Conceptual Safety Design Report Assessment for the Versatile Test Reactor

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Office of Enterprise Assessments
U.S. Department of Energy

Table of Contents

Acronyms

BEA	Battelle Energy Alliance, LLC
CFR	Code of Federal Regulations
CSDR	Conceptual Safety Design Report
CW	Co-located Worker
DBA	Design Basis Accident
DOE	U.S. Department of Energy
DOE-ID	DOE Idaho Operations Office
EA	Office of Enterprise Assessments
EG	Evaluation Guideline
MAR	Material at Risk
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
OFI	Opportunity for Improvement
PDSA	Preliminary Documented Safety Analysis
PRA	Probabilistic Risk Assessment
RG	Regulatory Guide
SBRT	Safety Basis Review Team
SC	Safety Class
SRL	Safety Review Letter
SS	Safety Significant
SSCs	Structures, Systems, and Components
VTR	Versatile Test Reactor

Conceptual Safety Design Report Assessment
for the Versatile Test Reactor
November 2019 – April 2020

Summary

Scope
This assessment evaluated the conceptual safety design report (CSDR) for the Versatile Test Reactor (VTR). The VTR is a proposed 300-megawatt, sodium-cooled reactor for testing and qualification of advanced nuclear fuels and materials. The location of the VTR will be determined upon completion of actions conducted under the National Environmental Policy Act. The assessment also included a review of the safety review letter (SRL) prepared by the U.S. Department of Energy (DOE) Idaho Operations Office.

Significant Results for Key Areas of Interest
The VTR CSDR complies with DOE-STD-1189-2016, *Integration of Safety into the Design Process*. The SRL complies with DOE-STD-1104-2016, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents*. The assessment did not identify any best practices, findings, or deficiencies.

<u>Conceptual Safety Design Report</u>
The CSDR adequately identifies and evaluates the facility-level hazards associated with VTR operations. The VTR design and safety analyses appropriately adopt several U.S. Nuclear Regulatory Commission (NRC) guidance documents and industry standards to develop a probabilistic risk assessment and to establish the reactor design criteria. The CSDR appropriately identifies a number of reactor and non-reactor (ex-vessel) safety class and safety significant structures, systems, and components, along with their safety functions and functional requirements, for reactor protection, prevention and mitigation of releases, radiological protection, and criticality prevention. The identified controls are suitable to address the analyzed hazards and are sufficient to support proceeding with preliminary design. Nevertheless, some reactor transients are precluded based on design assumptions that are not fully developed, and an ex-vessel accident progression for potential fire events involving fresh fuel is not fully justified based on conceptual design information. The approach to meeting the general design criteria of DOE Order 420.1C Chg 2, *Facility Safety,* is sufficiently described, with one exception related to the specific details in the VTR design that will facilitate deactivation, decommissioning, decontamination, and demolition.

<u>Safety Review Letter</u>
The SRL addresses the DOE-STD-1104-2016 approval bases and appropriately concludes that the CSDR is sufficiently conservative to support proceeding from the conceptual design phase to the preliminary design phase. The Safety Basis Review Team did not reproduce the detailed conceptual design analyses to verify their accuracy; however, this level of review is not required by DOE. Nonetheless, the significance of potential hazards and complexity of the VTR design suggest that an in-depth, independent verification of the final design and safety analysis, similar to an independent NRC licensing review, would help minimize safety and programmatic risks.

Follow-up Actions
The Office of Enterprise Assessments will continue to follow the evolution of safety-in-design for the VTR Project, including review of the VTR preliminary documented safety analysis and safety evaluation report, and the completion of actions undertaken to resolve comments by the assessment team on the CSDR.

Conceptual Safety Design Report Assessment
for the Versatile Test Reactor

1.0 INTRODUCTION

The U.S. Department of Energy (DOE) Office of Nuclear Safety and Environmental Assessments, within the Office of Enterprise Assessments (EA), conducted an assessment of the conceptual safety design report (CSDR) for the proposed Versatile Test Reactor (VTR). The location of the VTR will be determined following an alternatives analysis conducted under the National Environmental Policy Act (NEPA). For purposes of preparing the CSDR, the Materials and Fuel Complex at the Idaho National Laboratory was used as a representative location. The assessment also included a review of the DOE Idaho Operations Office (DOE-ID) safety review letter (SRL). This assessment, conducted from November 2019 through April 2020, is part of a series of ongoing targeted assessments of new DOE nuclear facility projects focusing on the adequacy of safety-in-design documents.

In accordance with the *Plan for the Office of Enterprise Assessments Assessment of the Conceptual Safety Design Report for the Versatile Test Reactor at the Idaho National Laboratory, November 2019 - April 2020*, this assessment evaluated the VTR CSDR and SRL against the requirements of DOE-STD-1189-2016, *Integration of Safety into the Design Process*; DOE-STD-1104-2016, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents*; and DOE Order 420.1C Chg 2, *Facility Safety*.

The VTR Project is sponsored by the DOE Office of Nuclear Energy (NE) and is supported by a team of national laboratories, universities, and subcontractors led by Battelle Energy Alliance, LLC (BEA) as the contractor design agency. DOE-ID provides oversight of BEA, and NE provides programmatic direction. The VTR Project entails the design and construction of a hazard category 1, 300-megawatt, sodium-cooled, fast-spectrum test reactor that provides high levels of fast neutron flux for testing and qualification of advanced nuclear fuels and materials. The VTR conceptual design is derived from the Power Reactor Innovative Small Module (PRISM) technology. The VTR Project is on an accelerated schedule, with a planned start of operations in 2025.

2.0 METHODOLOGY

The DOE independent oversight program is described in and governed by DOE Order 227.1A, *Independent Oversight Program*, which is implemented through a comprehensive set of internal protocols, operating practices, assessment guides, and process guides. This report uses the terms "best practices, deficiencies, findings, and opportunities for improvement (OFIs)" as defined in the order.

As identified in the assessment plan, this assessment considered the requirements of DOE-STD-1189-2016 and DOE-STD-1104-2016 for VTR CSDR development and Federal review, respectively. Key aspects of these requirements are provided in Criteria and Review Approach Document EA 31-29, Rev. 1, *Review of Nuclear Facility Safety Design Basis Development*.

In addition to the CSDR and SRL, the assessment team examined key supporting documents, including hazard analysis and engineering documents. The assessment included review and evaluation of the functional classifications, safety functions, and functional requirements assigned to safety structures, systems, and components (SSCs). The assessment team provided comments to and conducted conference calls with the BEA personnel and DOE-ID Safety Basis Review Team (SBRT) members responsible for developing and reviewing the CSDR. Appendix A lists the members of the assessment team, the Quality Review Board, and responsible EA management.

There were no items for follow-up during this assessment.

3.0 RESULTS

3.1 Conceptual Safety Design Report

The nuclear safety management rule (10 CFR 830, *Nuclear Safety Management*), Appendix A to Subpart B, identifies U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, as an acceptable methodology for preparing a reactor documented safety analysis. Given the distinction between the design of the VTR and the types of reactors for which NRC RG 1.70 was developed, the VTR safety analysis report will be formatted after NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, and adapted for the VTR design and DOE regulatory environment to ensure compliance with DOE requirements not covered by NUREG-1537, such as facility hazard categorization, specific radiological and chemical consequence limits, and designation of safety systems. In addition to DOE Order 420.1C general design criteria, which are not specifically tailored for a reactor, the VTR conceptual design addresses the sodium fast reactor design criteria identified in NRC RG 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*. A risk-informed process is implemented to address the sodium fast reactor design criteria and to guide the VTR design. This process includes the use of probabilistic risk assessment (PRA), which is described in a VTR PRA plan developed in accordance with DOE-STD-1628-2013, *Development of Probabilistic Risk Assessments for Nuclear Safety Applications*, to develop technology-inclusive, risk-informed, and performance-based principal design criteria consistent with Nuclear Energy Institute guidance document NEI 18-04, *Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development*. The use of NRC guidance documents and industry standards is supported by DOE regulations and is appropriate for the development of the VTR design and safety analysis.

3.1.1 Facility Design and Process Descriptions

The objective of the assessment of the facility design and process descriptions was to evaluate whether the level of detail in the CSDR supports facility-level hazard analyses. The assessment also sought to evaluate the adequacy of and basis for the identified design criteria. The CSDR describes the facility background and mission. The site description is brief because Idaho National Laboratory has been identified as a representative location for the purposes of the CSDR. The final location will be determined after the completion of a NEPA alternatives analysis. The descriptions of the VTR design, support systems, and processes are sufficient to support an overall understanding of the general facility arrangement relative to the analyzed hazards in the CSDR. Assumptions regarding material at risk (MAR) are conservatively based on bounding values for both reactor and non-reactor (ex-vessel) operations. The MAR is appropriately described in terms of quantity, form, and location. The facility and site information is adequate to support the scoping analyses, including the facility-level hazard analyses, in the CSDR.

The CSDR provides a crosswalk for the VTR approach to implementing DOE Order 420.1C general design criteria. With the exception of the lack of a specific discussion regarding the ability to deactivate, decommission, decontaminate, and demolish the reactor, the approach to meeting the applicable criteria of DOE Order 420.1C is adequately described.

The CSDR provides a high-level discussion of safety-in-design risks and opportunities, with a primary focus on cost and schedule. A reference is provided for the detailed risks and opportunities, which adequately addresses DOE-STD-1189-2016 expectations.

3.1.2 Hazard Analysis

The objective of the assessment of the hazard analysis was to evaluate hazard identification and evaluation, including the designation of hazard controls. The VTR PRA evaluates elements pertinent to the hazard analysis, including initiating event analysis, event sequence analysis, radiological consequence analysis, and identification of success criteria for designation of VTR safety functions and systems.

The VTR hazard categorization of hazard category 1 is appropriate per DOE-STD-1027-2018, *Hazard Categorization of DOE Nuclear Facilities*.

3.1.2.1 Hazard Identification

The CSDR identifies hazards associated with both reactor transients and ex-vessel operations and includes operational hazards. The reactor transients cover transient overpower, loss of heat sink, loss of flow, mishandling or malfunction of equipment, experiment malfunction, and mishandling or malfunction of fuel. The ex-vessel operational hazards include inadvertent criticality external to the reactor, fires/explosions, radioactive material release from casks and storage pit operations, direct radiation exposure, and chemical releases from the sodium that serves as the primary and secondary coolant.

The identified hazards pertinent to reactor transients are limited to events occurring during power operation either as a direct result of equipment or operator failure during normal operations, maintenance, or testing. The hazards are grouped into initiating events in the VTR PRA. External and natural phenomena hazards, as well as initiating events during shutdown, are out of scope for the conceptual design PRA. Limited scoping analyses of sodium fires and seismic hazards are provided in the CSDR and Preliminary Fire Hazards Analysis.

The CSDR screens several hazards from further evaluation (e.g., core blockages) with the anticipation that the final design will preclude such hazards. EA's assessment of the VTR preliminary documented safety analysis (PDSA) will evaluate the extent to which the final design is consistent with such expectations.

The CSDR adequately identifies and categorizes the hazards associated with reactor transients and ex-vessel operations. The identified VTR transient and ex-vessel hazards represent a set of bounding hazards that adequately supports facility-level hazard analyses, derivation of the suite of safety SSCs, and advancement to preliminary design.

3.1.2.2 Hazard Evaluation

The reactor hazard evaluation (and subsequent control selection) is based on the development of PRA event trees for the initiating events. Each event tree provides a time-independent, system-based response (PRA event sequence) to each initiating event. Radiological consequences are evaluated for the public and co-located worker (CW) for each event sequence end-state that results in a release. Safety basis events are developed from all of the PRA event sequences to demonstrate compliance with frequency-consequence evaluation limits. The consequence limits are consistent with the DOE public Evaluation Guideline (EG) of 25 rem and the CW threshold of 100 rem. Safety basis events identified in the PRA process are further used to define design basis accidents (DBAs) for transient safety analysis.

The ex-vessel hazard evaluation provides qualitative estimates of frequency, as well as a combination of qualitative and quantitative dose consequence estimates. The analysis covers the identified operational and seismic hazards for a range of fuels, experiments, and waste operations. DBAs are developed to envelope the potential hazards.

The reactor transient analyses and ex-vessel accident analyses use conservative assumptions for accident progression and analysis parameters, including MAR, airborne release fractions, respirable fractions, damage ratios, and public and CW atmospheric dispersion factors. The only exception is the ex-vessel accident analysis of a fire involving fresh fuel, which relies on fire severity limitations and fuel response that excludes rapid oxidation of the fuel and associated significant releases; such exclusion is not fully justified based on conceptual design information. The detailed reactor transients, ex-vessel accident analyses, and derived controls will be updated as the design matures to support PDSA development. These updates will be reviewed during the PDSA assessment. Overall, the reactor and ex-vessel hazard evaluations are adequate for this conceptual level of design maturity.

3.1.3 Hazard Controls

The objective of the assessment of hazard controls was to evaluate their derivation, classification, safety functions, and functional requirements. The CSDR identifies a suite of safety class (SC) and safety significant (SS) controls based on a set of criteria important to reactor integrity, frequency and dose consequences, and facility-level hazard analysis of supporting operations. The criteria for designation of safety controls include safety basis event violation of the offsite and onsite frequency-consequence curves for reactor transients, exceedance of the EG or CW threshold for the deterministically evaluated DBAs, protection from chemical hazards, importance for protection of the primary coolant boundary, importance for reactor shutdown, and performance of risk significant functions. Additionally, SS and SC controls are identified based on the integrated decision panel and the safety design integration team review to address uncertainties or assumptions within the PRA analysis for specific high consequences.

The safety controls designation criteria are consistent with DOE requirements and are suitably derived from the PRA safety basis event evaluations and deterministic DBA analyses. The CSDR appropriately identifies a number of reactor and ex-vessel SC and SS SSCs, along with their safety functions and functional requirements, for reactor protection, prevention and mitigation of releases, radiological protection, and criticality prevention. As expected for a CSDR, no specific administrative controls are identified. The identified controls are adequate to address the hazards and are sufficient to support proceeding with preliminary design.

3.1.4 Conceptual Safety Design Report Conclusion

The CSDR meets the requirements of DOE-STD-1189-2016 and adequately identifies and evaluates the hazards associated with VTR operations, including reactor transients and ex-vessel operations. The approach to meeting the general design criteria of DOE Order 420.1C is adequately described in the CSDR, with the exception of a specific discussion regarding the ability to deactivate, decommission, decontaminate, and demolish the reactor. The CSDR and supporting references include an appropriate consideration of risks and opportunities. The hazard analysis postulates an adequate set of hazard events and appropriately addresses significant reactor transients, hazardous materials, and energy sources. The identified controls are suitable to address the analyzed hazards and are sufficient to support proceeding with preliminary design. Nevertheless, some reactor transients are precluded based on design assumptions that are not fully developed (e.g., core blockages), and one accident progression is not fully justified based on conceptual design information (i.e., a fire involving fresh fuel). The detailed reactor transients, ex-vessel accident analyses, and derived controls will be reviewed during the assessment of the PDSA.

3.2 Federal Review and Approval

The assessment team reviewed the SRL to determine its adequacy as the approval basis for the CSDR per the requirements of DOE-STD-1104-2016. The DOE-ID SBRT reviewed the CSDR and prepared the SRL in accordance with its approved plan. The SBRT did not reproduce the detailed conceptual design analyses to verify their accuracy. DOE-STD-1104-2016 does not provide specific requirements regarding the level of detail of the SBRT review. (See **OFI-DOE-ID-1**.)

The SBRT concluded that the CSDR adequately implements the applicable requirements of DOE-STD-1189-2016 and provides sufficient information to support proceeding to preliminary design. The SRL appropriately concludes that the CSDR provides a technically sound and reasonable basis for safety-in-design and is sufficient to support proceeding from the conceptual design phase to the preliminary design phase.

4.0 BEST PRACTICES

There were no best practices identified as part of this assessment.

5.0 FINDINGS

There were no findings identified as part of this assessment.

6.0 DEFICIENCIES

There were no deficiencies identified as part of this assessment.

7.0 OPPORTUNITIES FOR IMPROVEMENT

The assessment team identified an OFI to assist cognizant managers in improving programs and operations. While OFIs may identify potential solutions to findings and deficiencies identified in assessment reports, they may also address other conditions observed during the assessment process. This OFI is offered only as a recommendation for line management consideration; it does not require formal resolution by management through a corrective action process and is not intended to be prescriptive or mandatory. Rather, it is a suggestion that may assist site management in implementing best practices or provide potential solutions to issues identified during the assessment.

DOE Idaho Operations Office

OFI-DOE-ID-1: Given the significance of potential hazards and the complexity of the VTR design, an in-depth, independent verification by the SBRT of the final design and safety analysis, similar to an independent NRC licensing review, would help minimize safety and programmatic risks.

8.0 FOLLOW-UP ITEMS

All but three comments identified by the assessment team were resolved. EA will review the actions undertaken to address the three comments during the assessment of the PDSA and safety evaluation report. The specific topics in the CSDR that the three comments address can be summarized as follows:

- Adequacy of the VTR design to meet the DOE Order 420.1C criteria regarding the ability to deactivate, decommission, decontaminate, and demolish the reactor

- Implementation of designs that are assumed to preclude the need to analyze some reactor transients that are not currently evaluated in the CSDR

- Completeness of ex-vessel accident sequences and use of sufficiently conservative assumptions for potential fire events involving fresh fuel.

Appendix A
Supplemental Information

Dates of Assessment
Document Review: November 2019 – April 2020

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