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1 THE FACILITY

1.1 Introduction

This safety evaluation (SE) documents the results of the U.S Nuclear Regulatory Commission (NRC) staff's (staff) technical review of the construction permit (CP) application submitted by TerraPower, LLC (TerraPower) on behalf of U.S. SFR Owner, LLC (USO), under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," for the Kemmerer Power Station Unit 1 (KU1) power reactor proposed to be built in Lincoln County, Wyoming. The proposed reactor is a non-light-water reactor (non-LWR). An environmental review was also performed for the KU1 CP application, and its evaluation and conclusions are documented in a Final Environmental Impact Statement (FEIS), published on October 21, 2025, as NUREG-2268 "Environmental Impact Statement for the Construction Permit Application for Kemmerer Power Station Unit 1" (90 FR 48507).

The staff acknowledged receipt of USO's application for a CP by letter on May 14, 2024 (ML24127A183) and published notice in the Federal Register (FR) on May 21, 2024 (89 FR 44715):

- Description and safety assessment of the site required by 10 CFR 50.34, "Contents of applications; technical information," paragraph (a)(1).
- Environmental report submitted to meet the requirements of 10 CFR 50.30, "Filing of applications for licenses; oath or affirmation," paragraph (f).
- General information submitted to meet the requirements of 10 CFR 50.33, "Contents of applications; general information."

The staff conducted an acceptance review for docketing of USO's KU1 CP application and, by letter dated May 21, 2024 (ML24135A109), determined that USO's KU1 CP application was complete and acceptable for docketing. The application was assigned Docket Number (No.) 50-613. A notice of docketing of USO's KU1 CP application was published in the FR on June 4, 2024 (89 FR 47997).

The staff's safety evaluation of the CP application to construct the 10 CFR Part 50 utilization facility is based on information in the application, as revised and supplemented. Unless otherwise stated, this SE evaluates the information contained in the original application dated March 28, 2024 (ML24088A060); as supplemented by Rev 1, dated October 3, 2025 (ML25276A288) and supplements dated June 17, 2025 (ML25171A021) "Transmittal of Probabilistic Site Response Analysis Calculation of Ground Motion Response Spectra and Safe Shutdown Earthquake Spectra"; September 9, 2025 (ML25251A127) "Revision to Exemption Request from 10 CFR 50.33(f) and 10 CFR 50 Appendix C Financial Qualification Documentation Requirements and Revision to the General and Financial Information"; September 15, 2025 (ML25259A175) "Intermediate Heat Exchanger Tube-to-Tubesheet Welds Design Information"; September 16, 2025 (ML25259A180) "Request for Confirmation of Information"; September 17, 2025 (ML25260A002) "Response to NRC Audit Question 3-85"; October 3, 2025 (ML2576A027) "DOE Confirmation of Active and Good Faith Negotiations for

Disposal Contract Letter to USO”; October 1, 2025 (ML25274A130) “Natrium Demonstration Plant Long-Lived Passive Structural Materials of Construction Selection and Development”; October 1, 2025 (ML25274A124) “Research and Development Supplemental Information”; September 10, 2025 (ML25253A386) “Preventive Measures Classification Methodology and Preliminary Results”; July 23, 2025 (ML25205A087) Transmittal of TerraPower, LLC, “Natrium Demonstration DID Evaluation Report,” NAT-4770 Revision 1; April 7, 2025 (ML25108A080) “Transmittal of Response to NRC Audit Question 3-77 on KU1 PSAR”; January 23, 2025 (ML25028A117) “Transmittal of Responses to NRC Audit Questions on KU1 PSAR”; January 3, 2025 (ML25003A162) “Regulatory Interpretation of the Applicability of 10 CFR 50.10 and 10 CFR 51.4 Definitions of Activities Constituting Construction to the Installation of Conduit and Cable Trays”; December 19, 2024 (ML25016A155) Transmittal of Responses to NRC Audit Questions on KU1 PSAR”; October 28, 2024 (ML24310A087) Transmittal of Responses to NRC Audit Questions on KU1 PSAR”; and September 6, 2024 (ML24253A220) “Transmittal of Responses to NRC Audit Questions on KU1 PSAR”.

1.1.1 Areas of Review

The KU1 CP application review consisted of two concurrent reviews: (1) a safety review of the KU1 preliminary safety analysis report (PSAR) and supporting technical information, and (2) an environmental review of the KU1 Environmental Report. The staff reviewed the KU1 PSAR and supporting technical information against applicable regulatory requirements using appropriate regulatory guidance and standards, as discussed below, to assess the sufficiency of the preliminary design of the KU1 power reactor. As part of this review, the staff evaluated descriptions and discussions of KU1’s structures, systems, and components (SSCs), with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff also evaluated the preliminary design of KU1 to ensure the sufficiency of principal design criteria (PDC), design bases, and information relative to materials of construction, general arrangement, and approximate dimensions to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. In addition, the staff reviewed USO’s identification and justification for the selection of those variables, conditions, or other items that USO determined to be probable subjects of technical specifications for the facility in accordance with 10 CFR 50.34(a)(5). The staff also reviewed USO’s evaluation of the SSCs to ensure that they would adequately provide for the prevention of accidents and the mitigation of consequences of accidents. The staff considered the preliminary analysis and evaluation of the design and performance of the SSCs of the KU1 facility which the applicant prepared with the objective of assessing the risk to public health and safety resulting from operation of the facility.

1.1.2 Regulatory Basis, Acceptance Criteria, and Exemptions

The staff reviewed the KU1 PSAR and supporting technical information against applicable regulatory requirements, using appropriate regulatory guidance and standards, to assess the sufficiency of the preliminary facility design and analysis for the issuance of a CP.

In accordance with paragraph (a) of 10 CFR 50.35, “Issuance of construction permits,” when an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a CP which approves all proposed design features, the Commission may issue a CP if the Commission finds that:

1. the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;
2. such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report (FSAR);
3. safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that
4. on the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, "Reactor Site Criteria," the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

Paragraph (a) of 10 CFR 50.34 also provides that the above findings will be modified accordingly when the applicant has supplied initially all of the technical information required to complete the application.

As provided in 10 CFR 100.2, "Scope," the siting requirements in 10 CFR Part 100 "apply to applications for site approval for the purpose of constructing and operating stationary power and testing reactors pursuant to the provisions of [10 CFR Part 50]." The KU1 application is for a CP for a stationary power reactor. Therefore, the staff evaluated the characteristics of the proposed KU1 site using the applicable criteria in 10 CFR Part 100, in addition to those in 10 CFR Part 50. The staff's review evaluated the geography and demography of the site; nearby industrial, transportation, and military facilities; site meteorology; site hydrology; and site geology, seismology, and geotechnical engineering to determine whether issuance of the CP would be inimical to the public health and safety. The staff's review also evaluated SSCs and equipment designed to ensure safe operation, performance, and shutdown when subjected to extreme weather, floods, seismic events, missiles (including aircraft impacts), chemical and radiological releases, and loss of offsite power.

The CP, if issued, would constitute an authorization for USO to proceed with construction but would not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated into the permit. USO did not request such approval here. Such approval, if appropriate, would be made following the evaluation of the final design of the facility, as described in the FSAR as part of USO's operating license (OL) application for KU1, should the applicant apply for an OL.

In addition to the findings listed in 10 CFR 50.35, a CP application must also provide sufficient information to allow the Commission to make the following determinations in accordance with 10 CFR 50.40, "Common standards," and 10 CFR 50.50, "Issuance of licenses and construction permits":

1. There is reasonable assurance: (i) that the construction of the facility will not endanger the health and safety of the public, and (ii) that construction activities can be conducted in compliance with the Commission's regulations.
2. The applicant is technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
3. The applicant is financially qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
4. The issuance of a permit for the construction of the facility would not be inimical to the common defense and security or to the health and safety of the public.
5. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of this CP, subject to the conditions for protection of the environment set forth herein, is in accordance with Subpart A of 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
6. The application meets the standards and requirements of the Atomic Energy Act and the Commission's regulations, and that notifications, if any, to other agencies or bodies have been duly made.

The staff's evaluation of the KU1 preliminary design and analysis was based primarily upon the following 10 CFR requirements:

- 10 CFR 50.30, "Filing of application; oath or affirmation"
- 10 CFR 50.33, "Contents of applications; general information"
- 10 CFR 50.34, "Contents of applications; technical information" particularly 10 CFR 50.34(a), "Preliminary safety analysis report"
- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors"
- 10 CFR 50.35, "Issuance of construction permits"
- 10 CFR 50.40, "Common standards"
- 10 CFR 50.50, "Issuance of licenses and construction permits"
- 10 CFR 50.55, "Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses"
- 10 CFR 50.55a, "Codes and standards"
- 10 CFR 50.150, "Aircraft impact assessment"
- 10 CFR Part 20, "Standards for Protection against radiation"
- 10 CFR Part 26, "Fitness for duty programs," Subpart K, "FFD [Fitness for Duty] Programs for Construction"
- 10 CFR Part 100, "Reactor site criteria"

The regulations of 10 CFR 50.40, "Common standards," require that:

the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply

with the regulations in this chapter, including the regulations in part 20 of this chapter, and that the health and safety of the public will not be endangered.

With respect to 10 CFR Part 20, which is referred to in 10 CFR 50.40, the staff assessed whether USO had identified the relevant requirements for an operating facility and provided descriptions of the preliminary facility design to determine whether the PSAR provides an acceptable basis for the development of SSCs, and whether there is reasonable assurance that USO will comply with the regulations in 10 CFR Part 20 during KU1 facility operation. Because USO has not applied for licenses to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material in accordance with 10 CFR Part 30, "Rules of general applicability to domestic licensing of byproduct material," 10 CFR Part 40, "Domestic licensing of source material," and 10 CFR Part 70, "Domestic licensing of special nuclear material," respectively, or a license to operate a production or utilization facility under 10 CFR Part 50, the requirements of 10 CFR Part 20 do not apply at this time. As such, the staff did not evaluate the application against the requirements in 10 CFR Part 20, but such evaluation would occur for the OL application, should the applicant apply for an OL.

As required by 10 CFR 50.34(a)(3)(i), USO must describe the PDCs for KU1 in the PSAR. PDCs for a Natrium Advanced Reactor were developed by TerraPower in topical report (TR) NATD-LIC-RPRT-0002-A, which the NRC approved (ML24283A066). This TR is incorporated by reference in PSAR section 5.3 which repeats the PDCs from the approved TR and provides additional contextual information on how the PDCs are implemented in the KU1 design. The staff's evaluation of this incorporation by reference is in Section 5.3 of this SE.

USO used the licensing modernization project (LMP) methodology described in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development" (ML19241A472), as endorsed by the NRC in Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors" (ML20091L698). These guidance documents define risk-informed, performance-based, and technology-inclusive processes for the selection of licensing basis events (LBEs); safety classification of SSCs; and the determination of defense-in-depth (DID) adequacy for non-light-water reactors (non-LWRs). NEI 18-04 provides a frequency consequence target curve that is used to assess events, SSCs, and programmatic controls. LBEs are categorized by the frequency of occurrence, separated into anticipated operational occurrences, design-basis events (DBEs), and beyond-design-basis events. Because the LMP methodology is a novel approach used for the first time for a commercial power reactor in this application, an orientation to the NEI 18-04 process and how it is reflected in the structure of the PSAR is provided in SE section 1.3.3.3.1 below.

USO followed the format of NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology," Revision 1 (ML22060A190) to develop portions of the KU1 PSAR. NEI 21-07, Revision 1, describes the scope and level of detail in specific portions of the first eight chapters of a PSAR that are associated with LMP-based safety analysis. The staff endorsed NEI 21-07, Revision 1, as one acceptable approach to develop portions of the first eight chapters of the PSAR in RG 1.253 "Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Advanced Reactors" (ML23269A222).

DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications – Roadmap" (ML23277A139) provides guidance for the staff review of new non-LWR applications following the LMP methodology. Specifically, it includes (1) a general overview of the information that should be included in a non-LWR application submitted under 10 CFR Part 50 or 10 CFR Part 52; (2) a review roadmap for NRC staff with the principal purpose of ensuring consistency, quality, and uniformity of the staff reviews; and (3) a well-defined base from which the staff can evaluate proposed differences in the scope of reviews (e.g., CP versus OL). RGs 1.233 and 1.253 as well as other key guidance documents are referenced in DANU-ISG-2022-01.

As appropriate, the staff used additional guidance and codes and standards (e.g., NRC RGs, Institute of Electrical and Electronics Engineers (IEEE) standards, and American National Standards Institute/American Nuclear Society (ANSI/ANS) standards) in its review of the KU1 CP application.

Exemptions

Enclosure 4 of USO's KU1 CP application included four exemption requests associated with 10 CFR Part 50 regulations. The requested exemptions are described and addressed in appendix B and chapter 14 of this SE.

1.1.3 Review Procedures

USO's KU1 CP application only seeks authorization to construct the proposed KU1 facility. Accordingly, the KU1 design may be adequately described at a functional or conceptual level in the PSAR. As stated throughout the PSAR, USO will include additional design and analysis details with its FSAR as part of its OL application, should it apply for an OL.

The objective of the staff's evaluation was to assess the sufficiency of information contained in the KU1 application for the issuance of a CP in accordance with the requirements of 10 CFR Part 50. An in-depth evaluation of the KU1 design will be performed following the docketing of an OL application and its accompanying FSAR, should USO apply for an OL.

1.1.4 Resolving Technical Issues

The staff conducted an audit to maximize the efficiency of the staff's review. The audit enabled the staff to gain a more detailed understanding of USO's application and identify supplemental information that required docketing to support the staff's SE. During the audit, USO provided clarifications through its responses to the staff's questions, and following the audit USO provided updates to the PSAR and submitted docketed supplements to the application. The results of the staff's audit of the KU1 CP application are available at (MLXXXXXXXXXX)

During its review of the KU1 CP application, the staff also prepared and issued three requests for confirmation of information (RCIs) (ML25261A106) and the applicant responded to them (ML25259A180).

1.1.5 Ongoing Research and Development

The provisions of 10 CFR 50.34(a)(8) allow for ongoing research and development (R&D) to confirm the adequacy of the design of SSCs to resolve safety questions prior to the completion of construction. In accordance with 10 CFR 50.34(a)(8), USO identified several R&D activities, which are described in PSAR chapter 13. Chapter 13 of this SE evaluates these activities.

1.1.6 Advisory Committee on Reactor Safeguards Review

To support the Advisory Committee on Reactor Safeguards (ACRS) in providing an independent review and report to the Commission regarding the KU1 CP application, the staff presented the results of its SE to the ACRS subcommittee on October 8-9 and 22-23, 2025, and to the full committee on November 5, 2025. After the meetings, to meet the requirements of 10 CFR 50.58, "Hearings and report of the Advisory Committee on Reactor Safeguards," the ACRS issued a letter to the Commission with its recommendations regarding the KU1 CP application. The ACRS letter is provided in appendix C of this SE.

1.1.7 Application Availability

Publicly available documents related to the KU1 CP application may be obtained online in the ADAMS public documents collection at <https://www.nrc.gov/reading-rm/adams.html>. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For assistance with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209 or by email to PDR.Resource@nrc.gov.

The versions of the KU1 PSAR are publicly available in ADAMS. Other public documents and correspondence related to this application may be found by searching KU1's Docket Number, 50-0613, in ADAMS. Portions of the application or correspondence containing sensitive information (e.g., proprietary information) are withheld from public disclosure pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

1.1.8 NRC Staff Contact Information

The project manager for this SE was Mallecia Sutton, Senior Project Manager, Division of Advanced Reactors and Non-power and Utilization Facilities, U.S. Nuclear Regulatory Commission. Ms. Sutton may be contacted regarding this SE at 301-415-0673 or via email at Mallecia.Sutton@nrc.gov. Appendix E to this SE provides a listing of principal contributors, including areas of technical expertise and chapters of authorship.

1.2 Summary and Conclusions on Principal Safety Considerations

The staff evaluated the descriptions and discussions of the proposed KU1 facility, as described in USO's CP application, as supplemented. Based on its review, the staff makes the following findings:

1. Applicable standards and requirements of the Atomic Energy Act and Commission regulations have been met.
2. The acceptance criteria in or referenced in DANU-ISG-2022-01, and other applicable guidance documents have been satisfied for a preliminary design supporting a CP application where the criteria were found to be applicable to the design.

3. Required notifications to other agencies or bodies related to this licensing action have been duly made.
4. Based on the preliminary design of the facility, there is reasonable assurance that the final design will conform to the design basis with adequate margin for safety.
5. There is reasonable assurance that the facility can be constructed in conformity with the permit, the provisions of the Atomic Energy Act, and the Commission's regulations.
6. The staff has evaluated the accident analyses presented by USO in the PSAR and determined that the calculated potential radiation dose consequences outside the KU1 site from postulated accidents are not likely to exceed the dose guidelines of 10 CFR 50.34(a)(1)(ii)(D) using site atmospheric dispersion characteristics as required by 10 CFR Part 100. Furthermore, SSCs have been designed to provide for the prevention of accidents and the mitigation of consequences of accidents.
7. Releases of radioactive materials and wastes from the facility are not expected to result in concentrations outside the limits specified by 10 CFR Part 20, Subpart D, "Radiation Dose Limits for Individual Members of the Public," and are as low as is reasonably achievable (ALARA).
8. The financial information, technical analyses, programs, and organization described in the application, as supplemented, demonstrate that USO is financially and technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
9. The preliminary emergency plan provides reasonable assurance that USO will be prepared to assess and respond to emergency events.
10. The application presents information at a level of detail that is appropriate for general familiarization and understanding of the proposed facility.
11. The application describes the relationship of specific facility design features to reactor operation.
12. Issuance of the CP will not be inimical to the common defense and security or to the health and safety of the public.

Therefore, the staff finds that, subject to certain conditions, the preliminary design and analysis of KU1, as described in the PSAR, is, where relevant, consistent with guidance and is sufficient and meets the applicable regulatory requirements for the issuance of a CP in accordance with 10 CFR 50.35.

Appendix A to this SE identifies certain permit conditions that the staff recommends the Commission include if the CP is issued.

In PSAR section 1.1.5, USO identified several ongoing R&D activities to confirm the adequacy of the design of SSCs to resolve safety questions prior to the completion of construction. The staff is tracking these activities, which are also listed in appendix A to this SE, and will verify that they are resolved prior to the completion of construction.

Based on these findings as documented in this SE, and subject to the permit conditions identified in appendix A of this SE, the staff recommends that the Commission make the following conclusions for the issuance of a CP for the KU1 facility in accordance with 10 CFR 50.35, 50.40, and 50.50:

1. USO has described the proposed design of KU1, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.
2. Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the FSAR.
3. Safety features or components that require R&D have been described by USO and an R&D program will be conducted that is reasonably designed to resolve any safety questions associated with such features or components.
4. On the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.
5. There is reasonable assurance: (i) that the construction of KU1 will not endanger the health and safety of the public, and (ii) that construction activities can be conducted in compliance with the Commission's regulations.
6. USO is technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
7. USO is financially qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
8. The issuance of a permit for the construction of KU1 would not be inimical to the common defense and security or to the health and safety of the public.
9. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of the CP, subject to the conditions for protection of the environment set forth therein, is in accordance with Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)," of 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
10. The application meets the standards and requirements of the Atomic Energy Act and the Commission's regulations, and notifications to other agencies or bodies have been duly made.

1.3 General Description

1.3.1 Introduction

PSAR chapter 1 provides an overview of the facility, including a brief description of the proposed plant location, the type of reactor being proposed, the intended use of the reactor, and a summary description of the overall plant configuration. PSAR section 1.1.4 includes a summary description of plant SSCs, including reactor systems and components, secondary systems and components, support systems and components, and the major structures of the nuclear island (NI).

Safety-significant SSCs (i.e., those that are assigned a safety classification of safety-related (SR) or non-safety-related with special treatment (NSRST)) are described in further detail in PSAR chapter 7 and evaluated in chapter 7 of this SE. The seismic monitoring system is evaluated in section 6.4.3.1 of this SE. Liquid and solid radwaste processing (RWL and RWS) are described in more detail in PSAR chapter 9 and evaluated in chapter 9 of this SE. However, many non-safety related with no special treatment (NST) SSCs are primarily or only described in PSAR section 1.1.4, including:

- secondary systems and components
 - salt systems
 - power cycle systems
 - main power system
- support systems and components
 - NI ancillary electrical system
 - NI plant communication system
 - NI cranes and hoists
 - energy island (EI) auxiliary electrical system
 - NI air and inert gas distribution systems
 - NI major maintenance equipment
 - NI emergency operating lighting
- support buildings
 - NI electrical equipment modules (e-modules)
 - EI facilities

These systems and their classification as NST are evaluated in this section of the SE.

The discussion on secondary systems and components includes a reference to NATD-LIC-RPRT-0001-A, “Regulatory Management of Sodium Nuclear Island and Energy Island Design Interfaces” (ML24011A321). Specifically, as stated in Section 1.1.4.4.9 of the PSAR, “The EI facilities support thermal energy storage and steam generation plant operations that are independent from reactor power operations due to the TSS as described in NATD-LIC-RPRT-0001.” This NRC-approved TR describes an evaluation of NRC regulations pertinent to the design interface of Sodium NI and EI systems. As confirmed through audit, although the applicant mentioned this TR in its application, the applicant is not attempting to rely on any of the analysis within. Further, the staff’s analysis of the application does not rely on the information, analysis, or conclusions in the TR or the NRC staff’s SE approving the TR. The staff additionally notes that the NRC has issued an exemption to USO excluding certain NST EI SSCs from the definition of construction (ML25119A331). In its SE approving this exemption, the NRC staff stated that it had “reviewed the design described in the exemption and the proposed PSAR and found it to be consistent with the design features supporting the NI-EI

independence discussed in NATD-LIC-RPRT-0001-A and the associated NRC staff SE.” The NRC staff has reviewed the current version of the PSAR and confirmed that that statement is still true.

KU1 PSAR section 1.2 provides a site description overview. Section 1.3 of the KU1 PSAR summarizes approaches described in NEI 18-04, Revision 1, as applied to KU1, for the selection of licensing basis events, safety classification of SSCs, and determination of DID adequacy. Section 1.3 of the KU1 PSAR also provides an overview of how the design addresses the fundamental safety functions (FSFs) of retaining radionuclides, controlling heat generation, and controlling heat removal.

KU1 PSAR section 1.4 includes tables that provide:

- a discussion of the facility’s conformance with regulatory guides;
- a listing of the TRs and technical reports that are incorporated by reference into the PSAR;
- a discussion of the facility’s conformance with generic safety issues (GSIs), unresolved safety issues (USIs), and Three Mile Island (TMI) action items;
- the consensus codes and standards used in the design.

1.3.2 Regulatory Evaluation

Regulations applicable to the information included in PSAR chapter 1 include 10 CFR 50.34. In particular, 10 CFR 50.34(a)(1)(ii) requires a description and safety assessment of the site and a safety assessment of the facility, and 10 CFR 50.34(a)(2) requires a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

Section 1 of DANU-ISG-2022-01 provides guidance for, in part, the review of chapter 1 of an LMP-based PSAR. This section references RG 1.253, Revision 0, which endorses, with clarifications and additions, NEI 21-07, Revision 1.

NEI 21-07 indicates that PSAR section 1.1 content, which provides the summary description of the plant and its SSCs, should reflect the preliminary nature of the design, as appropriate, and should be sufficient to “permit the reader to understand fundamental concepts of the plant and how it operates,” “support reader understanding of the design and how the LMP-based affirmative safety case will be developed,” and “understand the initial plant functionality.” The guidance for PSAR section 1.2 states that the section should provide a high-level overview of the site and general vicinity of the licensed activities. The guidance for PSAR section 1.3 states that the section should provide “a high-level overview of the LMP-based affirmative safety case methodology,” focused on the FSFs and the DID aspects of the design.

DANU-ISG-2022-01 section 1.1.5 provides additional guidance including that the following summary tables should be included in chapter 1 of a PSAR:

- The GSIs, USIs, and TMI action items technically relevant to the design. Section 1.1.5 of DANU-ISG-2022-01 notes that appendix B of the ISG provides useful information on the applicability of NRC regulations to non-light-water power reactors that should be considered when reviewing such tables.

- DANU-ISG-2022-01 appendix B notes that while 10 CFR 50.34(f), “TMI-related requirements,” does not apply to the design, the Commission has given the staff direction in SECY-15-0002 (ML15266A023) confirming that early directions for 10 CFR Part 52 new power reactor applications be applied consistent to 10 CFR Part 50 new power reactor applications. Based on this, DANU-ISG-2022-01 appendix B states that the staff should ensure that an applicant addresses the technically relevant TMI-related items during the review process and propose license conditions requiring the appropriate item in the interim.
- RGs directly applicable to the design, and whether the applicant proposes an alternative approach to satisfy a regulation rather than following the guidance in one of the RGs.
- The consensus codes and standards used in the design, and whether the applicant proposes to request an exemption from or alternative to such standards that are incorporated by reference into 10 CFR 50.55a.

1.3.3 Technical Evaluation

1.3.3.1 Description of Plant SSCs

In general, the summary design information provided in PSAR section 1.1.4 addresses the summary description required by 10 CFR 50.34(a)(2), while for those NST SSCs only described in PSAR section 1.1.4 this information is the only information provided relative to the safety assessment of the facility required by 10 CFR 50.34(a)(1)(ii). As such, the staff’s focus for these SSCs was to ensure that the information provided was consistent with the guidance in NEI 21-07 and RG 1.253 such that the staff could understand the functionality of each SSC and the role it plays in the safety analysis, to support a conclusion that the information adequately supports 10 CFR 50.34(a)(1)(ii) and (a)(2).

1.3.3.1.1 Secondary Systems and Components

Section 1.1.4.2 of the KU1 PSAR describes the secondary systems and components. The KU1 design includes a separate NI and EI. Secondary systems and components are contained on the EI, while the NI contains the reactor and associated sodium and fuel handling systems. The interface between the NI and EI is shown in PSAR figure 1.1-1. PSAR section 1.1.4.2 states that the NI boundary conditions have been designed so the interface with the EI does not impact the KU1 safety analysis, and all EI systems and functions are classified as NST. The summary description provided in PSAR section 1.1 is discussed below; more detail on the interfaces between the EI and NI (particularly through the molten salt and electrical systems) are discussed in SE chapter 7.

Salt and Power Cycle Systems

Section 1.1.4.2.1 of the PSAR describes the salt systems, which consist of the nuclear island salt system (NSS), energy island salt system (ESS), and the thermal salt storage system (TSS). The section also describes the role of the salt systems in transferring energy from the intermediate heat transfer system (IHT) to the TSS salt storage tanks and beyond to the steam generation system (SGS); how the NSS isolation valves provide the boundary between the NSS and ESS (and thus the NI and EI); and how the salt system is monitored from the main control room during normal operations using the NSS molten salt flow rate and hot and cold salt

temperatures and pressures. The sodium-salt heat exchanger (SHX) provides the interface between the salt-based NSS and the sodium-based IHT; IHT and SHX are described in more detail in section 7.1.4 of this SE.

Section 1.1.4.2.2 of the PSAR describes the power cycle systems, which consist of the SGS, condensate and feedwater system (CFW), steam turbine system (STS), generator system (GEN), and heat rejection system (HRS). These systems are used to generate electricity and reject waste heat to the atmosphere via cooling towers.

As described in chapter 3 of the PSAR, the salt systems and the downstream power cycle systems are not relied on for decay heat removal following a plant transient or accident. Transients initiated by a failure in these systems are evaluated in PSAR chapter 3 as “energy island transients” and result in either a reactor runback (an automatic, controlled reduction in power and flow) or scram. Design basis events (DBEs) involving EI systems are described in PSAR sections 3.7.2.3 and 3.7.2.4, a beyond design basis event (BDBE) is described in PSAR section 3.8.2.3, and a design basis accident (DBA) is described in PSAR section 3.9.2.2. For all transients, the reactor remains in a safe condition with no fuel failure and with heat removal provided by NI SSCs. As discussed in PSAR section 1.1.4.2 and confirmed by the staff’s review of chapter 5, the preliminary classification of all EI systems is NST. Based on the description provided in PSAR chapter 1 and the disposition of EI transients in PSAR chapter 3, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the salt and power cycle systems and the role of these systems in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR content regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

Main Power System

Section 1.1.4.2.3 of the PSAR describes the main power system. This system includes a normal main power subsystem that supplies power from the generator to the 230 kilovolt switchyard and plant auxiliary loads through the unit auxiliary transformer, and an alternate main power supply subsystem that transfers power from the switchyard to plant auxiliary loads through the reserve auxiliary transformer when the main power subsystem is unavailable. Power is supplied to plant loads through these transformers from the switchyard during startup and shutdown. The unit auxiliary transformer and reserve auxiliary transformer supply the NI and EI auxiliary electrical systems, which are described in more detail in PSAR sections 1.1.4.3.6 and 1.1.4.3.14, respectively, and will be discussed in the following section of this SE. Safety-significant power systems are discussed in more detail in PSAR section 7.7.

While electric power is needed for some NSRST functions, as described in PSAR section 7.7, these loads are supplied via the NI uninterruptible alternating current (AC) power supply system (NUP) and NI direct current (DC) power supply system (NDC) for up to 72 hours using batteries. These systems are evaluated in section 7.7 of this SE. The 72 hour mission time is consistent with the SBO coping period specified in SRP Section 8.4, “Station Blackout” (ML100740424) and SRP 194, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors” (ML13081A756). Based on the adequacy of the NSRST systems to support this mission time, the staff determined that it is reasonable for the main power system to be classified as NST. Considering the summary-level information provided in PSAR chapter 1 and the more detailed information provided in PSAR chapter 7, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and

operation of the main power system and the role of these systems in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR content regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

1.3.3.1.2 Support Systems and Components

NI Ancillary Electrical System, Plant Communication System, and Emergency Operating Lighting System

Section 1.1.4.3.11 of the PSAR describes the NI ancillary electrical system, which includes NI lighting systems (NLS) and NI grounding and earthing and lighting protection (NGL). Power for NLS is provided by the NI AC electrical power low voltage system and backed up for certain key lighting systems by self-contained batteries, diesel generators, or NUP. NGL protects facility staff and equipment from transient over-voltages, gives a ground reference for instrumentation signals, and provides protection for lightning strikes and switching surges.

Section 1.1.4.3.12 of the PSAR describes the NI plant communication system, which is used to provide internal and external communications during normal and emergency plant operations. The PSAR states that it consists of diverse and independent communication subsystems that will be designed, fabricated, erected, constructed, and tested in accordance with industry standards. Emergency communication capabilities are described in additional detail in PSAR section 11.3, which notes that additional information will be provided in the OL application.

Section 1.1.4.3.17 of the PSAR describes the NI emergency operating lighting system. Emergency lights are used to support fire suppression and recovery actions in the NI control building (NCB), in and between the main control room (MCR), and remote shutdown complex (RSC). The lights are DC self-contained battery-operated units, and when in proximity to safety-significant components are appropriately designed to mitigate impacts due to seismic events; this is consistent with the seismic interaction design requirements described in PSAR section 6.4.1.5 and evaluated by the staff in section 6.4.3.1.5 of this SE.

The staff did not identify any SR functions that rely on the NI ancillary electrical, NI plant communication, or NI emergency operating lighting systems, though the staff noted that lighting may be needed for manual actions that are NSRST for DID adequacy. The applicant stated that lighting is evaluated as part of the human factors engineering (HFE) program, as discussed in PSAR section 11.2. The staff evaluated the HFE program in section 11.2 of this SE, but noted that a complete evaluation of the adequacy of HFE within the KU1 design must be deferred until the OL application. The NSRST manual actions include manual scram, manual primary sodium pump (PSP) trip, manual intermediate sodium pump (ISP) trip, and manual sodium processing system (SPS) pump trip, as described in PSAR chapter 5. These functions are performed in the MCR and RSC, and communications systems are not needed to perform these functions. The MCR and RSC are discussed in PSAR section 7.6.7. Based on the summary-level information provided in PSAR chapter 1, as augmented by information on safety classifications, manual actions, and emergency communications needs as discussed in PSAR chapters 5, 7, and 11, the staff determined that the preliminary information was sufficient to understand the fundamental design and operation of the lighting and communications systems and the role of these systems in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR content regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

NI Cranes and Hoists

Section 1.1.4.3.13 of the PSAR describes the NI cranes and hoists (NCH) system, which consists of the bridge cranes and monorail hoists in the reactor building (RXB), fuel handling building (FHB), and reactor auxiliary building (RAB). The RXB and FHB cranes lift and move critical heavy loads. These cranes are designed to be single-failure proof and to the standard of ASME NOG-1 as Type I cranes. Other monorail hoists in FHB and RAB do not carry critical loads and are not single failure proof. Cranes used to handle fuel are not part of NCH system and are described in additional detail in PSAR section 7.3. Loads from NCH are accounted for in the building structural designs as described in PSAR sections 6.4 and 7.8. Based on the summary-level information provided in PSAR chapter 1, as augmented by more detailed information on how loads are accounted for in structural design in PSAR chapters 6 and 7, the staff determined that the preliminary information was sufficient to understand the fundamental design and operation of the NCH systems. Because of this, this aspect of the content is consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01 and as such the staff determined it is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2). The role of NCH cranes in preventing the occurrence of postulated initiating events (PIEs) is not evaluated in the PSAR as part of the LMP-based safety analysis. The applicant indicated in a supplement to the CP application (ML25253A386) that it will be assessed using a new process for evaluating preventative measures, and would be fully addressed in the KU1 OL application, should the applicant apply for an OL. The staff evaluation of this process is in SE section 5.1. Given the RXB and FHB bridge cranes are being designed to the ASME NOG-1 standard as a single-failure proof (Type I) crane, which will make them highly reliable, the staff determined the design information was adequate for a CP application and thus acceptable. The staff expects to review the design and safety classification of NCH in more detail during the OL application review, should the applicant apply for an OL.

EI Auxiliary Electrical System

PSAR section 1.1.4.3.14 describes the EI auxiliary electrical system, which consists of medium voltage and low voltage systems serving the EI. The EI auxiliary electrical system also includes the standby diesel generators (SDGs), which are available to provide backup power to the NI AC electrical power medium voltage system for asset protection purposes.

Like other EI systems, the EI auxiliary electrical system is NST. Despite this, there are some analyzed LBEs in PSAR chapter 3 where power from the SDGs is modeled (e.g., the loss of offsite power with non-passive intermediate air cooling (IAC) described in PSAR section 3.6.1.2). However, for each of these LBEs there are also similar LBEs without power from the SDGs that demonstrate the fuel acceptance criteria are satisfied (e.g., the loss of offsite power with passive IAC described in PSAR section 3.7.1.2); this shows that the SDGs are not needed to maintain events within the frequency-consequence target curve and confirms that they are acceptable to classify as NST.

Based on the above, and the discussion provided in chapters 1, 3, and 7 of the PSAR relative to the EI auxiliary electrical system, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the EI auxiliary electrical system and the role of these SSCs (in particular, the SDGs) in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the

staff determined that the PSAR content regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

NI Major Maintenance Equipment

PSAR section 1.1.4.3.19 describes the NI major maintenance equipment (NME), which consists of a cask, a transfer adapter with closure valve, an inert gas management system, and a component lift system with a grapple. The NME provides a temporary inert gas environment to prevent excess air ingress into the primary cover gas and leakage of primary cover gas to the RXB during installation and removal of in-vessel fuel handling equipment. The installation and removal of the in-vessel fuel handling equipment occurs during operating mode 4, which specifies that the reactor is shutdown with no more than one control rod assembly coupled to its drive (i.e., the reactor is subcritical) and a sodium temperature of less than 400° F. The in-vessel equipment used to handle fuel is not part of NME and is described in additional detail in PSAR section 7.3.3. PSAR Section 1.1.4.3.19 also states that details regarding refueling operations, including PRA evaluation of the installation and removal of the in-vessel fuel handling equipment, would be developed in support of the OL review. Based on the information provided in PSAR chapter 1, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the NI major maintenance systems, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such the staff determined it is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2). With respect to the LMP-based safety-analysis, the NME performs no identified role in preventing or mitigating any LBE. However, it maintains an inert gas environment connected to the reactor head component nozzles while they are open for installation of the in-vessel equipment. Design features of the in-vessel fuel handling equipment, as described in PSAR section 7.3.3, provide assurance that an NME handling event that causes the in-vessel equipment to drop during installation does not result in damage to fuel. The staff expects to review the design and safety classification of the NME, as well as NME operations during refueling, in more detail during the OL application review, should the applicant apply for an OL.

1.3.3.1.3 Support Buildings

Support buildings described in PSAR chapter 1 consist of the nuclear island e-modules and EI facilities. The e-modules contain electrical systems and components. As described in PSAR section 7.7, safety-significant NI electrical components are housed in the NI control building substructure and are thus not contained within e-modules. EI facilities include all the buildings on the EI, which as previously discussed, are classified as NST along with the rest of the EI SSCs. Based on the information provided in PSAR chapter 1, as augmented by the discussion in chapter 7 clarifying the role of the e-modules, the staff understand the fundamental design and operation of these SSCs and their role in the LMP-based safety analysis. As such, the staff determined that the content regarding these systems is consistent with the applicable guidance and therefore acceptable.

1.3.3.2 Site Description

The staff reviewed site description information in section 1.2 of the KU1 PSAR and site information provided in PSAR chapter 2. The staff review is documented in SE chapter 2.

1.3.3.3 Plant Safety Overview

The staff reviewed information in section 1.3 of the KU1 PSAR, including how fundamental safety functions are addressed. The staff review is documented in SE chapters 3, 4, and 5.

1.3.3.3.1 Licensing Basis Methodology

As described in PSAR section 1.3.1, the KU1 licensing basis was developed using the process described in NEI 18-04. This process is referred to interchangeably throughout this SE as the NEI 18-04 process, the LMP process, the NEI 18-04 methodology, or the LMP methodology. As described earlier, NEI 18-04 was endorsed, with certain clarifications, in RG 1.233. Guidance for the content of applications using this process is provided in NEI 21-07, which was endorsed, with certain clarifications and additions, in RG 1.253, and is also provided in DANU-ISG-2022-01 as well as the references therein.

PSAR section 1.3.1 states that conformance with the NEI 18-04 process is demonstrated in PSAR chapters 3 through 8. The staff's detailed evaluation of the applicant's implementation of the NEI 18-04 process is documented throughout the corresponding chapters of this SE. Because this application is the first time LMP is used for a commercial power reactor, an orientation to the NEI 18-04 process and how it is reflected in the structure of the PSAR is provided below.

From a safety analysis perspective, the LMP process has the overall process flow shown below. First, an applicant develops an initial list of PIEs and a PRA. Development of the PRA necessitates supporting information, including elements such as initiating event identification, event sequence analysis, and human reliability analysis. The PRA is used to refine and further develop the initial set of LBEs, which are then categorized based on their event sequence frequency. The LBEs are then analyzed for their radiological consequences, which involves analyzing system transients (with nuclear and thermal hydraulic analyses), source term generation, and atmospheric transport (i.e., consequence analysis). LBE frequencies and consequences are evaluated against a frequency-consequence (F-C) target curve, which is also used to determine which events are risk significant based on their location relative to the target.

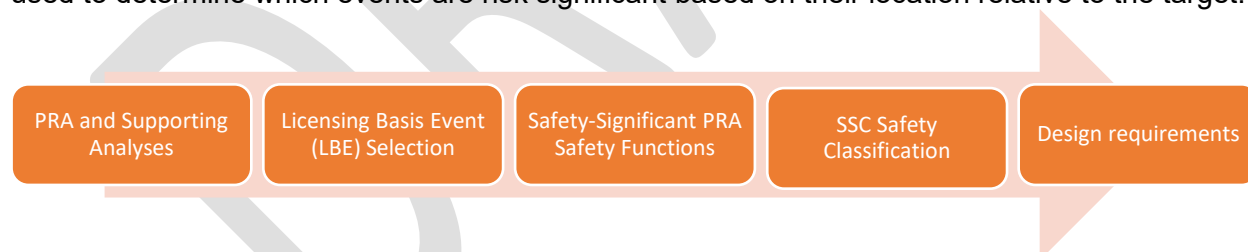


Figure 1.3-1: General flow of NEI 18-04 process

Once events have been selected, an applicant would evaluate the safety functions modeled in the PRA (also referred to as PRA safety functions, or PSFs) involved in those events to determine if they are safety-significant by assessing the role they play in maintaining LBEs below the F-C target curve, preventing and mitigating DBAs, and their contributions to integrated risk metrics and meeting DID adequacy. DID adequacy is evaluated in an integrated process that ensures multiple functions are available to perform each of the generically applicable FSFs described in NEI 18-04. PSFs that are determined to be safety-significant because they are needed to maintain LBEs below the F-C target curve or because of their contribution to integrated risk metrics are also designated risk-significant. The SSCs are then assigned a safety classification and risk significance based on the function(s) they perform. Design requirements for each SSC are then determined based, in part, on the safety

classification and what is needed to ensure the SSC can appropriately perform its safety function(s). The same process is used to identify programmatic special treatments applicable to each SSC.

This process flow is reflected in the structure of the PSAR for the KU1 CP application. In the PSAR, the PRA is described in section 3.1, source terms in section 3.2, and the methodologies for consequence and other supporting analyses in section 3.3. The overall selected set of selected LBEs and descriptions of each LBE, including frequencies and radiological consequences, consistent with the NEI 21-07 guidance, are provided in sections 3.4 through 3.9 of the PSAR. Integrated risk metrics and DID evaluation are discussed in chapter 4 of the PSAR. The methodology for identifying safety-significant PSFs is in section 5.1 of the PSAR, with results in section 5.2 of the PSAR. In the SE, the staff discusses the functions and their safety classifications in sections 5.2, 5.4, and 5.5 to align more with the NEI 21-07 guidance. The design requirements and methodology that are applied based on those safety classifications and functions are provided in chapter 6, and descriptions of the design and how it meets those requirements to perform the safety-significant PSF are in chapter 7. Finally, the programs discussed in chapter 8 serve as programmatic special treatments for the SSCs described in chapter 7.

The staff identified that the content of the PSAR differs slightly from NEI 21-07. PSAR chapter 2 includes site information, as discussed in DANU-ISG-2022-01. The supporting analyses discussed in NEI 21-07 chapter 2, like fuel design, nuclear and thermal-hydraulic design, and criticality safety analyses, are described in PSAR chapter 3. Where NEI 21-07 chapter 6 includes the design requirements and descriptions of SR SSCs and chapter 7 includes design requirements and descriptions of NSRST SSCs, PSAR chapter 6 provides the design requirements for SR and NSRST SSCs and PSAR chapter 7 provides a description of all SR and NSRST SSCs. The staff determined that this reorganization of the information in the PSAR relative to the guidance does not affect the intended scope of the information.

1.3.3.3.2 Fundamental Safety Functions

As described in PSAR section 1.3.2, the FSFs for KU1 include control of heat generation, control of heat removal, and radionuclide retention. The staff determined these FSFs are consistent with those described in NEI 18-04 which are intended to be generically applicable to all reactors.

PSAR section 1.3.2.2 explains how control of heat generation is established using two different banks of control rods, which can be inserted using both active (power runback, which inserts control rods using the control rod drive system) and passive (gravity scram) means. DID for the gravity scram is provided by a control rod drive (CRD) driveline scram follow feature, in which the protection system commands the CRD to insert control rods into the core coincident with a scram signal. Though not described in PSAR chapter 1, the alternative shunt trip discussed briefly in PSAR chapters 3 and 5 provides a DID pathway for tripping the reactor trip breakers and initiating a gravity scram. In addition to the active and passive means of controlling reactivity, there is inherent reactivity feedback provided by the reactor core. Design features related to control of heat generation are described in additional detail in chapter 7 of the PSAR, with additional discussion on their performance in transient analyses in chapter 3.

PSAR section 1.3.2.3 explains how the IAC and reactor air cooling system (RAC) satisfy the FSF of controlling heat removal. The PSAR states that IAC provides active and passive means

of controlling heat removal by pulling heat from the reactor core using the intermediate heat exchangers into the IHT and discharging them to the atmosphere through sodium-air heat exchangers. The active means of IAC relies on the intermediate sodium pumps and IAC blower, while the passive means relies on natural circulation. The RAC provides heat removal by directing air from outside the reactor building past the reactor and out through an outlet stack. Motive force for the air is provided by natural circulation. The PSAR states that the RAC is an inherent means of controlling heat removal; however, the staff considers it to be passive because it relies on maintaining an open flow path. Both passive IAC and RAC cooling rely on natural circulation of sodium in the reactor vessel to draw heat from the core. IAC and RAC and supporting features in the IHT and primary heat transport system are described in detail in chapter 7 of the PSAR, with additional discussion on their performance in transient analyses in chapter 3.

PSAR section 1.3.2.1 describes the applicant's approach to ensuring radionuclide retention, which is achieved using a functional containment strategy. A high-level definition of functional containment is provided in SECY-18-0096, which was approved by the Commission, as "a barrier, or set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment." Under this definition, compared to the containments of the operating light-water reactor fleet, a functional containment is not necessarily comprised of a single pressure-retaining containment structure. Instead, in the approach described in SECY-18-0096, functional containment performance criteria are developed to address each barrier's role in mitigating releases to meet plant-level performance criteria (i.e., regulatory dose requirements). Enclosure 2 to SECY-18-0096 describes a proposed risk-informed, performance-based, technology-inclusive approach to derive these functional containment performance criteria. The staff notes that while functional containment is commonly associated with the development of the Next Generation Nuclear Plant (NGNP), a gas-cooled reactor, and the use of tristructural isotropic (TRISO) fuel, the methodology described in SECY-18-0096 and Enclosure 2 that was approved by the Commission is applicable to any non-LWR technology.

The staff identified that the NEI 18-04 methodology is consistent with the approach described in the enclosure to SECY-18-0096 and provides the necessary information to develop the functional containment performance criteria. Under the NEI 18-04 process, mechanistic source term and radiological consequence analyses are used to determine offsite doses for the LBEs and thus evaluate the adequacy of functional containment barriers. Functional containment performance criteria are then used to inform reliability and capability targets, which are intended to ensure successful completion of PRA safety functions, including radionuclide retention. This overall process is reflected in the PSAR and the staff's SE. The LBEs that result in releases for which functional containment performance must be analyzed are described in PSAR sections 3.5 through 3.9, and evaluated by the NRC staff in sections 3.4 through 3.8 of this SE. PSAR section 3.2 describes the mechanistic source term methodology and the source terms analyzed for the KU1 CP application. The staff evaluates source term methodology and analyses in section 3.2 of this SE. PSAR sections 3.3.1.4 and 3.3.2.2 describe the methodology to calculate radiological consequences for the non-DBA LBEs and DBAs, respectively. The staff evaluates radiological release consequences methodology and analyses in section 3.3.1.5 of this SE. In the CP application analyses, functional containment barrier performance is based on assumptions, which feed into the capability targets; the design will be fully evaluated by the staff at the OL. PSAR section 6.2 describes the applicant's assessment of reliability and capability targets for safety-significant SSCs, which is evaluated by the staff in section 6.2 of this SE.

Section 1.3.2.1 provides a list of SSCs associated with the functional containment strategy. The staff reviewed this list and determined it was consistent with the identification of radionuclide

retention functions for SSCs as reflected through the safety analysis, safety classification, and plant design information provided in chapters 3, 5, 6, and 7 of the PSAR.

With respect to functional containment, the staff concluded that the overall approach is acceptable to apply to the KU1 design based on the use of the NEI 18-04 methodology for licensing basis development, which is technology-inclusive and consistent with the process described in SECY-18-0096 Enclosure 2, and the reasonable identification of those barriers responsible for radionuclide retention. However, the staff did not come to a final determination on the adequacy and acceptability of functional containment performance due to the preliminary nature of the design and analysis as discussed in chapters 3, 4, 5, 6, and 7 of this SE. The staff will confirm the acceptability of the functional containment performance criteria and the associated design and performance evaluation at the OL, should the applicant apply for an OL.

1.3.3.4 Conformance with Regulatory Criteria and Referenced Material

Table 1.4-1 of the KU1 PSAR provides a list of applicable RGs. The table includes a discussion of whether the design is in full conformance or partial conformance with the RG. As appropriate, the RG conformance is evaluated in the applicable sections of this SE. The staff noted that table 1.4-1 also provides a listing of RGs are not considered by USO to be applicable to at the CP phase and will be addressed as part of the OL application.

Applicable TRs and technical reports are identified in table 1.4-2 and table 1.4-3 of the KU1 PSAR, respectively. Table 1.11-1 of this SE provides a listing of the TRs that have been incorporated by reference into the PSAR and where in this SE the staff evaluated the applicability of the TRs and any associated L&Cs associated with these reports. Table 1.11-2 of this SE provides a listing of the technical reports that are incorporated by reference into the PSAR and submitted as part of the CP application and where in this SE report these technical reports are evaluated.

PSAR table 1.4-4 provides a listing of TMI-related items that USO identified as either fully or partially applicable to the KU1 design. DANU-ISG-2022-01, appendix B, table 4 provides generic applicability determinations of the TMI items for non-LWRs, with entry conditions for technical relevancy listed for some items. The discussion associated with table 4 states the TMI items are only requirements applicable to 10 CFR Part 52 applications, but as discussed in section 1.3.1 of this SE, DANU-ISG-2022-01, appendix B also states that the staff should ensure that an applicant addresses the technically relevant TMI-related items during the review process and propose license conditions requiring the appropriate items in the interim. The staff compared the TMI related items in table 1.4-4 of the KU1 PSAR against the information found in DANU-ISG-2022-01 table 4. Table 1.11-3 of this SE provides a listing of the TMI items that are either fully or partially applicable, where in this SE these requirements are evaluated, and the staff's disposition of the applicability of the items.

Regarding GSIs and USIs, the applicant determined in PSAR section 1.4.3 that none of the non-TMI items provided in appendix B of NUREG-0933, Resolution of Generic Safety Issues (formerly titled "A Prioritization of Generic Safety Issues") applied to the KU1 facility. The staff notes that there is no requirement in 10 CFR Part 50, similar to the 10 CFR Part 52 requirements, to address USIs and GSIs in NUREG-0933. Although not a requirement, guidance in section 1.1.5 of DANU-ISG-2022-01 suggests that the technically relevant USIs and medium and high priority GSIs be evaluated for non-LWR applicants. The staff performed an independent review of the USIs and GSIs to determine if any of these items are technically

relevant to the KU1 facility. The staff identified several items with technical relevance to the KU1 design but concluded that they are either appropriately addressed through the design process (including by conformance to applicable guidance or consensus codes and standards) as described in section X of this SE or can be assessed at the OL stage of the review. Examples include Task Action Plan Item A-25, which concerns non-safety related loads on Class 1E power sources and is addressed through conformance with RG 1.75, and GSI-186, which concerns heavy load drops and is addressed through design of cranes to the ASME NOG-1 standard and implementation of a heavy load program in conformance with RG 1.244 (which will be implemented at the OL stage). As such, the staff determined that the USIs and GSIs are appropriately addressed for the CP application.

Table 1.4-5 of the KU1 PSAR provides a listing of the codes and standards used in the design. Providing a listing of the codes and standards used in the design is consistent with the guidance found in DANU-ISG-2022-01 section 1.1.5. As applicable, these codes and standards are evaluated as part of the SSC evaluations found in this SE.

Evaluation of Topical Reports

TRs incorporated by reference are listed in table 1.4-2 of the KU1 PSAR. The applicability of the TRs and any associated L&Cs are evaluated in the primary location where they are referenced in the PSAR. Table 1.11-1 of this SE details where each TR is evaluated by the staff.

1.3.4 Conclusion

The staff reviewed the information on the plant SSCs provided in PSAR section 1.1.4, the plant safety overview information provided in PSAR section 1.3, and the conformance with regulatory criteria and reference material provided in PSAR section 1.4, as discussed above, and determined it is acceptable because it is consistent with the applicable guidance on content of applications from RG 1.253, NEI 21-07, and DANU-ISG-2022-01. Several aspects of the information provided in chapter 1, including the site description provided in PSAR sections 1.2 are evaluated in more detail in other portions of the staff's SE as described above.

1.4 Shared Facilities and Equipment

The staff determined that this section is not applicable because KU1 is a single-unit site, as stated explicitly in PSAR section 5.3.1.5.

1.5 Comparison with Similar Facilities

The applicant did not provide a comparison with similar facilities, because it is not requested by RG 1.253 or NEI 21-07. However, the staff includes a brief discussion of similar facilities and relevant licensing history here. KU1 is a pool-type sodium-cooled fast reactor (SFR) using metallic uranium-zirconium alloy fuel. This configuration has similarities to several different reactors that have operated in the US and internationally. Information on prior experience in operating and licensing similar facilities is available in NUREG/KM-0007, "NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors" (ML14128A346). While this discussion focuses on domestic operating and design experience since many international reactors differ substantially in the fuel or primary coolant system design from the KU1 design, further information on international reactor designs is provided in NUREG/KM-0007. The

international operating experience, particularly regarding sodium fires and design of intermediate and secondary systems, was considered by the staff during the review.

The US operating experience with liquid-metal cooled reactors began with the construction of the Experimental Breeder Reactor I (EBR-I) at Argonne West (now Idaho National Laboratory) in Idaho, a test reactor which operated from 1951–1963. Subsequent test reactors include the Sodium Reactor Experiment (SRE), which operated from 1957–1964 at the Santa Susana Field Laboratory in California; the Experimental Breeder Reactor II (EBR-II), which operated from 1964–1994 at Argonne West (now Idaho National Laboratory, or INL) in Idaho; the Southwest Experimental Fast Oxide Reactor (SEFOR), which operated from 1969–1972 in northwest Arkansas; and the Fast Flux Test Facility (FFTF), which operated from 1980–1993 at Pacific Northwest Laboratory (PNL, now Pacific Northwest National Laboratory, or PNNL) in Washington. The FFTF design was reviewed by the NRC and an SE on the final safety analysis report (FSAR) was issued in 1978 as NUREG-0358. One commercial reactor, Enrico Fermi Nuclear Generating Station, Unit 1 (Fermi 1), operated from 1963–1966 in Michigan.

Other SFRs were proposed in the 1970s through the present, and several of them went through various stages of the licensing process with the NRC; most notably, the Clinch River Breeder Reactor Project (CRBRP) and General Electric's PRISM reactor. CRBRP, a loop-type, oxide-fueled SFR similar to FFTF, applied for a CP and submitted a PSAR to the NRC in 1975. The NRC issued an SE report for CRBRP in 1983 as NUREG-0968 (ML082381008), but due to project cancellation a CP was never issued. Regulatory engagements regarding PRISM, a pool-type, metal-fueled SFR, began in 1986 with the submittal of a preliminary safety information document. The NRC issued a pre-application safety analysis report for PRISM in 1994 as NUREG-1368 (ML063410561).

Of these reactors, all used sodium coolant except EBR-I, which used a sodium-potassium eutectic (NaK). EBR-I, SRE, EBR-II, and Fermi 1 all primarily used metallic fuel like KU1, though EBR-II was also used to test a wide variety of different fuels. SEFOR and FFTF primarily used oxide fuel, though FFTF also later tested a number of metal fuel pins. Of the reactors that operated in the US, the KU1 overall system design is the most similar to EBR-II because it was a pool-type reactor, though there are substantial differences in size and the detailed design between the two plants. The size and core configuration of FFTF was also somewhat similar to KU1, in that it had similarly sized fuel assemblies and used a limited free bow core restraint system like KU1, but it had coolant loops that circulated primary coolant outside of the reactor vessel. As such, though most key features of KU1 have been demonstrated through operating experience, there is no single facility that fully captures all of them.

The staff's review was, in particular, informed by SFR operating experience where fuel was damaged. At EBR-I, fuel damage occurred during a coolant flow test due to thermal expansion of the core components that had not been accounted for by design. At SRE, an organic lubricant infiltrated the primary system and reacted with the sodium, forming blockages that led to fuel damage. At Fermi 1, fuel damage was caused by a flow blockage when a metal plate detached from the bottom of the reactor and covered a fuel assembly inlet. This operating experience has informed the subsequent licensing history of SFRs, which has typically required evaluation of flow blockages.

Of the conceptual designs that were not operated, the PRISM design is most similar to KU1. Both are pool-type, metal-fueled SFRs, with seismic isolation systems and safety-related decay heat removal provided by an air-based cooling system (called reactor air cooling system (RAC)

for KU1 and reactor vessel auxiliary cooling system (RVACS) for PRISM). The designs are similar enough that the staff understands, from audit discussions, that the PRISM probabilistic risk assessment (PRA) model was used as a basis for the KU1 PRA. However, there are still substantial differences between the facilities. KU1 has a molten salt energy storage system separating the sodium from the water-based power conversion system, where PRISM ran steam generators directly off the intermediate loop. KU1 has mechanical primary coolant pumps, while PRISM had electromagnetic pumps. Where KU1 uses a functional containment concept with a variety of barriers, including the NSRST head access area as a functional containment boundary, PRISM had a leak-tight containment dome that covered the reactor head. KU1 also has a separate decay heat removal pathway through the intermediate air cooling (IAC) system, a feature not present in PRISM.

1.6 Summary of Operations

Plant operations are addressed to the extent necessary for a CP application in chapters 8, 9, 10, 11, and 12 of this SE.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

The Nuclear Waste Policy Act of 1982 (42 USC § 10101) provides that the U.S. Government is responsible for the permanent disposal of high-level radioactive waste and spent nuclear fuel, but the cost of disposal should be the responsibility of the generators and owners of such waste and spent fuel. Guidance for the staff to evaluate compliance with the Nuclear Waste Policy Act is provided in DANU-ISG-2022-01.

The applicant's letter submitting the CP application stated that "TerraPower is in good faith negotiations with the Department of Energy to enter into a contract for the disposal of high-level waste and nuclear fuel under section 302(b) of the Nuclear Waste Policy Act of 1982, as amended." The staff notes that while USO is a wholly owned subsidiary of TerraPower, the expectation is that the permit holder, USO, will have a contract for waste disposal with the DOE as described above. By letter dated October 3, 2025 (ML25276A027), USO provided documentation from DOE that USO is actively and in good faith negotiating on a contract under section 302(b) of the Nuclear Waste Policy Act. Because USO has provided documentation of good faith negotiations with the Department of Energy, the staff finds that USO is in compliance with the Nuclear Waste Policy Act at the CP stage, consistent with DANU-ISG-2022-01.

1.8 Tables

Table 1.11-1 Topical Reports Incorporated by Reference into the Application

Topical Report Number	Topical Report Title	ADAMS Accession Number	Safety Evaluation Report Section
TP-QA-PD-0001	Quality Assurance Program Description Topical Report	ML23116A179	8
NAT-3056	Plume Exposure Pathway Emergency Planning Zone Sizing Methodology		11.3
NAT-2806	Fuel and Control Assembly Qualification		3.10
NATD-LIC-RPRT-0002	Principal Design Criteria for the Natrium Advanced Reactor		5.3
NAT-3226	An Analysis of Potential Volcanic Hazard at the Proposed Natrium Site Near Kemmerer, Wyoming		2.7
NAT-2965	Human Factors Engineering Program Plan and Methodologies		11.2
TP-LIC-RPT-0003	Radiological Source Term Methodology		3.2
TP-LIC-RPT-0004	Design Basis Accident Methodology for in-vessel events without Radiological Release		3.3
TP-LIC-RPT-0005	Radiological Release Consequences Methodology		3.3
TP-LIC-RPT-0006	Stability Methodology		3.11
TP-LIC-RPT-0008	Partial flow Blockage		3.3
NAT-4950	Instrumentation and Control Architecture and Design Basis		7.6.3, 7.6.5, and 7.6.7
NAT-8922	Reactor Seismic Isolation System Qualification		7.1.2
TP-LIC-RPT-0007	DBA Transient Methods for Events with Radiological Release		3.3

Table 1.11-2 Technical Reports Incorporated by Reference in the Application

Technical Report Number	Technical Report Title	Safety Evaluation Report Section
TP-LIC-RPT-0011	Core Design and Thermal Hydraulic Technical Report	3.11, 3.12
TP-LIC-RPT-0012	Preliminary Emergency Planning Zone Determination Analysis	11.3

Table 1.11-3 Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
10 CFR 50.34(f)(1)(i)	PRA to seek improvements in reliability of heat removal systems	3.1.1
10 CFR 50.34(f)(1)(iii)	Reactor coolant pump seal damage	Listed in DANU-ISG-2022-01, appendix B table 4 as applicable only for reactor designs that have a coolant pump with seals that retain inventory credited for core cooling. This is not applicable to the KU1 design because it does not have coolant pumps with seals that retain inventory credited for core cooling.
10 CFR 50.34(f)(1)(xii)	Perform an evaluation of alternative hydrogen control systems	Not listed in DANU-ISG-2022-01, appendix B table 4 as being generically applicable but it is listed in KU1 table 1.4-4 PSAR as being applicable. See section 7.2.4 of this SE.
10 CFR 50.34(f)(2)(i)	Control room simulator	11.2
10 CFR 50.34(f)(2)(ii)	Plant procedure improvement program	11.2
10 CFR 50.34(f)(2)(iii)	Control room human factors	11.2
10 CFR 50.34(f)(2)(iv)	Safety parameter display system	11.2
10 CFR 50.34(f)(2)(v)	Automatic indication of status of safety systems	11.2
10 CFR 50.34(f)(2)(vi)	High point venting of reactor coolant system (RCS)	Listed in DANU-ISG-2022-01, appendix B table 4 as applicable only if reactor coolant flow is credited for core cooling and coolant flow can be impeded by non-condensable gases. KU1 table 1.4-4 PSAR notes that this is not applicable to the design. The staff agrees with this assessment.
10 CFR 50.34(f)(2)(vii)	Radiation shielding design review	10.1
10 CFR 50.34(f)(2)(viii)	Post-accident sampling	KU1 table 1.4-4 notes that the design is partially compliant with this item. The staff's evaluation is in section 7.2.4 of this SE.
10 CFR 50.34(f)(2)(x)	Relief and safety valves – provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves.	Listed in DANU-ISG-2022-01 table 4 as applicable only if RCS has relief valves and failure of these valves would lead to core cooling challenges. KU1 table 1.4-4 notes that the design is partially compliant with this item. The staff's evaluation is in section 7.2.3 of this SE
10 CFR 50.34(f)(2)(xi)	Relief and safety valves – provide direct indication of relief and safety valve position in the control room	11.2

Table 1.11-3 Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
10 CFR 50.34(f)(2)(xiv)	Containment isolation	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. This is not applicable to the KU1 design because it uses a functional containment.
10 CFR 50.34(f)(2)(xv)	Containment purging	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. This is not applicable to the KU1 design because it uses a functional containment.
10 CFR 50.34(f)(2)(xvii)	Control room instrumentation for containment functions	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. Although KU1 uses a functional containment, KU1 PSAR Table 1.4-4 notes that portions of this item (i.e., those related to containment pressure, containment radiation intensity, and providing for effluent monitoring) are applicable. See section 11.2 of this SE.
10 CFR 50.34(f)(2)(xviii)	Coolant instrumentation	7.1.3, 11.2
10 CFR 50.34(f)(2)(xix)	Post-accident monitoring	7.6.5, 11.2
10 CFR 50.34(f)(2)(xv)	Provide an onsite technical support center and onsite operational support center	KU1 PSAR table 1.4-4 notes that the new EP rule does not require these specific facilities (see SE 11.3).
10 CFR 50.34(f)(2)(xxvi)	Leakage control outside containment	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. KU1 PSAR table 1.4-4 notes that the design uses a functional containment and not a single containment structure and that barriers and boundaries classified as safety-related will be leak testable to demonstrate their performance. See SE section 8
10 CFR 50.34(f)(2)(xvii)	In-plant Radiation Monitoring	10.1
10 CFR 50.34(f)(2)(xviii)	Preclude control room habitability issues during accidents	10.1
10 CFR 50.34(f)(3)(i)	Industry experience	11.2
10 CFR 50.34(f)(3)(ii)	QA list includes all SSCs important to safety	11.4
10 CFR 50.34(f)(3)(iii)	QA program	8.1

Table 1.11-3 Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
10 CFR 50.34(f)(3)(iv)	Dedicated containment penetrations	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. Not applicable to KU1 design because it uses a functional containment.
10 CFR 50.34(f)(3)(vi)	Containment	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. Not applicable to KU1 design because it uses a functional containment.
10 CFR 50.34(f)(3)(vii)	Management plan for design and construction activities	11.1