in lieu of plant-specific information.

Key: LOCA, loss-of-coolant accident; NA, not applicable; SG, steam generator.

K.1.2.5.1 Evacuation Information

This analysis conservatively assumes that 95 percent of the population within the 16-km (10-mi) emergency planning zone participated in an evacuation. It was also assumed that the five percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hr after plume passage, based on the measured concentrations of radioactivity in the surrounding area and the comparison of projected doses with Environmental Protection Agency (EPA) guidelines. Longer term countermeasures (e.g., crop or land interdiction) were based on EPA Protective Action Guides.

Each beyond-design-basis accident scenario has a warning time and a subsequent release time. The warning time is the time at which notification is given to offsite emergency response officials to initiate protective measures for the surrounding population. The release time is the time when the release to the environment begins. The minimum time between the warning time and the release time is one-half hour. The minimum time of one-half hour is enough time to evacuate onsite personnel (i.e., noninvolved workers). This also conservatively assumes that an onsite emergency has not been declared prior to initiating an offsite notification. Intact containment severe accident scenarios, which were not analyzed because of their insignificant offsite consequences, take place on an even longer time frame.

K.1.2.6 Accident Impacts

Accident impacts are presented in terms of increased risk. Increased risk is defined as the additional risk resulting from using a partial MOX core rather than an LEU core. For example, if the risk of an LCF from an accident with an LEU core is 1.0×10^{-6} and the risk of an LCF from the same accident with a MOX core is 1.1×10^{-6}, then the increased risk of an LCF is 1.0×10^{-7} ($1.1\times10^{-6} - 1.0\times10^{-6} = 1.0\times10^{-7}$).

Tables K–13 through K–18 present the consequences and risks of the postulated set of accidents at Catawba, McGuire, and North Anna, respectively. The receptors include a noninvolved worker located 640 m (0.4 mi) from the release point, the MEI, and the population within an 80-km (50-mi) radius of the reactor site. The consequences and risks are presented for both the current LEU-only and the proposed 40 percent MOX core configurations.

Table K–19 shows the ratios of accident impacts with the proposed 40 percent MOX core to the impacts with the current LEU core. This table shows that the increased risk from accidents to the surrounding population from a MOX core is, on average, less than 5 percent. For the fuel-handling accident at all three plants, the risk is reduced when using MOX fuel.

Severe accident scenarios that postulate large abrupt releases could result in prompt fatalities if the radiation dose is sufficiently high. Of the accidents analyzed in the SPD EIS, the ISLOCA and steam generator tube rupture at Catawba and McGuire, and the ISLOCA at North Anna were the only accidents that resulted in doses high enough to cause prompt fatalities. However, the number of prompt fatalities is expected to increase only for the ISLOCA scenarios. Table K–20 shows the estimated number of prompt fatalities estimated to result from these accidents.

Table K–13. Design Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	7.50×10^{-6}	LEU	3.78	1.51×10^{-3}	1.81×10^{-7}	1.44	7.20×10^{-4}	8.64×10^{-8}	3.64×10^{3}	1.82	2.19×10^{-4}
		MOX	3.85	1.54×10^{-3}	1.86×10^{-7}	1.48	7.40×10^{-4}	8.88×10^{-8}	3.75×10^{3}	1.88	2.26×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.275	1.10×10^{-4}	1.78×10^{-7}	0.138	6.90×10^{-5}	1.10×10^{-7}	1.12×10^{2}	5.61×10^{-2}	8.98×10^{-5}
		MOX	0.262	1.05×10^{-4}	1.68×10^{-7}	0.131	6.55×10^{-5}	1.05×10^{-7}	1.10×10^{2}	5.48×10^{-2}	8.77×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary—given exposure (762 m [2,500 ft]) to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–14. Beyond-Design-Basis Accident Impacts for Catawba With LEU and MOX Fuels

Accident	Frequency (per year)	LEU or MOX Core	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Latent Cancer Fatality[a]	Impacts at Site Boundary: Risk of Latent Cancer Fatality (over campaign)[b]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[c]	Impacts on Population Within 80 km: Risk of Latent Cancer Fatalities (over campaign)[d]
Steam generator tube rupture[e]	6.31×10^{-10}	LEU	3.46×10^{2}	0.346	3.49×10^{-9}	5.71×10^{6}	2.86×10^{3}	2.88×10^{-5}
		MOX	3.67×10^{2}	0.367	3.71×10^{-9}	5.93×10^{6}	2.96×10^{3}	2.99×10^{-5}
Early containment failure	3.42×10^{-8}	LEU	5.97	2.99×10^{-3}	1.63×10^{-9}	7.70×10^{5}	3.85×10^{2}	2.11×10^{-4}
		MOX	6.01	3.01×10^{-3}	1.65×10^{-9}	8.07×10^{5}	4.04×10^{2}	2.21×10^{-4}
Late containment failure	1.21×10^{-5}	LEU	3.25	1.63×10^{-3}	3.15×10^{-7}	3.93×10^{5}	1.96×10^{2}	3.79×10^{-2}
		MOX	3.48	1.74×10^{-3}	3.38×10^{-7}	3.78×10^{5}	1.89×10^{2}	3.66×10^{-2}
Interfacing systems LOCA	6.90×10^{-8}	LEU	1.40×10^{4}	1	1.10×10^{-6}	2.64×10^{7}	1.32×10^{4}	1.46×10^{-2}
		MOX	1.60×10^{4}	1	1.10×10^{-6}	2.96×10^{7}	1.48×10^{4}	1.63×10^{-2}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] McGuire timing and release fractions were used to compare like scenarios.

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–15. Design Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundaries			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Loss-of-coolant accident	1.50×10^{-5}	LEU	5.31	2.12×10^{-3}	5.10×10^{-7}	2.28	1.14×10^{-3}	2.74×10^{-7}	3.37×10^{3}	1.68	4.03×10^{-4}
		MOX	5.46	2.18×10^{-3}	5.25×10^{-7}	2.34	1.17×10^{-3}	2.82×10^{-7}	3.47×10^{3}	1.73	4.16×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.392	1.57×10^{-4}	2.51×10^{-7}	0.212	1.06×10^{-4}	1.70×10^{-7}	99.1	4.96×10^{-2}	7.94×10^{-5}
		MOX	0.373	1.49×10^{-4}	2.38×10^{-7}	0.201	1.01×10^{-4}	1.62×10^{-7}	97.3	4.87×10^{-2}	7.79×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual–a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–16. Beyond-Design-Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Steam generator tube rupture	5.81×10^{-9}	LEU	6.10×10^{2}	0.610	5.66×10^{-8}	5.08×10^{6}	2.54×10^{3}	2.37×10^{-4}
		MOX	6.47×10^{2}	0.647	6.02×10^{-8}	5.28×10^{6}	2.64×10^{3}	2.45×10^{-4}
Early containment failure	9.89×10^{-8}	LEU	12.2	6.10×10^{-3}	9.65×10^{-9}	7.90×10^{5}	3.95×10^{2}	6.26×10^{-4}
		MOX	12.6	6.30×10^{-3}	9.97×10^{-9}	8.04×10^{5}	4.02×10^{2}	6.37×10^{-4}
Late containment failure	7.21×10^{-6}	LEU	2.18	1.09×10^{-3}	1.26×10^{-7}	3.04×10^{5}	1.52×10^{2}	1.76×10^{-2}
		MOX	2.21	1.11×10^{-3}	1.28×10^{-7}	2.96×10^{5}	1.48×10^{2}	1.71×10^{-2}
Interfacing systems LOCA	6.35×10^{-7}	LEU	1.95×10^{4}	1	1.02×10^{-5}	1.79×10^{7}	8.93×10^{3}	0.091
		MOX	2.19×10^{4}	1	1.02×10^{-5}	1.97×10^{7}	9.85×10^{3}	0.10

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–17. Design Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	2.10×10^{-5}	LEU	0.114	4.56×10^{-5}	1.53×10^{-8}	3.18×10^{-2}	1.59×10^{-5}	5.34×10^{-9}	39.4	1.97×10^{-2}	6.62×10^{-6}
		MOX	0.115	4.60×10^{-5}	1.55×10^{-8}	3.20×10^{-2}	1.60×10^{-5}	5.38×10^{-9}	40.3	2.02×10^{-2}	6.78×10^{-6}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.261	1.04×10^{-4}	1.66×10^{-7}	9.54×10^{-2}	4.77×10^{-5}	7.63×10^{-8}	29.4	1.47×10^{-2}	2.35×10^{-5}
		MOX	0.239	9.56x10-5	1.53×10^{-7}	8.61×10^{-2}	4.31×10^{-5}	6.90×10^{-8}	27.5	1.38×10^{-2}	2.21×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–18. Beyond-Design-Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities [c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Steam generator tube rupture[e]	7.38×10^{-6}	LEU	2.09×10^{2}	0.209	2.46×10^{-5}	1.73×10^{6}	8.63×10^{2}	0.102
		MOX	2.43×10^{2}	0.243	2.86×10^{-5}	1.84×10^{6}	9.20×10^{2}	0.109
Early containment failure[e]	1.60×10^{-7}	LEU	19.6	1.96×10^{-2}	5.02×10^{-8}	8.33×10^{5}	4.17×10^{2}	1.07×10^{-3}
		MOX	21.6	2.16×10^{-2}	5.54×10^{-8}	8.42×10^{5}	4.21×10^{2}	1.08×10^{-3}
Late containment failure[e]	2.46×10^{-6}	LEU	1.12	5.60×10^{-4}	2.21×10^{-8}	4.04×10^{4}	20.2	7.95×10^{-4}
		MOX	1.15	5.75×10^{-4}	2.26×10^{-8}	4.43×10^{4}	22.1	8.70×10^{-4}
Interfacing systems LOCA[e]	2.40×10^{-7}	LEU	1.00×10^{4}	1	3.84×10^{-6}	4.68×10^{6}	2.34×10^{3}	8.99×10^{-3}
		MOX	1.22×10^{4}	1	3.84×10^{-6}	5.41×10^{6}	2.70×10^{3}	1.04×10^{-2}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire release durations and warning time spans were used in lieu of site specific data.

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–19. Ratio of Accident Impacts for MOX-Fueled and LEU-Fueled Reactors (MOX Impacts/Uranium Impacts)

	Catawba			McGuire			North Anna		
Accident	Worker	MEI	Population	Worker	MEI	Population	Worker	MEI	Population
LOCA	1.019	1.028	1.033	1.028	1.026	1.030	1.009	1.006	1.025
FHA	0.953	0.949	0.977	0.952	0.948	0.982	0.916	0.903	0.939
SGTR	NA	1.061	1.035	NA	1.061	1.039	NA	1.163	1.066
EARLY	NA	1.007	1.049	NA	1.033	1.018	NA	1.102	1.010
LATE	NA	1.071	0.964	NA	1.014	0.974	NA	1.027	1.094
ISLOCA	NA	1.143	1.121	NA	1.123	1.103	NA	1.220	1.154

Key: Early, early containment; FHA, fuel-handling accident; ISLOCA, interfacing systems loss-of-coolant accident; Late, late containment; LEU, low-enriched uranium; LOCA, loss-of-coolant accident; MEI, maximally exposed individual; NA, not applicable; SGTR, steam generator tube rupture.

Table K–20. Prompt Fatalities for MOX-Fueled and LEU-Fueled Reactors

Accident Scenario	LEU	MOX
Steam generator tube rupture		
Catawba	1	1
McGuire	1	1
North Anna	0	0
Interfacing systems loss-of-coolant accident		
Catawba	815	843
McGuire	398	421
North Anna	54	60

Key: LEU, low-enriched uranium.

K.1.2.6.1 Catawba

Design Basis Accidents. Table K–13 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at Catawba. The greatest risk increase to the surrounding population for a design basis accident with a MOX core configuration is approximately 3.3 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.82 LCF for an LEU core and 1.88 LCF for a partial MOX core. The increased risk to the noninvolved worker is one fatality every 210 million years (4.8×10^{-9} per 16-year campaign); the MEI, one fatality every 420 million years (2.4×10^{-9} per 16-year campaign); and the population, one fatality every 160,000 years (6.4×10^{-6} per 16-year campaign). (The numbers in parenthesis indicate the corresponding risk per year [i.e., one fatality every million years is equivalent to 1.0×10^{-6} fatalities per year].)

Beyond-Design-Basis Accidents. Table K–14 shows the risks and consequences associated with four beyond-design-basis accidents at Catawba. Table K–20 shows prompt fatalities. The greatest risk increase to the surrounding population from a beyond-design-basis accident with a MOX core configuration is approximately 12 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding population within 80 km (50 mi) would be approximately 14,000 fatalities for an LEU core and 15,600 fatalities for a partial MOX core. The increased risk to the population is one fatality every 570 years (1.7×10^{-3} per 16-year campaign). The increased risk of a prompt fatality is one every 32,000 years (3.0×10^{-5} per 16-year campaign).

K.1.2.6.2 McGuire

Design Basis Accidents. Table K–15 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at McGuire. The greatest risk increase to the surrounding population for a design basis accident with a MOX core configuration is 2.9 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.68 LCF for an LEU core and 1.73 LCF for a partial MOX core. The increased risk to the noninvolved worker is one fatality every 69 million years (1.5×10^{-8} per 16-year campaign); the MEI, one fatality every 120 million years (8.0×10^{-9} per 16-year campaign); and the population, one fatality every 78,000 years (1.3×10^{-5} per 16-year campaign).

Beyond-Design-Basis Accidents. Table K–16 shows the risks and consequences associated with four beyond-design-basis accidents at McGuire. Table K–20 shows prompt fatalities. The greatest risk increase to the surrounding population for a beyond-design-basis accident with a MOX core configuration is approximately 10 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding population within 80 km (50 mi) would be approximately 9,300 fatalities with an LEU core and 10,300 with a partial MOX core. The increased risk to the population is one fatality every 110 years (9.3×10^{-3} per 16-year campaign). The increased risk of a prompt fatality is one every 4,300 years (2.3×10^{-4} per 16-year campaign).

K.1.2.6.3 North Anna

Design Basis Accidents. Table K–17 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at North Anna. The greatest risk increase to the surrounding population for a design-basis-accident with a MOX core configuration is approximately 2.5 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.97×10^{-2} LCF for an LEU core and 2.02×10^{-2} LCF for a partial MOX core. The increased risk to the noninvolved worker is one fatality every 7.8 billion years (1.3×10^{-10} per 16-year campaign); the MEI, one fatality every 31 billion years (3.2×10^{-10} per 16-year campaign); and the population, one fatality every 6.2 million years (1.6×10^{-7} per 16-year campaign).

Beyond-Design-Basis Accidents. Table K–18 shows the risks and consequences associated with four beyond-design-basis accidents at North Anna. Table K–20 shows prompt fatalities. The greatest risk increase to the surrounding population from a beyond-design-basis accident with a MOX core configuration is approximately 15 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding populations within 80 km (50 mi) would be approximately 2,400 fatalities for an LEU core and 2,800 fatalities for a partial MOX core. The increased risk to the population is one fatality every 730 years (1.4×10^{-3} per 16-year campaign). The increased risk of a prompt fatality is one every 43,000 years (2.3×10^{-5} per 16-year campaign).

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Appendix M
Analysis of Environmental Justice

M.1 INTRODUCTION

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse health or environmental effects of their programs, policies, and activities on minority and low-income populations.

The Council on Environmental Quality has oversight responsibility for compliance with the National Environmental Policy Act (NEPA). In December 1997, the Council released guidance on environmental justice (CEQ 1997). The Council's guidance was adopted as the basis for the analysis of environmental justice contained in the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS).

M.2 DEFINITIONS AND APPROACH

The following definitions were used in the analysis of environmental justice (CEQ 1997):

- Low-income population: Low-income populations in an affected area should be identified with the annual statistical poverty thresholds from the U.S. Bureau of the Census' Current Population Reports, Series P-60 on Income and Poverty. In identifying low-income populations, agencies may consider as a community either a group of individuals living in geographic proximity to one another, or a set of individuals (such as migrant workers or Native Americans), where either type of group experiences common conditions of environmental exposure or effect.

- Minority: Individual(s) who are members of the following population groups: American Indian or Alaskan Native; Asian or Pacific Islander; Black, not of Hispanic origin; or Hispanic.

- Minority population: Minority populations should be identified where either: (a) the minority population of the affected area exceeds 50 percent or (b) the minority population percentage of the affected area is meaningfully greater than the minority population percentage in the general population or other appropriate unit of geographic analysis. In identifying minority communities, agencies may consider as a community either a group of individuals living in geographic proximity to one another, or a geographically dispersed/transient set of individuals (such as migrant workers or American Indians), where either type of group experiences common conditions of environmental exposure or effect. The selection of the appropriate unit of geographic analysis may be a governing body's jurisdiction, a neighborhood, census tract, or other similar unit that is to be chosen so as to not artificially dilute or inflate the affected minority population. A minority population also exists if there is more than one minority group present and the minority percentage, as calculated by aggregating all minority persons, meets one of the above-stated thresholds.

- Disproportionately high and adverse human health effects: When determining whether human health effects are disproportionately high and adverse, agencies are to consider the following three factors to the extent practical:

  a. Whether the health effects, which may be measured in risks and rate, are significant (as employed by NEPA), or above generally accepted norms. Adverse health effects may include bodily impairment, infirmity, illness, or death; and

b. Whether the risk or rate of hazard exposure by a minority population or low-income population to an environmental hazard is significant (as employed by NEPA) and appreciably exceeds or is likely to appreciably exceed the risk or rate to the general population or other appropriate comparison group; and

c. Whether health effects occur in a minority or low-income population affected by cumulative or multiple adverse exposures from environmental hazards.

- Disproportionately high and adverse environmental effects: When determining whether environmental effects are disproportionately high and adverse, agencies are to consider the following three factors to the extent practical:

  a. Whether there is or will be an impact on the natural or physical environment that significantly (as employed by NEPA) and adversely affects a minority or low-income population. Such effects may include ecological, cultural, human health, economic, or social impacts on minority communities or low-income communities, when those impacts are interrelated to impacts on the natural or physical environment; and

  b. Whether environmental effects are significant (as employed by NEPA) and are or may be having an adverse impact on minority populations or low-income populations that appreciably exceeds or is likely to appreciably exceed those on the general population or other appropriate comparison group; and

  c. Whether the environmental effects occur or would occur in a minority population or low-income population affected by cumulative or multiple adverse exposures from environmental hazards.

Data for the analysis of minorities were extracted from Table P12 of Summary Tape File 3A published on CD ROM by the United States Bureau of the Census (DOC 1992). Data for the analysis of low-income populations were extracted from Table P121 of Standard Tape File 3A.

Potentially affected areas examined in the SPD EIS include the areas surrounding proposed reactor sites for mixed oxide (MOX) fuel irradiation: Catawba Nuclear Station, McGuire Nuclear Station, and North Anna Power Station.

M.3 SPATIAL RESOLUTION

For the purposes of enumeration and analysis, the Census Bureau has defined a variety of areal units (DOC 1992). Areal units of concern in this document include (in order of increasing spatial resolution): States, counties, census tracts, block groups, and blocks. The "block" is generally the smallest of these entities and offers the finest spatial resolution. This term refers to a relatively small geographical area bounded on all sides by visible features such as streets and streams, or by invisible boundaries such as city limits or property lines. During the 1990 census, the Census Bureau subdivided the United States and its territories into 7,017,425 blocks. For comparison, the numbers of counties, census tracts, and block groups used in the 1990 census were 3,248; 62,276; and 229,192; respectively. While blocks offer the finest spatial resolution, economic data required for identification of low-income populations are not available at the block-level of spatial resolution. In the analysis below, block groups are used throughout as the areal unit. Block groups generally contain between 250 and 500 housing units (DOC 1992).

During the decennial census, the Census Bureau collects data from individuals and then aggregates the data according to residence in geographical areas such as counties or block groups. Boundaries of the areal units are selected to coincide with geographical features, such as streams and roads, or political boundaries, such as county and city borders. Boundaries used for aggregation of the census data usually do not coincide with boundaries used in the calculation of health effects. Radiological health effects due to an accident at one of the reactor sites for MOX fuel irradiation are evaluated for persons residing within a distance of 80 km (50 mi) of the accident site. In general, the boundary of the circle with an 80-km (50-mi) radius centered at the accident site will not coincide with boundaries used by the Census Bureau for enumeration of the population in the potentially affected area. Some block groups lie completely inside or outside of the area included in the calculation of health effects. However, block groups intersecting the boundary of the potentially affected area are only partly included. Because the geographical distribution of persons residing within a block group is not available from the census data, partial inclusions introduce uncertainties into the estimate of the population at risk.

In order to evaluate populations at risk in partially included block groups, it was assumed that residents are uniformly distributed throughout the area of each block group. For example, if 85 percent of the area of a block group lies within 80 km (50 mi) of the accident site, then it was assumed that 85 percent of the population residing in that block group would be at risk. An upper bound for the population at risk was obtained by including the total population of partially included block groups in the population at risk. Similarly, a lower bound for the population at risk was obtained by excluding the population of partially included blocks from the population at risk. As a general rule, if the areas of geographic units defined by the Census Bureau are small in comparison with the potentially affected area, then the uncertainties due to partial inclusions will be relatively small. Uncertainties in the estimates of populations surrounding reactor sites for MOX fuel irradiation are described in M.5.1 below.

M.4 POPULATION PROJECTIONS

Health effects were calculated for populations projected to reside in potentially affected areas during the year 2015. Extrapolations of the total population for individual States are available from both the Census Bureau and various State agencies (Campbell 1996). The Census Bureau also projects populations by ethnic and racial classification in 1-year intervals for the years from 1995 to 2025. Data used to project minority populations in the SPD EIS were extracted from the Census Bureau's Web site at www.census.gov/population/www/projections/stproj.html). Minority populations determined from the 1990 census data were taken as a baseline. Then it was assumed that percentage changes in the minority and majority populations of each block group for a given year (compared with the 1990 baseline data) will be the same as percentage changes in the State minority and majority populations projected for the same year. An advantage to this assumption is that the projected populations are obtained with consistent methodology regardless of the State and associated block group involved in the calculation. A disadvantage is that the methodology is insensitive to localized demographic changes that could alter the projection for a specific area.

M.5 RESULTS FOR THE REACTOR SITES

M.5.1 Minority and Low-Income Population Estimates

Table M–1 shows total populations, minority populations, and percentage minority populations that resided within 80 km (50 mi) of the various sites at the time of the 1990 census. The 80-km (50-mi) distance defines the radius of potential radiological effects for calculations of radiation dose to the general population. Table M–2 shows similar data for projected populations in 2015. As discussed above, minority populations residing in potentially affected areas in 1990 were adopted as a baseline. Populations in 2015 were then projected from the baseline data under the assumption that percentage changes in the majority and minority populations residing in the affected areas will be identical to those projected for State populations. The Census Bureau estimates that the national minority percentage will increase from approximately 24 percent in 1990 to nearly 34 percent by 2015 (Census 1996). Percentage minority populations surrounding all three of the proposed reactor sites were less than the national minority percentage in 1990. The projected minority populations residing within 80 km (50 mi) of the Catawba and McGuire reactor sites are expected to remain below the national percentage in 2015. Minority populations surrounding the North Anna site are projected to be somewhat larger than the national average (36 percent versus 34 percent) in 2015, as shown in Table M–2. In Tables M–1 and M–2, the sum of percentages of the different populations may total slightly more or less than 100 percent due to roundoff.

Table M–3 illustrates the uncertainties in the population estimates for the year 2015 due to the partial inclusion of block groups within the boundaries of potentially affected areas. Column 2 of the table lists the number of block groups that are partly within the circle of 80-km (50-mi) radius centered at the various facilities. Column 3 shows the number of block groups that lie completely within the circle. Potentially affected areas surrounding all three of the proposed reactor sites include two States. Columns 2 and 3 show the number of partial or total inclusions for the affected States. Column 4 of the table, denoted as "T/P," shows the number of totally included block groups divided by the number of partially included block groups. In order to minimize the uncertainties in the population estimate, it is desirable that this ratio be as large as possible. Column 5 shows upper bounds for the estimates of the total population listed in column 6. As discussed above, upper bounds were obtained by including the total population of all block groups that lie at least partially within the affected area. Lower bounds for the estimate of total population shown in column 7 were obtained by including only the populations of totally included block groups. Analogous statements apply to columns 8 through 10.

As would be expected from the value of T/P shown in column 4, uncertainties in the total population estimate for the McGuire site were the smallest among the three proposed reactor sites (+3.7 percent and −2.4 percent), as were the uncertainties in the estimate of the minority population at risk near the Catawba site (+5.7 percent and −3.3 percent). Uncertainties in the population estimates for the North Anna site were the largest among the three sites (+6.5 percent and −4.5 percent for total population; +5.9 percent and −4.2 percent for minority population). None of the uncertainties shown in Table M–3 are large enough to noticeably affect the conclusions regarding radiological health effects or environmental justice.

The percentage of low-income persons living within 80 km (50 mi) of the proposed reactor sites was also projected to 2015. In 1990, the percentage of low-income persons (i.e., those with reported incomes below the poverty threshold) residing in the contiguous United States was 13.1 percent. The percentage of low-income persons living within 80 km (50 mi) of the proposed reactor sites was lower than the national average in every case. Around the Catawba site, the percentage of low-income persons living within 80 km (50 mi), in 1990, was 10.5 percent. At the McGuire site, the percentage was 9.8 percent, and around the North Anna site, the percentage was 6.9 percent.

The estimated number of low-income persons living within 80 km (50 mi) of the Catawba site in 2015 is 157,477 or 7.0 percent of the projected population. The estimated number of low-income persons living within 80 km (50 mi) of the McGuire site in 2015 is 171,182 or 6.6 percent of the projected population. The estimated number of low-income persons living within 80 km (50 mi) of the North Anna site in 2015 is 110,531 or 5.4 percent of the projected population. Based on the fact that all of these areas had low-income percentages lower than the national average in 1990 and that the percentages are projected to decline from the 1990 levels, it is estimated that the percentage of low-income persons living within 80 km (50 mi) of the proposed reactor sites will remain lower than the national average for all three sites.

Table M–1. Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Candidate Sites in 1990

Reactor Site	Total Pop.	Minority Pop.	Percent Minority Pop.	Asian or Pacific Islander Pop.	Percent Asian or Pacific Islander Pop.	Black Pop.	Percent Black Pop	Hispanic Pop.	Percent Hispanic Pop.	Native American Pop.	Percent Native American Pop.	Other Race	Percent Other Race Pop.	White Pop.	Percent White Pop.
Catawba	1,519,392	315,089	20.7	10,942	0.7	288,382	19.0	10,666	0.7	5,098	0.3	442	0.0	1,203,861	79.2
McGuire	1,738,966	305,717	17.6	12,007	0.7	275,789	15.9	12,094	0.7	5,828	0.3	479	0.0	1,432,770	82.4
North Anna	1,286,156	281,652	21.9	18,783	1.5	241,619	18.8	17,550	1.4	3,686	0.3	947	0.1	1,003,557	78.0

Table M–2. Projected Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Candidate Sites in 2015

Reactor Site	Total Pop.	Minority Pop.	Percent Minority Pop.	Asian or Pacific Islander Pop.	Percent Asian or Pacific Islander Pop.	Black Pop.	Percent Black Pop	Hispanic Pop.	Percent Hispanic Pop.	Native American Pop.	Percent Native American Pop.	Other Race	Percent Other Race Pop.	White Pop.	Percent White Pop.
Catawba	2,265,495	597,376	26.4	37,756	1.7	507,810	22.4	40,504	1.8	10,700	0.5	606	0.0	1,668,119	73.6
McGuire	2,575,369	620,701	24.1	43,333	1.7	517,577	20.1	46,486	1.8	12,635	0.5	670	0.0	1,954,668	75.9
North Anna	2,042,200	731,773	35.8	106,086	5.2	508,719	24.9	111,992	5.5	4,976	0.2	1,165	0.1	1,309,262	64.1

Table M–3. Uncertainties in Estimates of Total and Minority Populations for the Year 2015

Reactor Site	No. of Partially Included Block Groups	No. of Fully Included Block Groups	T/P	Upper Bound for Total Population	Estimate of Total Population	Lower Bound for Total Population	Upper Bound for Minority Population	Estimate of Minority Population	Lower Bound for Minority Population
Catawba	54 (NC) 52 (SC)	851 (NC) 314 (SC)	11.0	2,395,224	2,265,495	2,191,319	627,435	597,376	579,620
McGuire	64 (NC) 24 (SC)	1,190 (NC) 129 (SC)	15.0	2,672,795	2,575,369	2,513,292	636,842	620,701	611,521
North Anna	84 (VA) 10 (MD)	710 (VA) 5 (MD)	7.6	2,175,504	2,042,200	1,949,928	775,277	731,773	700,983

M.5.2 Environmental Effects on Minority and Low-Income Populations Residing Near Proposed Reactor Sites

The analysis of environmental effects on populations residing within 80 km (50 mi) of proposed reactor sites is presented in Chapter 4 of the SPD EIS. This analysis shows that no radiological fatalities are likely to result from implementation of the proposed action or alternatives. Radiological risks to the public are small regardless of the racial and ethnic composition of the population, and regardless of the economic status of individuals comprising the population. Nonradiological risks to the general population are also small regardless of the racial and ethnic composition or economic status of the population. Thus, disproportionately high and adverse impacts on minority and low-income populations residing near the various facilities are not likely to result from implementation of the proposed action or alternatives.

M.6 REFERENCES

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ENVIRONMENTAL SYNOPSIS OF INFORMATION PROVIDED IN RESPONSE TO THE REQUEST FOR PROPOSALS FOR MOX FUEL FABRICATION AND REACTOR IRRADIATION SERVICES

April 1999

1.0 INTRODUCTION

In the aftermath of the Cold War, significant quantities of weapons-usable fissile materials (primarily plutonium and highly enriched uranium) have become surplus to national defense needs both in the United States and Russia. President Clinton announced, on September 27, 1993, the establishment of a framework for United States efforts to prevent the proliferation of weapons of mass destruction. As key elements of the President's policy, the United States will:

X Seek to eliminate, where possible, accumulation of stockpiles of highly enriched uranium and plutonium,
X Ensure that where these materials already exist, they are subject to the highest standards of safety, security, and international accountability, and
X Initiate a comprehensive review of long-term options for plutonium disposition, taking into account technical, nonproliferation, environmental, budgetary, and economic considerations.

In January 1994, President Clinton and Russian President Yeltsin agreed that the proliferation of weapons of mass destruction and their delivery systems represent an acute threat to international security. They declared that both Nations would cooperate actively and closely with each other, and also with other interested nations, for the purpose of preventing and reducing this threat.

The Secretary of Energy and the Congress took action in October 1994 to create a permanent Office of Fissile Materials Disposition (MD) within the Department of Energy (DOE) to focus on the important national security objective of eliminating surplus weapons-usable fissile materials. As one of its major responsibilities, MD is tasked with determining how to disposition surplus weapons-usable plutonium. In January 1997, DOE issued a Record of Decision (ROD) for the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (S&D PEIS)[1]. In that decision document, DOE decided to pursue a strategy that would allow for the possibility of both the immobilization of surplus plutonium and the use of surplus plutonium as mixed oxide (MOX) fuel in existing domestic, commercial reactors. In July, 1998, DOE issued the *Draft Surplus Plutonium Disposition Environmental Impact Statement* (SPD Draft EIS)[2] which analyzes sites for plutonium disposition activities and plutonium disposition technologies to support this strategy.

To support the timely undertaking of the surplus plutonium disposition program, DOE initiated a procurement action to contract for fuel fabrication and reactor irradiation services. On May 19, 1998, DOE issued a Request for Proposals (RFP) for these services (Solicitation Number DE-RP02-

[1] DOE/EIS-0229; December 1996
[2] DOE/EIS-0283D; July 1998

98CH10888). The services requested in this procurement process include design, licensing, construction, operation, and eventual decontamination and decommissioning of a MOX facility as well as irradiation of the MOX fuel in existing domestic, commercial reactors should the decision be made by DOE in the SPD EIS ROD to go forward with the MOX program.

In accordance with DOE's National Environmental Policy Act (NEPA) regulations (10 CFR 1021.216), DOE required offerors to submit reasonably available environmental data and analyses as a part of their proposals. DOE independently evaluated and verified the accuracy of the data provided by the offeror in the competitive range, and prepared and considered an Environmental Critique before the procurement selection was made.

As required by Section 216, the Environmental Critique included a discussion of the purpose of the procurement; the salient characteristics of the offeror's proposal; any licenses, permits or approvals needed to support the program; and an evaluation of the potential environmental impacts of the offer. In March 1999, after considering the Environmental Critique, DOE awarded a contract for MOX fuel fabrication and reactor irradiation services. Under this contract, MOX fuel would be fabricated at a DOE site to be selected in the SPD EIS ROD and then irradiated in six domestic commercial nuclear reactors at three commercial reactor sites. Additionally, under the contract only limited activities may be performed prior to issuance of the SPD EIS ROD. These activities include non-site-specific work primarily associated with the development of the initial conceptual design for the fuel fabrication facility, and plans (paper studies) for outreach, long lead-time procurements, regulatory management, facility quality assurance, safeguards, security, fuel qualifications, and deactivation. There would be no construction started on a MOX fuel fabrication facility until the SPD EIS ROD is issued. The MOX facility, if built, would be government-owned, licensed by the Nuclear Regulatory Commission (NRC), and located at one of four candidate DOE sites.

This Synopsis is based on the Environmental Critique and provides a publicly available assessment of the potential environmental impacts associated with the proposal based on an independent review of the representations and data contained in the proposal. The Synopsis serves as a record that DOE has considered the environmental factors and potential consequences of the reasonable alternatives analyzed during the selection process. The Synopsis will be filed with the U.S. Environmental Protection Agency and made publicly available. The Synopsis will also be incorporated into a Supplement to the SPD Draft EIS, which is to be issued in the near future.

2.0 ASSESSMENT METHODS

The analyses in this Synopsis (and in the Environmental Critique) were performed using information submitted by the offeror in the competitive range, independently developed information, publicly available information, and standard computer models and techniques.

In order to evaluate the reasonableness of the offeror's projected environmental impacts compared to those projected by DOE, the offeror's data for the MOX facility was compared to information in the SPD Draft EIS; for the use of MOX fuel in domestic commercial reactors, the offeror's data was compared to information in the S&D PEIS. [3]

Data developed independently to support these analyses include the projection of populations around the proposed reactor sites[4] and information related to the topography surrounding the proposed reactor sites for evaluating air dispersal patterns. Information was also provided by Oak Ridge National Laboratory (ORNL) on the expected ratio of radionuclide activities in MOX fuel compared to that in low enriched uranium (LEU) fuel for use in reactor accident analyses. Standard models for determining radiation doses from normal operations and accident scenarios, and air pollutant concentrations at the proposed disposition facility sites and reactors were run using data provided by the offeror. Reactor accident analyses assumed a 40 percent MOX core because this is a conservative estimate of the amount of MOX fuel that would be used in each of the reactors. The environmental analyses were prepared using the following computer models: GENII for estimating radiation doses to the public from normal operation of the MOX fuel fabrication facility and the proposed reactors; MACCS2 for design-basis and beyond-design-basis accident analyses at the proposed reactors; and ISC3 and SCREEN3 for estimated air pollutant concentrations as a result of normal MOX facility and reactor operations.

3.0 DESCRIPTION OF THE OFFER

The offeror has proposed to build a MOX facility on a DOE site[5] with subsequent irradiation services being provided in six existing reactors at three commercial nuclear power plants in the Eastern United States.

The proposed MOX facility design, which is based on an existing MOX facility in France, will be modified to meet U.S. regulations. Under the proposed design, plutonium dioxide powder would be received from DOE's proposed pit disassembly and conversion facility. The plutonium dioxide would be aqueously processed (polished) to ensure that it meets the agreed-to fuel specification for MOX fuel. Following the polishing step, the plutonium in solution would then be converted back into plutonium dioxide. At that point, the process proposed by the offeror would be similar to that described in Chapter 2 of the SPD Draft EIS[6]. The plutonium dioxide would be mixed with uranium dioxide and formed into MOX fuel pellets.

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[3] Such information is also summarized in the SPD Draft EIS.

[4] Population projections for the area encompassed in a 50-mile radius around the proposed reactor sites were projected to 2015 to approximate the mid-point of the irradiation services program. By 2015, the MOX program would be firmly established at all of the proposed reactor sites and would be expected to remain stable through the end of the program. Using 1990 census data as the base year and state-provided population increase factors for all counties included in this analysis, the population around the sites was projected for 2015. Baseline projections were needed for two of the reactor sites because the population information provided in the proposal was based on 1970 census data. Recent (i.e., 1990) census data were provided for the other proposed site and projected by the offeror to the years 2010 and 2020. From these data points, 2015 projections were interpolated.

[5] This site would be selected in the SPD EIS ROD. As explained in the SPD Draft EIS, DOE's preference is to locate the MOX fuel fabrication plant at DOE's Savannah River site.

[6] The SPD Draft EIS also included evaluation of an aqueous processing facility in Appendix N, that could be added to either the pit conversion or the MOX facility. Based on public comments received and information presented by the offeror subsequent to the release of the SPD Draft EIS, DOE is now considering whether to add the aqueous polishing process to the front end of the MOX facility. The environmental impacts associated with this option will be presented in Chapter 4 of the SPD Final EIS.

These pellets would be baked at high temperature, ground to exact dimensions, then loaded into fuel rods. The MOX fuel rods would then be bundled with standard LEU fuel rods to form MOX fuel assemblies. The MOX fuel assemblies would be shipped to the proposed reactor sites in DOE-provided safe, secure transport vehicles on a near just-in-time basis to minimize the amount of time the fresh MOX fuel would be stored at a reactor site prior to loading into the reactor.

Three sites, each with two operating pressurized light water reactors (PWRs), have been proposed for MOX fuel irradiation. The proposed sites are: the Catawba nuclear generation station near York, South Carolina; the McGuire nuclear generation station near Huntersville, North Carolina; and the North Anna nuclear generation station near Mineral, Virginia. All of these sites have been operating safely for a number of years. Table 1 provides some general information about each of the proposed plants.

Table 1. Reactor Plant Operating Information

Plant	Operator	Capacity (net MWe)	Date of First Operation (mo/yr)
Catawba No. 1	Duke Power Co.	1,129	01/85
Catawba No. 2	Duke Power Co.	1,129	05/86
McGuire No. 1	Duke Power Co.	1,129	07/81
McGuire No. 2	Duke Power Co.	1,129	05/83
North Anna No. 1	Virginia Power Co.	900	04/78
North Anna No. 2	Virginia Power Co.	887	08/80

Table 2 shows the results of the most recent Systematic Assessment of Licensee Performance performed by NRC for each of the proposed reactors. As can be seen in this table, all the proposed reactors have been operated and maintained in a safe manner.

Table 2. Systematic Assessment of Licensee Performance Results

	Catawba	McGuire	North Anna
Date of Latest SALP	06/97	04/97	02/97
Operations	Superior	Superior	Superior
Maintenance	Good	Good	Superior
Engineering	Superior	Good	Good
Plant Support	Superior	Superior	Superior

As proposed by the offeror, both MOX and LEU fuel assemblies would be loaded into the reactor. The MOX fuel assemblies are scheduled to remain in the core for two 18-month cycles and the LEU assemblies for either two or three cycles. After completing a normal (full) fuel cycle, the spent MOX fuel assemblies would be removed from the reactor in accordance with the plant's standard refueling procedures and placed in the plant's spent fuel pool for cooling along with other spent fuel. The offeror has stated that no changes are expected in the plant's spent fuel storage plans to accommodate the spent MOX fuel. Eventually, the fuel would be shipped to a potential geologic repository to be developed by DOE for permanent disposal of commercial spent fuel.

4.0 ENVIRONMENTAL IMPACTS

Human health risk, waste management, land use, infrastructure requirements, accidents, air quality, water quality, and socioeconomics have been evaluated in this Synopsis. Cultural, paleontological and ecological resources, and transportation requirements are not expected to be impacted other than as discussed in the SPD Draft EIS and were not evaluated in this Synopsis. Although four sites are being considered by DOE for the proposed MOX facility, this Environmental Synopsis focuses primarily on environmental impacts at DOE's Savannah River Site (SRS) for the potential MOX facility because, as stated in Section 1.6 of the SPD Draft EIS, it is DOE's preferred location for the MOX facility. However, this Synopsis also discusses non-radiological impacts at other potential MOX facility sites, where appropriate. Unless otherwise noted, impacts would likely be similar at other sites.

4.1 MOX Fuel Fabrication Facility

4.1.1 Human Health Risk

The annual radiological dose from normal operations to the general population residing within 50 miles of the proposed MOX facility at the preferred site, SRS, was calculated based on radiological emissions estimated by the offeror. The major contributor to this dose would be attributable to the offeror's estimated annual release of 0.25 mg of plutonium.[7] In contrast to the "atmospheric release only" assumption presented in the SPD Draft EIS, the MOX facility data provided by the offeror includes both liquid and airborne releases because the proposed process includes some aqueous processing. Table 3 shows the projected radiological dose that would be received by the general population as a result of normal operations of the MOX facility proposed by the offeror.

The average individual living within 50 miles of the SRS site would be expected to receive an annual dose of 2.3×10^{-4} mrem/yr from normal operation of the MOX facility. The maximally exposed individual (MEI) would be expected to receive an annual dose of 3.7×10^{-3} mrem/yr from operation of the MOX facility at SRS. This dose is well below regulatory limits, which require doses resulting from DOE operations to be below 10 mrem/yr from airborne pathways, 4 mrem/yr from drinking water pathways, and 100 mrem/yr from all pathways combined. The additional dose to the general population would also be small in comparison with the average dose received from other SRS activities. For example, in 1997, the average individual living within 50 miles of SRS received a dose of 1.4×10^{-2} mrem/yr from site activities. (SPD Draft EIS, pg. 3-141)

[7]The isotopic distribution of the potential plutonium releases were modeled based on the isotopic distribution developed by Los Alamos National Laboratory for use in the SPD Draft EIS.

Table 3. Estimated Radiological Impacts on the Public from Operations of the MOX Facility at SRS

	Maximally Exposed Ind. (mrem/yr)	Latent Fatal Cancer Risk from 10 Year Operating Life	Est. Dose to Pop. within 50 mi. radius (person-rem/yr)	Latent Fatal Cancers from 10 Year Operating Life	Avg. Dose to Ind. within 50 mi. radius (mrem/yr)	Latent Fatal Cancer Risk from 10 Year Operating Life
Offeror	$3.7x10^{-3}$	$1.9x10^{-8}$	0.181	$9.1x10^{-4}$	$2.3x10^{-4}$	$1.2x10^{-9}$
SPD Draft EIS*	$3.1x10^{-4}$	$1.6x10^{-9}$	0.029	$1.5x10^{-4}$	$3.7x10^{-5}$	$1.9x10^{-10}$
SRS Base**	0.2	$1.0x10^{-6}$	8.6	$4.3x10^{-2}$	$1.4x10^{-2}$	$7.0x10^{-8}$

* Includes contributions from polishing process discussed in Appendix N in addition to those shown in Chapter 4.
** SPD Draft EIS pg. 3-141

Table 4 shows the potential radiological impacts on involved workers at the proposed MOX facility conservatively calculated from 1997 data from the offeror's European operating facility. As shown in Table 4, the average radiation worker at the offeror's proposed MOX facility would receive an annual dose of 65 mrem/yr from normal operations. The offeror has stated that in 1997 the maximum dose to an individual worker at the offeror's MOX facility was 885 mrem, well below the DOE administrative control level of 2,000 mrem/yr and the Federal regulatory limit of 5,000 mrem/yr. The offeror also estimates that fewer radiation workers would be needed to operate the MOX facility than indicated in the SPD Draft EIS. The offeror estimates that approximately 330 radiation workers would be required, rather than the 410 estimated in the SPD Draft EIS.[8]

Table 4. Potential Radiological Impacts on Involved Workers from Operations of the MOX Facility

	No. of Radiation Workers	Average Worker Dose (mrem/yr)	Latent Fatal Cancer Risk from 10 Years of Operation	Total Dose to Workers (person-rem/yr)	Latent Fatal Cancers from 10 Years of Operations
Offeror	330	65	$2.6x10^{-4}$	22	0.088
SPD Draft EIS*	410	500	$2.0x10^{-3}$	205	0.82
SRS Base**	12,500	19	$7.6x10^{-5}$	237	0.95

* Includes contributions from polishing process discussed in Appendix N in addition to the doses shown in Chapter 4.
** SPD Draft EIS pg. 3-142.

4.1.2 Accidents

Design-basis and beyond-design-basis accidents were evaluated in the SPD Draft EIS for the MOX facility and the aqueous plutonium polishing process. Accidents evaluated for the MOX facility included a criticality, fires, and earthquakes. A spill, an uncontrolled reaction resulting in an explosion, a criticality, and an earthquake were evaluated for the plutonium polishing process. Any of these accidents could occur

[8] Although it is estimated that about 385 personnel would be required to operate the facility, only about 330 of the 385 would be considered radiation workers.

in the proposed MOX facility since it would use similar processes.

Including the plutonium polishing process in the MOX facility as proposed by the offeror would make a criticality the bounding design-basis accident for the facility. As shown in Table 5, no major radiological impacts to the general population would be expected from design-basis accidents at the proposed MOX facility. The frequency of this accident, a criticality in solution, is estimated to be between 1 in 10,000 and 1 in 1,000,000 per year.

The bounding beyond-design-basis accident would be an earthquake of sufficient magnitude to collapse the MOX facility. An earthquake of this magnitude would be expected to result in major radiological impacts. However, an earthquake of this magnitude would also be expected to result in widespread damage across the site and throughout the surrounding area. The frequency of an earthquake of this magnitude is estimated to be between 1 in 100,000 and 1 in 10,000,000 per year. Table 5 shows the impact of this accident on SRS. At the other candidate sites, the estimated dose to the general population from this accident would range from 2.0H10^{3} to 5.7H10^{4} with the corresponding number of LCFs expected to range from 1.0 to 28 LCFs. The maximum dose to a person at the site boundary at the time of the accident would be expected to range from 16 to 25 rem with a corresponding risk of latent cancer fatality of 8.0H10^{-3} to 1.2H10^{-2}. A noninvolved worker would be exposed to a dose in the range of 2.2H10^{2} to 6.4H10^{2} rem with a corresponding risk of latent cancer fatality of 8.8H10^{-2} to 2.3H10^{-1}.

Table 5. Bounding Accidents for the Proposed MOX Facility

	Noninvolved Worker (rem)	Probability of Cancer Fatality per Accident	Estimated Dose at Site Boundary (rem)	Probability of Cancer Fatality per Accident	Estimated Dose to Pop. Within 50 mi. radius (person-rem)	Latent Cancer Fatalities per Accident
Criticality at SRS*	3.0×10^{-1}	1.2×10^{-4}	1.6×10^{-2}	8.0×10^{-6}	1.6×10^{1}	8.0×10^{-3}
Beyond-design-basis earthquake**	2.2×10^{2}	8.8×10^{-2}	8.9	4.5×10^{-3}	2.1×10^{4}	10.6

*SPD Draft EIS pg. N-15
**SPD Draft EIS pgs. K-50 and N-15

No major consequences for the maximally exposed involved worker would be expected from leaks, spills, and smaller fires. These accidents are such that involved workers would be able to evacuate immediately or would not be affected by the events. However, explosions could result in immediate injuries from flying debris, as well as the uptake of plutonium and uranium particulates through inhalation. If a criticality were to occur, workers within tens of meters could receive very high to fatal radiation exposures from the initial neutron burst. The dose would strongly depend on the magnitude of the criticality (number of fissions), the distance from the criticality, and the amount of shielding provided by the structures and equipment between the workers and the criticality. Earthquakes could also result in substantial consequences to workers, ranging from workers being killed by collapsing equipment and structures to high radiation exposures and uptakes of radionuclides. For all but the most severe accidents, immediate emergency response actions should reduce the magnitude of the consequences to workers near the accident.

4.1.3 Waste Management

The MOX facility would be expected to produce TRU waste, low-level radioactive waste (LLW), mixed LLW, hazardous waste and sanitary waste in the course of its normal operations. As shown in Table 6, the offeror's estimated generation rates for radioactive wastes are consistent with those estimated in the SPD Draft EIS. None of these estimates is expected to impact the proposed sites in terms of their ability to handle these wastes. The ability to store, treat, and/or dispose of radioactive waste is limited at Pantex. If Pantex were chosen as the site for the MOX facility, the wastes would presumably be handled as discussed in the SPD Draft EIS. TRU waste would have to be stored in the MOX facility until it could be shipped to the Waste Isolation Pilot Plant (WIPP) for permanent disposal. Mixed LLW would be handled in the same manner as current mixed waste that is shipped offsite for treatment and disposal. LLW would be treated and stored onsite until shipped to the Nevada Test Site or a commercial facility for disposal.[9]

Table 6. Estimated Annual Waste Generation Rates

	TRU Waste	Mixed LLW	LLW	Hazardous Waste	Sanitary Waste
Offeror					
Liquid (l/yr)	500	0	300	1,200	11 million
Solid (m^3/yr)	~67	3	94	0.1	150
SPD Draft EIS*					
Liquid (l/yr)	0.5	0.1 l	0.3	1,740	18 million
Solid (m^3/yr)	~67	3	94	1.2	440
SRS Generation Rate**					
Liquid (l/yr)	na	na	na	Na	416 million
Solid (m^3/yr)	431	1,135	10,043	74	6,670

na – not available
*Includes contributions from the polishing process discussed in Appendix N of the SPD Draft EIS, in addition to the wastes shown in Chapter 4.
**SPD Draft EIS pg. 3-130.

4.1.4 Land Use

It is estimated that a total of 6.2 hectares (15.3 acres) would be needed for the MOX facility. This estimate includes 1.0 hectares (2.5 acres) for the process building, 0.2 hectares (0.58 acres) for support facilities, and 5 hectares (12.4 acres) for parking and a security buffer. This is very close to the 6.0 hectares (14.9 acres) estimated in the SPD Draft EIS (pg. E-10). As indicated in the SPD Draft EIS, there is sufficient space available to accommodate the proposed MOX facility at any of the candidate sites.

[9] DOE would ensure that any such disposal would be consistent with the RODs for the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200F, May 1997.

4.1.5 Infrastructure Requirements

The proposed MOX facility would use electricity, natural gas, water, and fuel oil. As shown in Table 7, the offeror's proposed facility would use more of these materials than estimated in the SPD Draft EIS.

Table 7. Estimated MOX Facility Infrastructure Requirements

	Electricity (MWh/yr)	Natural Gas (m^3/yr)	Water (10^6 l/yr)	Fuel Oil (l/yr)
Offeror	30,000	1,070,000	68	63,000
SPD Draft EIS*	17,520	920,000	44	43,000
SRS F-Area Available Capacity**	482,700	na***	1,216	na****

*Includes contributions from the polishing process as discussed in Appendix N in addition to the infrastructure requirements shown in Chapter 4.
**SPD Draft EIS pg. 3-165.
***Heat in F-Area provided by steam.
****Fuel oil trucked in as needed and stored at MOX facility.

4.1.5 Air Quality

Operation of the proposed MOX facility would result in the release of a small amount of nonradiological air pollutants that would be expected to slightly increase the ambient air pollutant concentrations at the selected site. The majority of these pollutants would be associated with routine maintenance and testing runs of the facility's emergency diesel generator and emissions from facility heating. Table 8 shows the estimated increases in ambient air pollutant concentrations for the proposed facility and the national standards for these pollutants. The projected emissions are a very small fraction of the national standards. Although some small radionuclide discharges are expected from the proposed MOX facility, these discharges are not expected to have a major impact on air quality. As explained in Section 4.1.1, these discharges would result in a very small dose to the general public.

Table 8. Estimated Nonradiological Ambient Air Pollutant Concentrations from the Proposed MOX Facility

	Carbon Monoxide 8 hour 1 hour	Nitrogen Dioxide Annual	PM_{10} Annual 24 hour	Sulfur Dioxide Annual 24 hour 3 hour
National Ambient Air Quality Standards (μg/m^3)	10,000 40,000	100	50 150	80 365 1,300
Offeror (μg/m^3)	0.123 0.371	0.011	0.001 0.011	0.039 0.531 1.39
SPD Draft EIS* (μg/m^3)	0.109 0.345	0.011	0.001 0.010	0.031 0.420 1.11
SRS Base** (μg/m^3)	64 279	9.3	4.14 56.4	15.1 219 962

*Includes contributions from the polishing process discussed in Appendix N in addition to the pollutant concentrations shown in Chapter 4.
**SPD Draft EIS pg. 4-6

4.1.6 Water Quality

Table 9 shows a comparison of water resources information described in the SPD Draft EIS to that provided by the offeror. Although the proposed water use is higher than that analyzed in the SPD Draft EIS, the amount of water needed is estimated to be from 0.9 to 6.0 percent of the site's estimated annual water requirements. Therefore, the additional water use is not expected to have a major impact on water resources. Although some small radionuclide discharges are expected from the proposed MOX facility, these discharges are not expected to have a major impact on water quality. As explained in Section 4.1.1, these discharges would result in a very small dose to the general public.

Table 9. Comparison of Water Resources Information for the MOX Facility

	Water Use (10^6 liters/yr)	Sanitary Wastewater Discharged (10^6 liters/yr)	Radionuclide Emissions to Water (Ci)
SPD Draft EIS	44	18	0
Offeror	68	11	0.0025

4.1.7 Socioeconomics

The proposed MOX facility would employ about 385 workers, somewhat fewer than the 435 workers estimated in the SPD Draft EIS. An increase of 385 workers would not be expected to have a major impact on any of the candidate sites. At three of the four candidate sites (i.e., INEEL, Pantex, and SRS), the workforce is projected to be falling at the same time the proposed MOX facility would begin operations. The additional MOX facility workers would help mitigate the negative socioeconomic impacts associated with such reductions. The SPD Draft EIS concluded that, at Hanford, although the increase in workforce requirements for proposed surplus plutonium disposition facilities (including MOX) would coincide with an increase in the site's overall workforce (as a result of the planned tank waste remediation system), the projected changes would not have a major impact on the level of community services currently offered in the region of influence. (SPD Draft EIS pg. 4-37)

4.2 Proposed Reactor Sites

The offeror is proposing to use a partial MOX core (up to approximately 40 percent of the fuel in the core at equilibrium) in each of the proposed reactors. The S&D PEIS analyzed a full MOX core at a generic reactor site.

4.2.1 Human Health Risk

Risk to human health was assessed for the proposed reactor sites based on information provided by the offeror and compared to the generic reactor information in the S&D PEIS. The offeror stated that there would be no difference in dose to the general public from normal operations based on the use of MOX fuel versus LEU fuel in the proposed reactors. This is consistent with findings in the S&D PEIS that showed a very small range in the expected difference (-1.1×10^{-2} to 2×10^{-2} person-rem, S&D PEIS pg. 4-729). The doses shown in this section reflect the projected dose in the year 2015.

The annual radiological dose from normal operations to the general population residing within 50 miles of the proposed reactor sites was estimated based on radiological emissions estimated by the offeror. As shown in Table 10, the average individual living within 50 miles of one of the proposed reactor sites could expect to receive an annual dose of between 2.7×10^{-3} to 9.9×10^{-3} mrem/yr from normal operation of these reactors regardless of whether the reactors were using MOX fuel or LEU fuel.

Table 10. Estimated Dose to the General Population from Normal Operations of the Proposed Reactors in the Year 2015 (Partial MOX or LEU Core)

	Maximally Exposed Individual (mrem/yr)	Latent Fatal Cancer Risk	Est. Dose to Pop. within 50 mi. radius (person-rem/yr)	Annual Number of Latent Cancer Fatalities	Avg. Dose to Ind. within 50 mi. radius (mrem/yr)
Catawba[a]	0.73	3.7×10^{-7}	6.1	3.1×10^{-3}	2.7×10^{-3}
McGuire[b]	0.31	1.6×10^{-7}	10.7	5.4×10^{-3}	4.2×10^{-3}
North Anna[c]	0.37	1.9×10^{-7}	20.3	1.0×10^{-2}	9.9×10^{-3}
S&D PEIS (high)*	0.17	8.5×10^{-8}	2.0	1.0×10^{-3}	7.8×10^{-4}

*S&D PEIS pg. 4-729
[a] The population for the year 2015 is estimated to be 2,265,000.
[b] The population for the year 2015 is estimated to be 2,575,000.
[c] The population for the year 2015 is estimated to be 2,042,000.

The offeror also stated that the workers at the proposed reactor sites would be expected to receive about the same amount of radiation dose as a result of their job activities regardless of the plant's decision to use

MOX fuel. As shown in Table 11, the average radiation worker at the proposed reactor sites could expect to receive an annual dose of between 46 and 123 mrem/yr from normal operations. This is lower than the worker dose range estimated in the S&D PEIS (281 to 543 mrem/yr). The offeror's statement that the use of MOX fuel would not change the estimated worker dose is consistent with data presented in the S&D PEIS that showed an incremental increase in worker dose of less than 0.1 percent due to the use of MOX fuel. (S&D PEIS pg. 4-730)

Table 11. Estimated Dose to Workers from Normal Operations of the Proposed Reactors with MOX Fuel

	No. of Radiation Workers*	Total Dose to Workers (person-rem/ year)	Annual Number of Latent Cancer Fatalities	Average Worker Dose (mrem/yr)	Annual Latent Fatal Cancer Risk
Catawba	3,400	265	0.11	78	3.1×10^{-5}
McGuire	4,000	492	0.20	123	4.9×10^{-5}
North Anna	2,240	103	0.041	46	1.8×10^{-5}
S&D PEIS (high)**	2,220	1,204	0.48	543	2.2×10^{-4}

*The number of radiation workers at the proposed reactor sites was estimated based on the total dose to workers given by the offeror divided by the average worker dose, also supplied by the offeror.
**S&D PEIS pg. 4-730; adjusted to reflect a two reactor site for comparison to the proposed reactor sites.

4.2.2 Accidents

Two design-basis accidents, a large break loss-of-coolant accident (LOCA) and a fuel handling accident (FHA), were evaluated for the Environmental Critique and are reflected in this Synopsis. These accidents were chosen because they are the limiting reactor and non-reactor design-basis accidents at the proposed facilities. As shown in Tables 12 through 14, only small increases in the estimated impacts would be expected from a LOCA at the proposed reactor sites due to the use of MOX fuel. In a FHA, the consequences (defined as latent cancer fatalities) would decrease as a result of using MOX fuel rather than LEU fuel. This is because the end-of-cycle krypton inventory is less in MOX fuel than in LEU fuel and krypton is one of the greatest contributors to radiation dose from a FHA.

Beyond-design-basis accidents, if they were to occur, would be expected to result in major impacts to workers, the surrounding communities, and the environment regardless of whether the reactor was using a LEU or a partial MOX core. As shown in Tables 15 through 17, the probability of a beyond-design-basis accident happening and the risk to an individual living within 50 miles of the proposed reactors is very low.

The largest estimated risk of a latent cancer fatality for the maximally exposed individual (MEI) at any of the proposed reactors is estimated to be 2.86H10^{-5} for a steam generator tube rupture at one of the North Anna reactors when using a partial MOX core. If this same accident were to happen at the reactor when it was using a LEU core, the estimated risk would be 2.46H10^{-5}. In either case, the risk of a latent cancer fatality is estimated to be less than 3 in 100,000 over the 16 year period the reactors would be using MOX fuel.

For beyond-design-basis accidents, the scenarios that lead to containment bypass or failure were evaluated because these are the accidents with the greatest potential consequences. The public and environmental consequences would be significantly less for accident scenarios that do not lead to containment bypass or failure. A steam generator tube rupture, early containment failure, late containment failure, and an interfacing systems loss-of-coolant accident (ISLOCA) were chosen as the representative set of beyond-design-basis accidents.

Commercial reactors, licensed by the NRC are required to complete Individual Plant Examinations (IPE) to assess plant vulnerabilities to severe accidents. An acceptable method of completing the IPEs is to perform a probabilistic risk assessment (PRA). A PRA analysis evaluates, in full detail (quantitatively), the consequences of all potential events caused by the operating disturbances (known as internal initiating events) within each plant. The PRA uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident. The PRAs for the proposed reactors provided the required data to evaluate beyond-design-basis accidents.

As shown in Table 18, the difference in accident consequences for reactors using MOX fuel versus LEU fuel is generally very small. For beyond-design-basis accidents, the consequences would be expected to be slightly higher, with the largest increase associated with an ISLOCA. This is because the MOX fuel will release a higher actinide inventory in a severe accident. The impacts of an ISLOCA are estimated to be about 10 to 15 percent (an average of about 13 percent) greater to the general population living within 50 miles of the reactor operating with a partial MOX core instead of a LEU core. It should be noted that this accident has a very low estimated frequency of occurrence, an average of 1 in 3.2 million per year of reactor operation for the reactors being proposed.

Table 12. Design-Basis Accident Impacts for Catawba with LEU and Mixed Oxide Fuels

			Noninvolved Worker			Maximally Exposed Offsite Individual			Population		
Accident Release Scenario	Accident Scenario Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality Given Dose to Noninvolved Worker[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (rem)	Probability of Latent Cancer Fatality Given Dose at Site Boundary[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (person-rem)	Number of Latent Cancer Fatalities in the Population within 80 km[3]	Risk of Latent Cancer Fatalities (over campaign)[4]
Loss-of-Coolant Accident	7.50×10^{-6}	LEU	3.78	1.51×10^{-3}	1.81×10^{-7}	1.44	7.20×10^{-4}	8.64×10^{-8}	$3.64 \times 10^{+3}$	1.82	2.19×10^{-4}
		MOX	3.85	1.54×10^{-3}	1.86×10^{-7}	1.48	7.40×10^{-4}	8.88×10^{-8}	$3.75 \times 10^{+3}$	1.88	2.26×10^{-4}
Spent Fuel Handling Accident[5]	1.00×10^{-4}	LEU	0.275	1.10×10^{-4}	1.78×10^{-7}	0.138	6.90×10^{-5}	1.10×10^{-7}	$1.12 \times 10^{+2}$	5.61×10^{-2}	8.98×10^{-5}
		MOX	0.262	1.05×10^{-4}	1.68×10^{-7}	0.131	6.55×10^{-5}	1.05×10^{-7}	$1.10 \times 10^{+2}$	5.48×10^{-2}	8.77×10^{-5}

[1] Increased likelihood (probability) of cancer fatality to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (762 m) - if exposed to the indicated dose.

[2] Increased likelihood (probability) of cancer fatality over the estimated 16 year campaign (frequency weighted) to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (762 m).

[3] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 kilometers (50 miles) if exposed to the indicated dose.

[4] Estimated number of cancer fatalities over the estimated 16 year campaign (frequency weighted) in the entire offsite population out to a distance of 80 kilometers (50 miles).

[5] Accident scenario frequency estimated in lieu of plant specific data.

Table 13. Design-Basis Accident Impacts for McGuire with LEU and Mixed Oxide Fuels

			Noninvolved Worker			Maximally Exposed Offsite Individual			Population		
Accident Release Scenario	Accident Scenario Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality Given Dose to Noninvolved Worker[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (rem)	Probability of Latent Cancer Fatality Given Dose at Site Boundary[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (person-rem)	Number of Latent Cancer Fatalities in the Population within 80 km[3]	Risk of Latent Cancer Fatalities (over campaign)[4]
Loss-of-Coolant Accident	1.50×10^{-5}	LEU	5.31	2.12×10^{-3}	5.10×10^{-7}	2.28	1.14×10^{-3}	2.74×10^{-7}	$3.37 \times 10^{+3}$	1.68	4.03×10^{-4}
		MOX	5.46	2.18×10^{-3}	5.25×10^{-7}	2.34	1.17×10^{-3}	2.82×10^{-7}	$3.47 \times 10^{+3}$	1.73	4.16×10^{-4}
Spent Fuel Handling Accident[5]	1.00×10^{-4}	LEU	0.392	1.57×10^{-4}	2.51×10^{-7}	0.212	1.06×10^{-4}	1.70×10^{-7}	99.1	4.96×10^{-2}	7.94×10^{-5}
		MOX	0.373	1.49×10^{-4}	2.38×10^{-7}	0.201	1.01×10^{-4}	1.62×10^{-7}	97.3	4.87×10^{-2}	7.79×10^{-5}

[1] Increased likelihood (probability) of cancer fatality to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (762 m) - if exposed to the indicated dose.

[2] Increased likelihood (probability) of cancer fatality over the estimated 16 year campaign (frequency weighted) to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (762 m).

[3] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 kilometers (50 miles) if exposed to the indicated dose.

[4] Estimated number of cancer fatalities over the estimated 16 year campaign (frequency weighted) in the entire offsite population out to a distance of 80 kilometers (50 miles).

[5] Accident scenario frequency estimated in lieu of plant specific data.

Table 14. Design-Basis Accident Impacts for North Anna with LEU and Mixed Oxide Fuels

			Noninvolved Worker			Maximally Exposed Offsite Individual			Population		
Accident Release Scenario	Accident Scenario Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality Given Dose to Noninvolved Worker[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (rem)	Probability of Latent Cancer Fatality Given Dose at Site Boundary[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (person-rem)	Number of Latent Cancer Fatalities in the Population within 80 km[3]	Risk of Latent Cancer Fatalities (over campaign)[4]
Loss-of-Coolant Accident	2.10×10^{-5}	LEU	0.114	4.56×10^{-5}	1.53×10^{-8}	3.18×10^{-2}	1.59×10^{-5}	5.34×10^{-9}	39.4	1.97×10^{-2}	6.62×10^{-6}
		MOX	0.115	4.60×10^{-5}	1.55×10^{-8}	3.20×10^{-2}	1.60×10^{-5}	5.38×10^{-9}	40.3	2.02×10^{-2}	6.78×10^{-6}
Spent Fuel Handling Accident[5]	1.00×10^{-4}	LEU	0.261	1.04×10^{-4}	1.66×10^{-7}	9.54×10^{-2}	4.77×10^{-5}	7.63×10^{-8}	29.4	1.47×10^{-2}	2.35×10^{-5}
		MOX	0.239	9.56×10^{-5}	1.53×10^{-7}	8.61×10^{-2}	4.31×10^{-5}	6.90×10^{-8}	27.5	1.38×10^{-2}	2.21×10^{-5}

[1] Increased likelihood (probability) of cancer fatality to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (1349 m) - if exposed to the indicated dose.

[2] Increased likelihood (probability) of cancer fatality over the estimated 16 year campaign (frequency weighted) to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (1349 m).

[3] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 kilometers (50 miles) if exposed to the indicated dose.

[4] Estimated number of cancer fatalities over the estimated 16 year campaign (frequency weighted) in the entire offsite population out to a distance of 80 kilometers (50 miles).

[5] Accident scenario frequency estimated in lieu of plant specific data.

Table 15. Beyond-Design-Basis Accident Impacts for Catawba with LEU and Mixed Oxide Fuels

			Maximally Exposed Offsite Individual			Population		
Accident Release Scenario	Accident Scenario Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality Given Dose at Site Boundary[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (person-rem)	Number of Latent Cancer Fatalities in the Population within 80 km[3]	Risk of Latent Cancer Fatalities (over campaign)[4]
Steam Generator Tube Rupture[5]	6.31×10^{-10}	LEU	$3.46\times10^{+2}$	0.346	3.49×10^{-9}	$5.71\times10^{+6}$	$2.86\times10^{+3}$	2.88×10^{-5}
		MOX	$3.67\times10^{+2}$	0.367	3.71×10^{-9}	$5.93\times10^{+6}$	$2.96\times10^{+3}$	2.99×10^{-5}
Early Containment Failure	3.42×10^{-8}	LEU	5.97	2.99×10^{-3}	1.63×10^{-9}	$7.70\times10^{+5}$	$3.85\times10^{+2}$	2.11×10^{-4}
		MOX	6.01	3.01×10^{-3}	1.65×10^{-9}	$8.07\times10^{+5}$	$4.04\times10^{+2}$	2.21×10^{-4}
Late Containment Failure	1.21×10^{-5}	LEU	3.25	1.63×10^{-3}	3.15×10^{-7}	$3.93\times10^{+5}$	$1.96\times10^{+2}$	3.79×10^{-2}
		MOX	3.48	1.74×10^{-3}	3.38×10^{-7}	$3.78\times10^{+5}$	$1.89\times10^{+2}$	3.66×10^{-2}
Interfacing System Loss of Cooling Accident	6.90×10^{-8}	LEU	$1.40\times10^{+4}$	1	1.10×10^{-6}	$2.64\times10^{+7}$	$1.32\times10^{+4}$	1.46×10^{-2}
		MOX	$1.60\times10^{+4}$	1	1.10×10^{-6}	$2.96\times10^{+7}$	$1.48\times10^{+4}$	1.63×10^{-2}

[1] Increased likelihood (probability) of cancer fatality to the maximally exposed offsite individual located at the site boundary (762 m) - if exposed to the indicated dose.
[2] Increased likelihood (probability) of cancer fatality over the estimated 16 year campaign (frequency weighted) to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (762 m).
[3] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 kilometers (50 miles) if exposed to the indicated dose.
[4] Estimated number of cancer fatalities over the estimated 16 year campaign (frequency weighted) in the entire offsite population out to a distance of 80 kilometers (50 miles).
[5] McGuire timing and release fractions were used to compare like scenarios.

Table 16. Beyond-Design-Basis Accident Impacts for McGuire with LEU and Mixed Oxide Fuels

Accident Release Scenario	Accident Scenario Frequency (per year)	LEU or MOX Core	Maximally Exposed Offsite Individual: Dose (rem)	Maximally Exposed Offsite Individual: Probability of Latent Cancer Fatality Given Dose at Site Boundary[1]	Maximally Exposed Offsite Individual: Risk of Latent Cancer Fatality (over campaign)[2]	Population: Dose (person-rem)	Population: Number of Latent Cancer Fatalities in the Population within 80 km[3]	Population: Risk of Latent Cancer Fatalities (over campaign)[4]
Steam Generator Tube Rupture	5.81×10^{-9}	LEU	$6.10\times10^{+2}$	0.610	5.66×10^{-8}	$5.08\times10^{+6}$	$2.54\times10^{+3}$	2.37×10^{-4}
		MOX	$6.47\times10^{+2}$	0.647	6.02×10^{-8}	$5.28\times10^{+6}$	$2.64\times10^{+3}$	2.45×10^{-4}
Early Containment Failure	9.89×10^{-8}	LEU	12.2	6.10×10^{-3}	9.65×10^{-9}	$7.90\times10^{+5}$	$3.95\times10^{+2}$	6.26×10^{-4}
		MOX	12.6	6.30×10^{-3}	9.97×10^{-9}	$8.04\times10^{+5}$	$4.02\times10^{+2}$	6.37×10^{-4}
Late Containment Failure	7.21×10^{-6}	LEU	2.18	1.09×10^{-3}	1.26×10^{-7}	$3.04\times10^{+5}$	$1.52\times10^{+2}$	1.76×10^{-2}
		MOX	2.21	1.11×10^{-3}	1.28×10^{-7}	$2.96\times10^{+5}$	$1.48\times10^{+2}$	1.71×10^{-2}
Interfacing System Loss of Cooling Accident	6.35×10^{-7}	LEU	$1.95\times10^{+4}$	1	1.02×10^{-5}	$1.79\times10^{+7}$	$8.93\times10^{+3}$	0.091
		MOX	$2.19\times10^{+4}$	1	1.02×10^{-5}	$1.97\times10^{+7}$	$9.85\times10^{+3}$	0.10

[1] Increased likelihood (probability) of cancer fatality to the maximally exposed offsite individual located at the site boundary (762 m) - if exposed to the indicated dose.

[2] Increased likelihood (probability) of cancer fatality over the estimated 16 year campaign (frequency weighted) to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (762 m).

[3] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 kilometers (50 miles) if exposed to the indicated dose.

[4] Estimated number of cancer fatalities over the estimated 16 year campaign (frequency weighted) in the entire offsite population out to a distance of 80 kilometers (50 miles).

Table 17. Beyond-Design-Basis Accident Impacts for North Anna with LEU and Mixed Oxide Fuels

			Maximally Exposed Offsite Individual			Population		
Accident Release Scenario	Accident Scenario Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality Given Dose at Site Boundary[1]	Risk of Latent Cancer Fatality (over campaign)[2]	Dose (person-rem)	Number of Latent Cancer Fatalities in the Population within 80 km[3]	Risk of Latent Cancer Fatalities (over campaign)[4]
Steam Generator Tube Rupture[5]	7.38×10^{-6}	LEU	$2.09\times10^{+2}$	0.209	2.46×10^{-5}	$1.73\times10^{+6}$	$8.63\times10^{+2}$	0.102
		MOX	$2.43\times10^{+2}$	0.243	2.86×10^{-5}	$1.84\times10^{+6}$	$9.20\times10^{+2}$	0.109
Early Containment Failure[5]	1.60×10^{-7}	LEU	19.6	1.96×10^{-2}	5.02×10^{-8}	$8.33\times10^{+5}$	$4.17\times10^{+2}$	1.07×10^{-3}
		MOX	21.6	2.16×10^{-2}	5.54×10^{-8}	$8.42\times10^{+5}$	$4.21\times10^{+2}$	1.08×10^{-3}
Late Containment Failure[5]	2.46×10^{-6}	LEU	1.12	5.60×10^{-4}	2.21×10^{-8}	$4.04\times10^{+4}$	20.2	7.95×10^{-4}
		MOX	1.15	5.75×10^{-4}	2.26×10^{-8}	$4.43\times10^{+4}$	22.1	8.70×10^{-4}
Interfacing System Loss of Cooling Accident[5]	2.40×10^{-7}	LEU	$1.00\times10^{+4}$	1	3.84×10^{-6}	$4.68\times10^{+6}$	$2.34\times10^{+3}$	8.99×10^{-3}
		MOX	$1.22\times10^{+4}$	1	3.84×10^{-6}	$5.41\times10^{+6}$	$2.70\times10^{+3}$	1.04×10^{-2}

[1] Increased likelihood (probability) of cancer fatality to the maximally exposed offsite individual located at the site boundary (1349 m) - if exposed to the indicated dose.
[2] Increased likelihood (probability) of cancer fatality over the estimated 16 year campaign (frequency weighted) to a hypothetical individual - a noninvolved worker at a distance of 640 meters or the maximally exposed offsite individual located at the site boundary (1349 m).
[3] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 kilometers (50 miles) if exposed to the indicated dose.
[4] Estimated number of cancer fatalities over the estimated 16 year campaign (frequency weighted) in the entire offsite population out to a distance of 80 kilometers (50 miles).
[5] McGuire release durations and warning times were used in lieu of site specific data.

Table 18. Ratio of Accident Impacts for Mixed Oxide Fueled and Uranium Fueled Reactors (Mixed Oxide Impacts/LEU Impacts)

	Catawba		McGuire		North Anna		S&D PEIS	
Accident Scenario	MEI	Population	MEI	Population	MEI	Population	MEI	Population
Design-Basis Accidents								
Loss-of-Coolant Accident	1.03	1.03	1.01	1.03	1.03	1.03	NA	NA
Fuel Handling Accident	0.95	0.98	0.90	0.94	0.95	0.98	NA	NA
Beyond-Design-Basis Accidents								
Steam Generator Tube Rupture	1.06	1.04	1.16	1.07	1.06	1.04	0.94	0.94
Early Containment Failure	1.01	1.05	1.10	1.01	1.03	1.02	0.96	0.97
Late Containment Failure	1.07	0.96	1.03	1.09	1.01	0.97	1.07	1.08
Interfacing System Loss of Cooling Accident	1.14	1.12	1.22	1.15	1.12	1.10	0.92	0.93

Key: MEI – Maximally Exposed Individual; NA – not available

Note: The number 1 represents the consequences equal to the accident occurring in the proposed reactors with an LEU core

Table 19 shows the number of prompt fatalities estimated from a postulated ISLOCA and a beyond-design-basis steam generator tube rupture. As shown in this table, the differences due to the use of MOX fuel rather than LEU are small. None of the other accidents evaluated in this Synopsis are expected to result in prompt fatalities.

Table 19. Estimated Prompt Fatalities from Beyond-Design-Basis Reactor Accidents

Reactor Site	LEU Core	MOX Core
Steam Generator Tube Rupture		
Catawba	1	1
McGuire	1	1
North Anna	0	0
Interfacing System Loss of Cooling Accident		
Catawba	815	843
McGuire	398	421
North Anna	54	60

4.2.3 Waste Management

The proposed reactors would be expected to continue to produce mixed LLW, LLW, hazardous waste, and nonhazardous waste as part of their normal operations. According to the offeror, the volume of waste generated is not expected to increase as a result of the reactors using MOX fuel. This is consistent with information presented in the S&D PEIS that stated the use of MOX fuel is not expected to increase the amount or change the content of the waste being generated. (S&D PEIS, pg. 4-734) Table 20 shows the annual waste volume that would be generated during operation of the proposed reactors.

Table 20. Estimated Waste Generation Rates

Reactor Site	Mixed LLW (m^3/yr)	LLW (m^3/yr)	Hazardous Waste (m^3/yr)	Nonhazardous Waste Solid (m^3/yr)
Catawba (per unit)	0.3	25	15	455
McGuire (per unit)	0.1	21	14	568
North Anna (per unit)	0.0	118	6	5,200
S&D PEIS*	na	178	na	na

na - not available.
*S&D PEIS pg. 4-734.

As shown in Table 20, the estimated LLW generation for each of the proposed reactors is less than the amount estimated in the S&D PEIS. None of these waste estimates are expected to impact the proposed reactor sites in terms of their ability to handle these wastes. The wastes would continue to be handled in the same manner as they are today with no change required due to the use of MOX fuel at the reactors.

4.2.4 Spent Fuel

As shown in Table 21, it is likely that some additional spent fuel would be generated by using a partial MOX core in the proposed reactors. The amount of additional spent nuclear fuel generated is estimated to range from approximately 2 to 16 percent of the total amount of spent fuel that would be generated by the proposed reactors during the time period MOX fuel would be used. The offeror intends to manage the spent MOX fuel the same as its spent LEU fuel, by storing it in the reactor's spent fuel pool or in dry storage. According to the offeror, the amount of additional spent fuel is not expected to impact spent fuel management at the reactor sites.

Table 21. Total Additional Spent Fuel Assemblies Generated for the MOX Fuel Option

	Number of Spent Fuel Assemblies Generated with no MOX Fuel	Number of Additional Spent Fuel Assemblies with MOX Fuel	Percent Increase
S&D PEIS (based on a shorter fuel cycle)			
Typical PWR*	48/yr	32/yr	66.7%
Offeror's Reactors			
Total Over MOX Campaign	3,732	199	5.3%

*S&D PEIS pg. 4-734

For the four units at Catawba and McGuire, all of the additional spent nuclear fuel assemblies would be generated during the transition cycles from LEU to MOX fuel. Additional assemblies help to maintain peaking below design and regulatory limits, and compensate for the greater end-of-cycle reactivity. Once equilibrium is reached in the partial MOX core, additional fuel assemblies would not be required.

Like Catawba and McGuire, the North Anna units are expected to require additional LEU assemblies during the first transition cores. However, additional assemblies will also be required during equilibrium cycles because the smaller North Anna cores (157 fuel assemblies compared to 193 each for the McGuire and Catawba units) are more prone to neutron leakage and provide less flexibility with respect to meeting power peaking limits.

As designs are finalized and optimized for MOX fuel it may be possible to reduce MOX fuel assembly peaking and thereby reduce the number of additional assemblies required (and spent fuel generated) at the proposed reactors. As it currently stands, the North Anna site could generate approximately 16 percent more spent fuel by using MOX fuel than if the plants continued to use LEU fuel. The total amount of additional spent fuel generated by all six proposed reactors is estimated to be approximately 92 metric tons heavy metal. However, such MOX spent fuel is included in the inventory for the potential Nuclear Waste Policy Act geologic repository being studied by DOE. DOE is in the process of completing an environmental impact statement for a geologic repository.

4.2.5 Land Use

The offeror has stated that the proposed reactor sites would not require any additional land to support the use of MOX fuel in their reactors. This statement is consistent with information presented in the S&D PEIS. (S&D PEIS, pg. 4-720)

4.2.6 Infrastructure Requirements

The offeror has stated that the proposed reactor sites would not require any additional infrastructure to support the use of MOX fuel in their reactors. This statement is consistent with information presented in the S&D PEIS. (S&D PEIS, pg. 4-721)

4.2.7 Air Quality

Continued operation of the proposed reactor sites would result in a small amount of nonradiological air pollutants being released to the atmosphere, mainly due to the requirement to periodically test emergency diesel generators. The estimated air pollutants resulting from operation of the proposed reactors would not be expected to increase due to the use of MOX fuel in these reactors. Table 22 shows the estimated air pollutant concentrations and the national standards for these pollutants at the proposed sites. The impact of radiological releases is included in Section 4.2.1.

Table 22. Nonradiological Ambient Air Pollutant Concentrations with or without MOX Fuel from the Continued Operation of the Proposed Reactors

	Carbon Monoxide 8 hour 1 hour	Nitrogen Dioxide Annual	PM_{10} Annual 24 hour	Sulfur Dioxide Annual 24 hour 3 hour
National Ambient Air Quality Standards ($\mu g/m^3$)	10,000 40,000	100	50 150	80 365 1,300
Catawba ($\mu g/m^3$)	978 1400	3.26	0.102 65.4	0.0418 26.9 60.4
McGuire ($\mu g/m^3$)	1060 1510	2.6	0.08 71.2	0.03 29.9 67.4
North Anna ($\mu g/m^3$)	416 594	0.01	0.004 15.4	0.02 63 142

4.2.8 Water Quality

The offeror stated that there would be no change in water usage or discharge of nonradiological pollutants resulting from use of MOX fuel in the proposed reactors. Each of the reactor sites discharges nonradiological wastewater in accordance with a National Pollutant Discharge Elimination System

(NPDES) Permit, or an analogous state-issued permit. Permitted outfalls discharge conventional and priority pollutants from the reactor and ancillary processes that are similar to discharges from most reactor sites. Discharge Monitoring Reports (DMRs) for North Anna (May 1994 through April 1998) and Catawba (calendar years 1995 through 1997) showed that for the most part, there were only occasional noncompliances with permit limitations, only one of which occurred at an outfall receiving reactor process discharges. (The offeror did not provide DMRs for McGuire.) During the period reviewed, Catawba experienced four noncompliances, two in 1995 and two in early 1996. North Anna has exceeded the chlorine limitation at its sewage treatment facility, but this would neither affect nor be affected by, the use of MOX fuel. The impact of radiological releases is included in Section 4.2.1.

4.2.9 Socioeconomics

The offeror has stated that the proposed reactor sites would not need to employ any additional workers to support the use of MOX fuel in their reactors so there would not be any expected socioeconomic impacts. This statement is consistent with information presented in the S&D PEIS which concluded that the use of MOX fuel could result in small increases in the worker population at the reactor sites (between 40 and 105), but that any increase would be filled from the area's existing workforce. Therefore, there would be little impact on the local economy and communities (S&D PEIS, pgs. 4-727).

5.0 REQUIRED PERMITS AND LICENSES

Both the MOX fabrication facility and the selected reactors will require permitting and licensing activities to support the proposed fabrication and use of MOX fuel. The MOX fabrication facility will be constructed and operated at an existing DOE-owned site, but will be licensed by the NRC. The selected reactors are all U.S. operating, commercial PWRs, licensed by the NRC. The MOX facility, in particular, has special licensing considerations apart from most facilities that are built and operated in the United States today. This section discusses the particular licensing and permitting requirements of both facilities.

Both DOE and NRC have their origins in the Atomic Energy Act (AEA). The AEA first established their predecessor agency, the Atomic Energy Commission (AEC) to promote and regulate the use of atomic energy in the United States. The AEC was subsequently split into two organizations that have since become DOE and NRC. DOE was authorized to manage defense-related nuclear activities, while NRC was given the responsibility of regulating civilian uses of nuclear materials. Both DOE and NRC publish their regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), with NRC publishing in Parts 0–199, and DOE, Parts 200–1099. DOE supplements its regulations with a series of Orders, while NRC uses Regulatory Guides to further establish specific methods of implementation of its regulations. The proposed actions that are the subject of this Synopsis are unique in that DOE and NRC each have regulatory responsibility for certain parts of the activities.

The AEA authorizes DOE to establish standards to protect health or minimize dangers to life or property for activities under DOE's jurisdiction. Through a series of DOE orders and regulations, an extensive system of standards and requirements has been established to ensure safe operation of facilities. The DOE orders have been revised and reorganized to reduce duplication and eliminate obsolete provisions (though some older orders remain in effect during the transition). For DOE orders, the new organization is by Series and is generally intended to include all DOE policies, manuals, requirements documents, notices, guides, and orders. For proposed actions involving fuel qualification, relevant DOE regulations include 10 CFR 820, Procedural Rules for DOE Nuclear Activities; 10 CFR 830, Nuclear Safety Management; 10 CFR.834, Radiation Protection of the Public and the Environment (Draft); 10 CFR 835, Occupational Radiation Protection; 10 CFR 1021, Compliance with the National Environmental Policy Act; and 10 CFR 1022, Compliance with Floodplains/Wetlands Environmental Review Requirements. DOE orders include those in new Series 400, which deals with Work Process; and within this Series, DOE Order 420.1 addresses Facility Safety; 425.1 addresses Startup and Restart of Nuclear Facilities; 452.1A addresses Nuclear Explosive and Weapons Surety Programs; 452.2A addresses the Safety of Nuclear Explosives Operations; 452.4 addresses the Security and Control of Nuclear Explosives; 460.1A addresses Packaging and Transportation Safety; 470.1 addresses the Safeguards and Security Program; and 474.1 addresses the Control and Accountability of Nuclear Materials. In addition, DOE (older number) Series 5400 addresses environmental, safety, and health programs for DOE operations. Not all of these DOE regulations and orders would apply to operation of the proposed MOX fuel fabrication facility, and most would not apply to use of the proposed reactors.

There are a number of Federal environmental statutes dealing with environmental protection, compliance, or consultation. In addition, certain environmental requirements have been delegated to state authorities for enforcement and implementation. Certain statutes and regulations require DOE to consult with Federal, State, and local agencies and federally recognized Native American groups. Most of these consultations are related to biotic resources, cultural resources, and Native American resources. Biotic resources consultations generally pertain to the potential for activities to disturb sensitive species or habitats. Cultural resources consultations relate to the potential for disruption of important cultural resources and archaeological sites. Finally, Native American consultations are concerned with the potential for disturbance of Native American sites and resources. DOE has conducted appropriate consultations at the candidate sites and will report the results of these consultations in the SPD Final EIS.

It is DOE policy to conduct its operations in an environmentally safe manner in compliance with all applicable statutes, regulations, and standards. Although this chapter does not address pending or future regulations, DOE recognizes that the regulatory environment is subject to change, and that the construction, operation, and decommissioning of any surplus plutonium disposition facility must be conducted in compliance with all applicable regulations and standards.

5.1 Regulatory Activities

It is likely that new or modified permits will be needed before the proposed surplus plutonium disposition facilities may be constructed or operated. Permits regulate many aspects of facility construction and operations, including the quality of construction, treatment and storage of hazardous waste, and discharges of effluents to the environment. These permits will be obtained from appropriate Federal, state, and local agencies. NRC issues operating licenses for major facilities such as commercial nuclear power reactors and fuel fabrication facilities, although the regulations under which these two facilities would be licensed are different.

5.1.1 The MOX Facility

The MOX facility would be licensed to operate by NRC under its regulations at 10 CFR 70, *Domestic Licensing of Special Nuclear Materials*. Because the facility would be located at a DOE site, however, certain DOE requirements affecting site interfaces and infrastructure will also be applicable. In addition, as would be the case regardless of where the facility were built, Federal or state regulations implementing certain provisions of the Clean Water Act, Clean Air Act, and Resource Conservation and Recovery Act would be applicable. These regulations are implemented through permits. Evaluation would be required to determine whether MOX facility emissions and activities would necessitate modification of any of these permits. Analyses in the SPD Draft EIS have shown that there would be minimal impact from construction and operation of the MOX facility.

MOX facility design and operating parameters will be imposed by requirements of 10 CFR 70. Facility robustness, worker health and safety, and material and personnel security are all specified by 10 CFR 70. This regulation incorporates and refers the licensee to provisions of other NRC regulations such as those found at 10 CFR 20, *Radiation Protection Standards.* Safety and environmental analyses will be required to support the license application for the MOX facility.

Integral to the NEPA process is consideration of how the proposed action might affect biotic, cultural, and Native American resources, and the need for mitigation of any potential impacts. Required consultations with agencies and recognized Native American groups have been conducted.

5.1.2 Reactors

Nuclear power reactors undergo a lengthy licensing process under 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*, beginning before facility construction commences. This process includes preparation of safety analysis and environmental reports. The safety analysis report remains a living document that serves as the licensing basis for the plant, and is updated throughout the life of the plant. Public hearings before a licensing board are conducted prior to a license being issued. Once issued, operating licenses may be amended only with proper evaluation, review and approval as specified in 10 CFR 50.90. This prescriptive process requires demonstration that a proposed change does not involve an unreviewed environmental or safety question and provides for public notice and opportunity to comment prior to issuance of the license amendment. Minor license amendments can be processed fairly expeditiously, but more involved amendments can require multiple submittals before the NRC is assured that the proposed action will not reduce the margin of safety of the plant. All submittals, except portions that contain proprietary information, are available to the public.

The regulatory process for requesting reactor license amendments to use MOX fuel will be the same as for any 10 CFR 50 Operating License amendment request. The reactor licensee submitting an operating license amendment request in accordance with 10 CFR 50.90 initiates this process. Safety and environmental analyses commensurate with the level of potential impact are submitted in support, and as part, of the amendment request. NRC reviews the submitted information and denies or approves the request. The review process can involve submittal of additional information and face-to-face meetings between the licensee and NRC, and can result in modified license amendment requests. NRC provides notice in the *Federal Register* for certain steps in the process. The notice for the amendment request initially appears in the *Federal Register* with a Notice of Opportunity for Public Hearing. *Federal Register* notices are also required for the Proposed No Significant Hazards Determination, associated environmental documents, Consideration of Issuance of the License Amendment, and issuance of the final amendment. Certain of these notices allow for the opportunity to provide written comments, and for potentially affected parties to petition to intervene or request public hearings.

The six reactors proposed to use MOX fuel have been operating for a number of years. Revisions to each of their operating licenses will be required prior to MOX fuel being brought to the reactor sites and loaded into the reactors. The license amendment request will need to include a discussion of all potential impacts and changes in reactor operation that could be important to safety or the environment. This will include fresh and spent fuel handling, security and operational changes, as well as complete core load analysis and safety analyses, including potential changes to the severe accident analyses. Because the offeror has indicated that no new construction would be required to accommodate the use of MOX fuel, it is unlikely that any biotic, cultural or Native American resources would be impacted by the proposed action. The analyses performed for the Environmental Critique have demonstrated very little difference between the impacts from using a partial MOX core over a LEU core.

The need for modifications to site permits will be evaluated by the individual plants as part of their licensing activities. The offeror has indicated, and the analyses and reviews performed for the Environmental Critique, support the assertion, that there would be minimal or no change in effluents, emissions, and wastes (both radiological and nonradiological). Therefore, it is expected that few, if any, environmental permits or agreements will require modification for use of MOX fuel.

6.0 CONCLUSION

No major impacts to the environment surrounding the proposed MOX facility or reactor sites are expected to result from normal operation of these facilities. Environmental impacts from operation of the proposed reactors are not expected to change appreciably due to the use of MOX fuel. Impacts from construction and operation of the MOX facility are expected to be generally consistent with those presented in the SPD Draft EIS, and impacts at the reactor sites are expected to be generally consistent with those in the S&D PEIS.