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TERRAPOWER, LLC. – FINAL SAFETY EVALUATION OF TOPICAL REPORT NAT-9394, "DESIGN BASIS ACCIDENT METHODOLOGY FOR EVENTS WITH RADIOLOGICAL RELEASE," REVISION 0 (EPID NO. L-2024-TOP-0009)

SPONSOR AND SUBMITTAL INFORMATION

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Brief Description of the Topical Report: By letter dated March 22, 2024, TerraPower submitted Topical Report (TR) TP-LIC-RPT-0007, “Design Basis Accident Methodology for Events with Radiological Release,” Revision 0 (ML24082A262), for the U.S. Nuclear Regulatory Commission (NRC) staff’s review. On April 22, 2024, the NRC staff determined that the TR provided sufficient information for the NRC staff to begin its detailed technical review (ML24107B046). On July 15, 2024, the NRC staff transmitted an audit plan to TerraPower (ML24197A156) and subsequently conducted an audit of materials related to the TR from July 23, 2024, to January 29, 2025. The NRC staff issued the audit summary dated June 6, 2025 (ML25157A115). On February 28, 2025, TerraPower submitted a revision of the TR (ML25063A329), which was renumbered from TP-LIC-RPT-0007 to NAT-9394, “Design Basis Accident Methodology for Events with Radiological Release,” Revision 0, to clarify portions of the TR as discussed in the audit summary.

NAT-9394, Revision 0, describes the methodology used to evaluate design basis accidents (DBAs) with the potential for radiological release for the Natrium reactor. This methodology consists of five discrete evaluation models (EMs), covering in-vessel transients, partial flow blockages, fuel misloads, fuel handling accidents (FHAs), and liquid sodium and gas leaks.

REGULATORY EVALUATION

Regulatory Basis

The regulations that are applicable to the review of this TR are:

- Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.34(a)(4) and 10 CFR 50.34(b)(4), which requires certain information to be submitted by applicants for construction permits and operating licenses, respectively. These sections require, in

Enclosure

part, analysis and evaluation of the design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from the operation of the facility and including the determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of the SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

- Regulation 10 CFR 50.43(e), which requires that reactor designs that differ significantly from light-water reactor designs licensed before 1997, or that use simplified, inherent, passive or other innovative means to accomplish their safety functions have an appropriate demonstration of their safety features. Sections 50.43(e)(1)(i) and (ii) require a demonstration of safety feature performance and interdependent effects through analysis, appropriate test programs, experience, or a combination thereof. Section 50.43(e)(1)(iii) requires that sufficient data exist regarding the safety features of the design to assess the analytical tools for safety analyses over a sufficient range of plant conditions, including certain accident sequences.

The NRC guidance documents that are applicable to the review of this TR are described below.

Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" (ML053500170), provides the evaluation model development and assessment process (EMDAP) as an acceptable framework for developing and assessing EMs for reactor transient and accident analyses. RG 1.203 outlines the four elements of an EMDAP, which is broken into 20 component steps. While the subject TR does not specifically reference RG 1.203, the NRC staff referenced various sections of RG 1.203 for best practices for EM development.[1]

For background, the Kemmerer Power Station Unit 1 (KU1) construction permit (CP) application (ML24088A059)[2] was submitted by TerraPower on behalf of US SFR Owner, LLC, for a Natrium reactor following the process outlined 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 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). This guidance defines 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 adequacy for non-light-water reactors. 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. DBAs are derived from DBEs by prescriptively assuming that only safety related (SR) SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34, "Contents of applications; technical information"

[1] TerraPower has developed methodologies for the Natrium design informed by RG 1.203, including methodologies for the analysis of DBAs without radiological release, partial flow blockage, and source term. The relationship between these methodologies and the DBAs with radiological release methodology is discussed in section 1.1, "Relationship to Other TerraPower TRs," of this SE.
[2] TerraPower, on behalf of US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, submitted the CP application for KU1 on March 28, 2024 (ML24088A059). The NRC staff's review of that CP application is ongoing. The staff is not making any determinations on the acceptability of the Natrium reactor design in this SE. The description of the Natrium reactor in this SE is based on the description in the TR.

dose limits, using conservative assumptions. The purpose of the subject TR is to provide a methodology for analyzing certain DBAs as defined in NEI 18-04.

TECHNICAL EVALUATION

1.0 INTRODUCTION

TerraPower requested that the NRC staff review the proposed methodology as an appropriate and adequate means for future applicants using the Natrium design (as described in the TR) to evaluate DBA events that potentially lead to radiological release. The DBAs considered in the TR can be broadly divided into in-vessel and ex-vessel scenarios. In-vessel scenarios include transients leading to fuel damage and subsequent release, flow blockage, FHAs, and loss of active cooling scenarios. Ex-vessel scenarios include FHAs, loss of active cooling scenarios, and radioactive sodium and gas leaks. The TR discusses assumptions, EM development, and EM assessment for five EMs to address these scenarios.

However, as noted in the executive summary of the TR, “[c]ertain aspects of the EM adequacy demonstration remain in development,” and as documented in the TR, there are portions of each EM which are incomplete. Thus, the NRC staff imposed limitations and conditions, provided at the end of the safety evaluation (SE), to address portions of the overall methodology which have not been completed.

1.1 Relationship to Other TerraPower TRs

The DBA with radiological release methodology TR is related to several other TerraPower methodology TRs that collectively provide a strategy for evaluating the consequences of potential accidental radiological releases for the proposed Natrium reactor design. TR section 4.1, “Background” provides a discussion of these relationships and includes TR figure 4.1-1, “EM Calculational Devices and Analysis Workflow,” which illustrates the connections between EMs.

The DBA with radiological release methodology does not identify the LBEs, DBAs, or other quantified event scenarios that result in radiological release for a given reactor licensing application. Rather, the LBEs, DBAs, and other quantified events appropriate for the licensing application are identified using the licensing modernization project methodology described in NEI 18-04. The DBAs are then analyzed using one of several methodologies. These methodologies include the DBA with radiological release methodology, described in this TR, the DBA without radiological release methodology, described in NAT-9390, “Design Basis Accident Methodology for In-Vessel Events without Radiological Release,” Revision 2, (ML24295A202) and evaluated by the NRC staff in ML25106A038, and the partial flow blockage methodology, described in NAT-9395, “Partial Flow Blockage Methodology,” Revision 0 (ML25129A064), which is undergoing review by the NRC staff.

The DBA with radiological release methodology is used to determine the extent of cladding or fuel failure, as well as quantifying liquid sodium or gas leaks, which are inputs into the source term methodology described in NAT-9392, “Radiological Source Term Methodology Report,” Revision 0 (ML24261B944), and evaluated by the NRC staff in ML25063A323. The output of the source term methodology is radiological releases to the atmosphere (source terms), which are input to the radiological consequence EMs described in TerraPower report NAT-9391, “Radiological Release Consequences Methodology Topical Report,” Revision 0, (ML24208A181) and evaluated by the NRC staff in ML25106A262, which is used to determine dose consequences associated with releases.

This TR also references NAT-2806-A, “Fuel and Control Assembly Qualification,” Revision 0, (ML24354A192) and TP-LIC-RPT-0011, “Core Nuclear and Thermal Hydraulic Design Report,” Revision 0, (ML24088A085) regarding fuel failure phenomena and steady state core analysis, respectively. TP-LIC-RPT-0011 is under review by the NRC staff as part of the KU1 CP application.

2.0 BACKGROUND

TR section 2.2, “Plant Description,” provides an overview of the Natrium reactor design. The Natrium reactor is a pool-type sodium-cooled fast reactor (SFR) with metal fuel. In the primary heat transport system, liquid sodium is transferred from the cold pool using mechanical primary sodium pumps to the lower plenum and through the reactor core, where it is heated. The hot sodium then enters the hot pool and transfers its heat via intermediate heat exchangers (IHXs) to the intermediate heat transport system (IHT) sodium loops before returning to the cold pool. Liquid sodium is circulated around the intermediate loops using mechanical intermediate sodium pumps (ISPs), which enables heat to be transferred from the core to a molten salt loop via a sodium-salt heat exchanger (SHX). This molten salt is pumped between the SHX and the energy island, where it can be stored and converted to electricity.

The Natrium plant’s safety related means of residual heat removal is the reactor air cooling system (RAC). The RAC cools the reactor by supplying natural draft outside ambient air down into the reactor cavity and past the outside of the reactor. The RAC is an open, passive system that is always in operation. The Natrium plant can also be cooled via the intermediate air cooling system (IAC). The IAC is non-safety related and serves as the normal shutdown cooling system. Each intermediate loop contains a sodium-to-air heat exchanger (AHX). Active forced circulation through both the IHT (via ISPs) and IAC (via air blowers) supports normal controlled cooling operations. If power is not available to support forced flow, the natural draft of air through the IAC can provide passive cooling.

The Type 1 fuel proposed for the Natrium core consists of metallic uranium-zirconium alloy slugs contained in right cylindrical fuel pins, arranged in a triangular pitch to form hexagonal fuel assemblies. Additional details regarding Natrium Type 1 fuel and its qualification are provided in NAT-2806-A.

TR section 2.2 additionally describes fuel handling and storage for the Natrium reactor. The Natrium design contains an in-vessel transfer machine (IVTM) and fuel transfer lift, which are installed during refueling outages. The IVTM moves fuel assemblies between the core, in-vessel fuel storage racks, and transfer station. The transfer station allows for fuel removal from the reactor vessel (RV) through the fuel transfer lift. The Natrium design also contains an ex-vessel fuel handling system, which transfers fuel entering the facility through inspection and conditioning and finally to the RV. The ex-vessel system also transfers irradiated assemblies from the RV to the ex-vessel storage tank (EVST). Irradiated assemblies in the EVST are eventually transferred to the pool immersion cell (PIC), where sodium residue is removed to allow for storage in water, and finally into the spent fuel pool (SFP). As separate water pool fuel handling system moves cleaned assemblies from the PIC into the SFP and transfers assemblies from the SFP into a cask for dry storage. Once the cask is prepared, it is transported to the long-term dry storage location.

The sodium processing system (SPS) is discussed in NAT-9392. The SPS controls and monitors primary and intermediate sodium chemistry, using cold traps and cesium traps to capture impurities and radionuclides from leaking or failed fuel. The sodium cover gas system (SCG) is not described in detail for the purposes of the TR and EM. However, the KU1 CP application states that the SCG controls, monitors, and supplies inert argon gas to various systems and components throughout the reactor building including the RV and gas spaces of the IHT.

3.0 ASSUMPTIONS

TR chapter 3, "Assumptions Requiring Verification," discusses the assumptions TerraPower used to define the scope of the TR EMs, determine conservative boundaries, or identify areas where future work is planned. The first two assumptions are applicable to all of the EMs discussed in the TR, while the other eleven are relevant to a specific EM. Assumption 3.1 states that event and accident scenarios considered in the TR will be limited to DBAs with potential release. Assumption 3.2 states that the TR is based on the current Natrium reactor design and will be updated as the design matures.

The NRC staff determined that these two assumptions are reasonable to apply to the five methodologies discussed in the TR, because they have all been developed specifically for analysis of DBAs for the Natrium design. Any applicant or licensee referencing this TR must justify that any departures from the Natrium design as described in the TR do not impact the conclusions of this TR or SE. This is captured in limitation and condition 1, below. Assumptions 3.3 - 3.12 are discussed in detail for each EM in section 5.0, "Event-Specific Methodologies," of the SE.

Assumption 3.13 states that the EM for in-vessel transients with radiological release assumes only Type 1 fuel will be used. The NRC staff determined that this assumption is reasonable and that it is necessary to limit the applicability of the in-vessel transients with release EM to Natrium Type 1 fuel or otherwise require an applicant or licensee referencing the TR to provide justification that using a different type of fuel does not affect the conclusions of the TR and this SE. The NRC staff additionally determined that this assumption should be applied to the partial flow blockage, FHAs, and fuel misload EMs, since these all also inherently assume Type 1 fuel is used. The assumption regarding the use of Type 1 fuel is captured as limitation and condition 1, below.

4.0 EM DEVELOPMENT AND ASSESSMENT

4.1 DBA Event Selection

In TR section 2.4, "DBA Event Selection," TerraPower discusses the identified DBAs for the Natrium design, broadly categorizing them into in-vessel core transients, local faults (e.g., partial flow blockages and fuel misloads), fuel handling events, and radioactive gas or liquid release events. TR table 2-1, "Natrium DBAs with Radioactive Material Release," lists the ten DBAs identified thus far which involve a potential release of radioactive material. TerraPower notes that these DBAs are provided to illustrate the methodology in the report, rather than define the set of events applicable to all Natrium plants. TerraPower additionally states that three DBAs associated with excessive sodium-water reaction in the PIC, loss of EVST cooling while storing fuel assemblies, and leakage from the gaseous radwaste processing system (RWG), respectively, are not covered by any EM contained within the TR. Accordingly, the NRC staff determined that, as this TR does not contain event-specific methodologies for these DBAs, future licensing submittals referencing this TR and using one of the contained EMs for these events will require further justification to ensure the selected EM is suitable. This is captured in limitation and condition 2, below.

TR section 2.4 additionally states that "DBAs which are not in-vessel are evaluated using the appropriate methodology in the DBA with release EM, an appropriate event-specific method, or evaluated with the source term EM using conservative assumptions." As such, the NRC staff determined that ex-vessel release analyses referencing this TR for their basis in future licensing submittals must provide sufficient detail to demonstrate that the appropriate methodology has been used. This is captured in limitation and condition 3, below.

4.1.1 PIRT Development

TR section 4.3, "Phenomena Identification and Ranking Tables (PIRT)," discusses the PIRTs developed to support the EMs contained in the TR. The PIRT concept is discussed in Step 4 of RG 1.203, which states that, as part of establishing the requirements for EM capability, key phenomena and processes should be identified and ranked with respect to their influence on the figures of merit (FOMs). This is accomplished by developing a PIRT. A given scenario is divided-up into characteristic time periods where dominant phenomena and processes remain relatively constant. For each time period, phenomena and processes are identified for each component. The phenomena and processes that the EM should simulate are determined by examining experimental data, expert opinion, and code simulations related to the specific scenario. After identification, the phenomena and processes are ranked by importance determined with respect to their effect on the relevant FOMs.

Throughout the TR, TerraPower refers to PIRTs performed for the following:

- Other qualified events (OQEs), which are developed based on accidents expected to have a frequency lower than beyond design basis events (BDBEs) and as such focused on unprotected in-vessel events
- Partial flow blockage within a subassembly
- Fuel and absorber pin behavior
- In-vessel DBAs without radiological release
- SPS leaks

The PIRT for fuel and absorber pin behavior, in-vessel DBAs without radiological release, and SPS leaks are discussed further in NAT-2806, NAT-9390, and NAT-9392, respectively, and considered by the NRC in their associated SEs. The PIRT for partial flow blockages is discussed further in NAT-9395, which is under NRC staff review. As such, the NRC staff focused its review on the OQE PIRT for this TR.

TR section 5.1.3, "EM Scope and Requirements," states that the in-vessel transients with radiological release EM is being developed to address the full scope of DBA, BDBE, and OQE events. This section further states that the identified phenomena from the OQE PIRT may be expected for more frequent BDBEs and are applicable and bounding for in-vessel DBAs. The OQE PIRT considered an unprotected loss of flow (ULOF), unprotected loss of heat sink (ULOHS), and unprotected transient over-power (UTOP). TerraPower states that a PIRT was developed for each event by a panel of internal and external experts and provides TR table 5-1, "Combined PIRT for ULOF, ULOHS and UTOP LBEs with Radiological Release with

High/Medium Importance Phenomena," which contains the combined results from the three event-specific PIRTs.

The NRC staff audited TerraPower's documentation detailing the OQE PIRT development process, and determined it aligns with the best practices discussed in RG 1.203, as is described in the TR. The NRC staff additionally determined that the PIRT phenomena are appropriate for the scenarios considered in the EM because they are consistent with the Natrium design and past SFR operating experience, and in the NRC staff's engineering judgment would be expected to be bounding for in-vessel DBAs.

4.1.2 EM Assessment Matrix Development

TR section 4.4, "Evaluation Model Assessment" discusses TerraPower's assessment plan for the EMs discussed in the TR. TerraPower states that for each EM with a PIRT, an assessment matrix will be created. TerraPower states that, based on the assessment matrix, testing needs will be identified. This is broadly consistent with the second principle of EMDAP, as discussed in RG 1.203, which is to develop an assessment base consistent with determined requirements. As RG 1.203 discusses, this assessment base is used to validate calculational devices or codes used by the EM, and may consist of legacy experiments or may require new experiments to be performed.

4.1.2.1 In-vessel Transients with Radiological Release Methodology

TR table 4-3, "Assessment Matrix for High/Medium Importance Fuel Failure Phenomena in OQEs," and table 4-4, "Assessment Matrix for High-Importance Fuel Failure Phenomena," list what test facilities may contribute test data for medium and high-ranked phenomena identified in their respective PIRTs. These assessment matrices include both legacy experiments as well as planned testing. TerraPower notes that these matrices focus on fuel failure phenomena, as the in-vessel transients with radiological release EM builds on the in-vessel transients without radiological release EM, which has its own assessment matrix documented in NAT-9390 and evaluated by the NRC staff in its associated SE. The NRC staff determined that TerraPower's approach to EM assessment development for fuel failure phenomena are acceptable because they align with the best practices discussed in RG 1.203, ensuring legacy experiments and planned testing to address medium and highly ranked phenomena are identified. However, the NRC staff has not determined the acceptability of the assessment matrices contained in tables 4-3 and 4-4, as they have not been completed. As discussed in limitation and condition 4, future licensing submittals referencing this TR will need to justify that code qualification, verification, and validation activities have been completed to a state that is appropriate for the intended licensing application.

4.1.2.2 Partial Flow Blockage Methodology

TR section 4.4 states that an assessment matrix for the partial flow blockage methodology is included in NAT-9395, which is under a separate NRC review. As the partial flow blockage methodology is undergoing a separate review, the NRC staff made no determination regarding the methodology in this SE.

4.1.2.3 Fuel Misload Methodology

TR section 4.4 states that fuel misloads do not include any phenomena beyond those normally modeled in steady state analysis, and thus TerraPower states that the fuel misload EM leverages the code qualification, verification, and validation activities discussed in NAT-2806-A and TP-LIC-RPT-0011. TerraPower states that these activities utilized for steady state core design, thermal hydraulics, and fuel performance serve the function of an assessment matrix for the fuel misload EM. The staff reviewed this and concluded it was reasonable, because past SFR experience indicates that misloads of the types considered by TerraPower in the fuel misload EM would not be expected to result in conditions that differ significantly from normal, steady state conditions. As such, the NRC staff determined that TerraPower's use of code qualification, verification, and validation performed for steady state calculations to support the assessment of the fuel misloads EM is acceptable because, in the staff's engineering judgment, the code qualification, verification, and validation activities expected to be needed for the steady state core design would also be applicable for fuel misload events.

4.1.2.4 FHA and Sodium Liquid and Gas Leak Methodology

TerraPower states that the PIRTs developed for the FHA EM and liquid sodium and gas leak EM in NAT-9392 focus on release and transport of radionuclides, rather than the dynamics and structural analysis or the calculation of leak rate and timing for the two EMs, respectively. The radiological source term methodology assessment base is described in NAT-9392. As such, assessment matrices have not been developed for these EMs in the DBAs with release TR. TerraPower states that assessment matrices for these EMs may be developed in the future. Therefore, the NRC staff made no determination on the EM assessment plans for the FHA and sodium leak and gas release EMs. As discussed in limitation and condition 4, future licensing submittals referencing this TR will need to justify that code qualification, verification, and validation activities have been completed to a state that is appropriate for the intended licensing application.

4.1.3 Quality Assurance

TR section 4.2 states that, for DBAs with potential radiological release, EM assessment is guided by TerraPower's "Acquired Software Quality Assurance Plan under Safety Analysis and Risk." TerraPower states that this plan provides a framework supporting quality assurance (QA) for software that perform safety related or non-safety related analyses. The plan discusses gap analysis and maturation activities while also including sections on commercial grade dedication plans for commercially acquired software used in safety related analyses. The NRC staff audited this document and determined it to be consistent with the discussion in this section of the TR. Additionally, in TR section 4.4, TerraPower states that EM assessment will follow TerraPower's QA program description discussed in TP-QA-PD-0001, "TerraPower QA Program Description," Revision 14-A (ML23213A199), which has been reviewed and approved by the NRC staff. The NRC staff determined that TerraPower's planned QA activities for the DBAs with radiological release methodology are appropriate and follow the program description provided in TP-QA-PD-0001.

5.0 EVENT-SPECIFIC METHODOLOGIES

TR chapter 5, "Event-Specific Methodology," outlines five EMs for different categories of DBAs with potential for radiological release. These EMs build on the EM development and assessment discussed in the previous chapters of the TR.

5.1 In-Vessel Transients with Radiological Release Methodology

TR section 5.1, "In-Vessel Transients with Radiological Release Methodology," outlines a methodology used to analyze DBAs that lead to cladding and fuel failures, resulting in the release of radionuclides to the primary coolant. This methodology is an extension of the methodology for in-vessel DBAs without radiological release, discussed in NAT-9390 and its associated SE.

5.1.1 Assumptions

TR section 5.1.2, "Assumptions," outlines the EM-specific assumption, reproducing TR section 3.1 item 3.13. Assumption 3.13 states that the EM for in-vessel transients with radiological release assumes only Type 1 fuel will be used. As discussed in SE section 3.0, "Assumptions," the NRC staff determined that assumption 3.13 is reasonable; this assumption is captured in limitation and condition 1, below.

5.1.2 Connection with the In-Vessel Transients without Radiological Release Methodology

TR table 4-1, "Figures of Merit for In-Vessel DBAs," discusses the three FOMs used in TerraPower's in-vessel transients without release methodology as described in NAT-9390 and its associated SE. These FOMs consist of fuel centerline temperature, coolant temperature, and acceptance criteria for peak cladding temperature (PCT) based on a time-at-temperature approach. The acceptance criteria for time-at-temperature no-failure (TATNF) for PCT accounts for strain, cladding wastage, and thermal creep. For in-vessel events without radiological release, TerraPower selected SAS4A/SASSYS-1 (SAS)[3] for its system analysis code. Final results from transient calculations in SAS are compared against these FOMs to determine if any limiting values are violated.

TR section 5.1.1, "Purpose and Scope," discusses the relationship between TerraPower's EM for in-vessel transients with radiological release and TerraPower's EM for in-vessel transients without radiological release. This section further states that, if TATNF is not violated for a transient, no radiological release occurs and thus the EM for in-vessel transients without release is sufficient. TerraPower states that TATNF contains various conservatisms such that, if violated, fuel does not necessarily fail. Upon a TATNF violation, **[[**
]] or the transient can be further analyzed through the Detailed Safety Analysis Workflow (DSAW).

[3] SAS is a physics simulation software developed by Argonne National Laboratory (ANL) to perform deterministic analysis of anticipated events and DBAs for SFRs. SAS is one-dimensional and composed of two computer codes, SAS4A and SASSYS-1. SAS4A contains detailed, mechanistic models of transient thermal, hydraulic, neutronic, and mechanical phenomena to describe the response of the reactor core, its coolant, fuel elements, and structural members to accident conditions. SASSYS-1 provides the capability to perform a detailed thermal-hydraulic simulation of the primary and intermediate sodium coolant circuits and the balance-of-plant steam-water circuit.

5.1.3 DSAW Description

DSAW is described in TR section 5.1.5, "In-Vessel Transient Evaluation Workflow," and illustrated in TR figure 5.1-1, "DSAW Data Flow." DSAW **[[**

]]. If the DSAW analysis of a given transient shows that these acceptance criteria are met, no fuel failure occurs. TR section 5.1.1 states that the DSAW does not allow for the **[[**

]]. As such, TR section 5.1.4, "EM Description," states that, if the DSAW results indicate assembly-wide fuel failures are expected, the **[[**

]].

TR section 9.1, "Appendix A – Additional Details of the DSAW Process," provides additional information on the DSAW process. **[[**

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The NRC staff audited the user and theory manuals for DSAW and its constituent codes, including **[[**

]].

The NRC staff determined that the DSAW process described in the TR is acceptable for evaluating fuel and cladding failure for the Natrium reactor design because DSAW has appropriate acceptance criteria for determining whether fuel or cladding would fail as the result of transient conditions and because the constituent codes have the capability of calculating these criteria. The NRC staff also determined that **[[** **]]** for conservatism is appropriate for the DSAW process. However, the NRC staff has not reviewed **[[** **]]**

for use in the DSAW process, though **[[ ]]**. The NRC staff also notes that the ultimate means of determining whether the DSAW process is adequately conservative is to compare the prediction of DSAW with applicable experimental data. Therefore, the NRC staff did not make a determination with respect to the conservatisms for DSAW described in the TR. This limitation is captured in limitation and condition 5.

5.1.4 **[[ ]]**

If DSAW indicates fuel or cladding failure is expected, **[[ ]]** using severe accident modules. TR section 5.1.4 discusses the modules TerraPower plans to use for addressing fuel or cladding failures that occur during in-vessel transients which consist of **[[ ]]**.

[[

]]. The NRC staff reviewed the **[[ ]]** code manual[5] regarding these modules and determined that their selection appears appropriate for TerraPower's intended use, with the exception of **[[ ]]**, which the staff did not make a determination on due to the module not being discussed in the code manual.

TR section 5.1.3 outlines three event phases associated with severe accidents, consisting of initiating, transition, and termination. The section further states that the range of phenomena associated with in-vessel transient DBAs would not transition to a severe accident as described in the initiating phase **[[ ]]**. TerraPower states that this EM is being developed to address the full scope of DBAs, BDBEs, and OQEs.

As discussed in TR section 5.1.3, DBAs for the Natrium reactor will not transition to a severe accident. The NRC staff reviewed this and determined that the applicability of this EM for licensing analyses is restricted to those events that do not experience severe accident phenomena **[[ ]]**. This limitation is captured in limitation and condition 6. However, the NRC staff notes that it may be appropriate to use this EM, including **[[ ]]**, for sensitivity studies or for analyses of transients which are more severe than DBAs, as justified.

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[4] **[[ ]]**.

[5] **[[ ]]**.

5.1.5 EM Assessment

TR section 5.1.6 presents an adequacy assessment of the in-vessel DBA with release EM, based primarily on assessment of **[[ ]]** and implementation of TerraPower's acquired software QA plan. The software QA plan is discussed in section 4.1.3 of this SE. Relevant discussion on assessment of **[[ ]]**, is provided in **[[ ]]**, where TerraPower's approach to assessing **[[ ]]** was determined to be acceptable, though additional work remains to be done. Beyond those aspects of **[[ ]]** included in **[[ ]]**, NAT-9394 includes additional models for **[[ ]]** as discussed in section 5.1.4 of this SE. As discussed in SE section 5.1.4 and limitation and condition 6, the in-vessel transients with radiological release methodology EM does not include accidents with severe accident phenomena and thus does not take advantage of the **[[ ]]**. Because of this, the evaluation provided in the NRC staff's SE for NAT-9390 is applicable to NAT-9394. As such the NRC staff finds the approach to assessing the adequacy of **[[ ]]** acceptable but notes that additional work is planned to complete the assessment; this aspect of the EM is thus subject to limitation and condition 4.

5.2 Partial Flow Blockage Methodology

TR section 5.2, "Partial Flow Blockage Methodology," provides a short overview of TerraPower's partial flow blockage methodology for the Natrium design. TerraPower references NAT-9395, that discusses the partial flow blockage methodology in detail and which is under a separate NRC review. The section further states that this summary was included to provide context on how the partial flow blockage methodology fits within the scope of the DBAs with radiological release methodology. TerraPower states that the partial flow blockage methodology is used to determine whether fuel or cladding fails. TR figure 4.1-1 shows that **[[ ]]**. As the partial flow blockage methodology is undergoing a separate review, the NRC staff made no determination regarding the methodology in this SE.

As described in SE section 3.0, the NRC staff also determined assumption 3.13 is applicable to the partial flow blockage EM; this assumption is captured in limitation and condition 1, below.

5.3 Fuel Misload Methodology

TR section 5.3, "Fuel Misload Methodology," discusses TerraPower's methodology for analyzing the consequences of having a fuel assembly in the wrong core location or in the wrong orientation. TR section 5.3.1, "Purpose and Scope," states that the Natrium core has two main enrichment zones, with the outer zone having higher enrichment to flatten power. **[[ ]]**.

5.3.1 Assumptions - Fuel Misload Methodology

TR section 5.3.2, “Assumptions,” outlines four assumptions applicable to the fuel misload methodology; these assumptions reproduce TR section 3.1 items 3.9 through 3.12.

Assumption 3.9 states that the steady state tools used to design the reactor core can model the misloaded core **[[**
]]. The NRC staff determined that this assumption is reasonable, noting that this expected for the fuel type and misloads considered in the EM.

Assumption 3.10 states that the final Natrium design **[[**
]]. The NRC staff determined this assumption is reasonable and that it is necessary to limit the applicability of the fuel misload EM to a design **[[**
]]. Limitation and condition 1a addresses this assumption.

Assumption 3.11 **[[**
]]. The NRC determined that this assumption is reasonable because **[[**
]]. The NRC staff also determined that **[[**
]].

Assumption 3.12 states that the misloaded assembly is **[[**
]]. The NRC staff determined that this assumption is reasonable in establishing **[[**
]].

As described in SE section 3.0, the NRC staff also determined assumption 3.13 is applicable to the fuel misload EM; this assumption is captured in limitation and condition 1, below.

5.3.2 Fuel Misload Phenomena

TR section 5.3.3, “EM Scope and Requirements,” discusses the one highly ranked phenomenon identified for the Natrium fuel misload methodology, which is the change in the local power distribution. The TR states that the magnitude of this change for a given misload event depends on the change of isotopic distribution of impacted pins between the intended and misloaded core configuration. In this section, **[[**
]].

5.3.3 Fuel Misload EM

TR section 5.3.4, "EM Description," discusses the EM for analyzing fuel misloads. The TR states that fuel misloads are analyzed using **[[**

]]. The TR states that the methodology for determining the limiting assembly and fuel pin for fuel misloads has not been finalized, with further work planned.

TR section 5.3.4 states that ten cases covering misloaded fresh, once-burnt, twice burnt, and thrice burnt fuel assemblies at beginning-of-life and beginning of equilibrium core conditions were analyzed in support of the KU1 CP application. **[[**

]]. The NRC staff audited the referenced fuel misload evaluations and found them to be consistent with the discussion in this section of the TR. The NRC staff did not make a determination regarding the acceptability of the CP application misload analyses.

The NRC staff determined that the fuel misload methodology is acceptable because **[[**

]]. The NRC staff made no determination on the acceptability of the referenced steady state core design methodology, which is undergoing a separate review.

5.4 FHA Methodology

TR section 5.4, "[FHA] Methodology," discusses TerraPower's methodology for analyzing structural-mechanical behavior that could lead to failure of dropped or impacted fuel assemblies. Transport and consequence of radiological release from FHAs is discussed in NAT-9392.

TR section 5.4.1, "Purpose and Scope," outlines five scenarios which could lead to fuel damage and radiological release:

1. Insertion or removal of a fuel assembly from reactor core
2. In-vessel fuel assembly movement
3. Ex-vessel fuel assembly movement between the EVST and washing station
4. Inadvertent action causing spent fuel assembly crush
5. Fuel assembly or loaded fuel cask drop in SFP

The TR states that these scenarios are analyzed to determine possible damage and final configuration for both the dropped and impacted fuel assemblies. The NRC staff determined that the events selected for consideration for FHAs are reasonable because they are consistent with the Natrium design discussed in TR section 2.2 and with the FHA events discussed in the NAT-9392.

5.4.1 Assumptions - FHAs

TR section 5.4.2, "Assumptions," outlines three assumptions applicable to the FHA methodology; these assumptions reproduce TR section 3.1 items 3.6 through 3.8. Assumption 3.6 states that, **[[**
]]. Assumption 3.7 states that, for scoping potential radiological release during a FHA, a fuel assembly with **[[**
]]. Assumption 3.8 states that, when performing a detailed analysis of a FHA, limiting conditions which result in the worst possible fuel damage and highest radiological release are considered. The NRC staff determined that these three assumptions are reasonable because: 1) assumption 3.6 is suitably conservative since it **[[**
]]; and 2) assumptions 3.7 and 3.8 are reasonable in establishing a conservative worst-case event for a given FHA.

As described in SE section 3.0, the NRC staff also determined assumption 3.13 is applicable to the FHA EM; this assumption is captured in limitation and condition 1, below.

5.4.2 FHA Acceptance Criteria and Phenomena

TR section 5.4.3, "Acceptance criteria," discusses acceptable performance for a dropped fuel assembly and any affected structures or targets during an FHA. Dropped fuel assemblies must not result in fuel cladding mechanical failure in either the dropped assembly or any impacted structures. Additionally, the dropped assemblies cannot create unacceptable core conditions that impact safe reactor operations, such as local criticality or reduced flow. The TR states that the TR FHA methodology only covers analysis of mechanical damage to affected assemblies and does not address the consequence of potential radiological release.

TR section 5.4.4, "EM Scope and Requirements," outlines potential phenomena that can cause mechanical damage to fuel assemblies during an FHA. The TR notes that **[[**

]]. TerraPower states that these phenomena include stress, strain, and loading limits of fuel assembly components, structural component fatigue, elastic and inelastic behavior of components under loading, and mechanical fracturing caused by dynamic loads or impacts. TerraPower states that the FHA EM should be able to model these phenomena.

5.4.3 FHA EM

TR section 5.4.5, "EM Description," describes the software for analyzing FHAs. A finite element analysis (FEA) software provides predictions on fuel assembly stress and impact forces for different fuel drop scenarios. These values are then compared to fuel assembly strength limits to determine the extent of fuel damage. The extent of fuel damage is then used to quantify the radiological release for the FHA. FEA software employs a finite element method, in which behavior of a system is solved by subdividing it into smaller parts, called finite elements. This is done via discretization in the space dimensions and implemented by the construction of a mesh of the object or system. The TR states that a finite element model of a Natrium fuel assembly was built for preliminary analysis of FHAs. Audit of NAT-5630, Rev. 0, "Finite Element Modeling and Analysis Methods for Core Assembly Drop Accidents," enabled the NRC staff to better understand the model used for Natrium FHA analyses.

TR table 4-2 states that **[[ ]]** are suitable for analyzing mechanical behavior during FHAs. During the audit, the NRC staff reviewed the technical manuals for the referenced FEA software, which appeared to be suitable for the EM. TR section 5.4.6, "EM Assessment," notes that TerraPower's assessment of the FEA software selected for Natrium's FHA EM is ongoing, with further work planned. Section 5.4.6 also discusses uncertainties that may arise when performing FEA for FHAs. These uncertainties consist of the geometric complexity of the fuel assembly and the difficulty in defining **[[ ]]**. TerraPower additionally states **[[ ]]**.

The NRC staff determined that the detailed mechanical approach for FHAs described in section 5.4 of the SE is not sufficiently developed for use in licensing analyses as **[[ ]]** and work required to assess FEA software for FHAs have not been completed. This limitation is captured in limitation and condition 7, below. However, the use of conservative bounding assumptions, **[[ ]]**, are acceptable for licensing applications, including support of a CP application, as they adequately bound worst possible radiological release for a given FHA.

5.5 Sodium Liquid and Gas Leak Methodology

TR section 5.5, "Sodium Liquid and Gas Leak Methodology," summarizes a methodology to determine and quantify leaks for dose calculations. The TR states that this analysis includes determining the extent of the leaks based on event initiation, including location, timing, system conditions, and propagation. The EM considers leaks from the SCG, SPS, and IHT. Leaks from the SCG would include **[[ ]]** which have leaked from the fuel. SPS leaks would contain **[[ ]]**, while leaks from the IHT would include **[[ ]]**.

The treatment of sodium liquid and gas leaks for the Natrium design is discussed in detail in NAT-9392. Accordingly, the NRC staff focused on aspects unique to this TR such as plans for determining suitable system leakage rates. In TR section 5.5.1, "Purpose and Scope," TerraPower states that a detailed methodology for mechanistically determining specific leak conditions has not yet been developed.

5.5.1 Assumptions - Sodium Liquid and Gas Leak Methodology

TR section 5.5.2, "Assumptions," outlines two assumptions applicable to the leak methodology; these assumptions reproduce TR section 3.1 items 3.3 and 3.4. Assumption 3.3 states that system leakage scenarios are assumed to occur during normal operation and not as a result of a different event. Assumption 3.4 presumes that **[[ ]]**. The NRC staff determined that these assumptions are reasonable, noting that, **[[ ]]**, any applicant or licensee referencing this TR must provide appropriate justification that the **[[ ]]** is suitably conservative. This limitation is captured by limitation and condition 5, below.

5.5.2 Sodium Liquid and Gas Leak EM Description

TR section 5.5.3, "EM Scope and Requirements," states that the EM established for analyzing sodium and gas leaks should have the capability of modeling important processes and phenomena identified during a representative PIRT. Phenomena associated with SPS leaks are detailed in table 2-5 of NAT-9392, which was evaluated by NRC staff in its associated SE, in which the NRC staff determined that the identified phenomena from the PIRT are reasonable for source term analysis for sodium leaks from the Natrium design.

TR figure 5.5-1, "Sodium Cleanup System and IHT Leak EM Diagram," provides an overview of the different components of the sodium liquid and gas leak EM. TerraPower states that the **[[** output from GOTHIC, MELCOR, or a manual calculation which feeds **]]** into the system leak rate is the portion of the EM relevant to the overall DBA with radiological release methodology. TR section 5.5.4, "EM Description," and section 5.5.5, "EM Assessment," provide a brief overview and assessment of the EM, referencing NAT-9392 for further information.

As discussed, the NRC staff focused its review of this methodology on plans for determining suitable system leakage rates. The NRC staff determined that the use of conservative bounding assumptions to determine a maximized release is reasonable for licensing applications. The NRC staff also determined that plans to potentially use **[[ ]]** to determine leakage rate is reasonable; however, the NRC staff made no determinations on the final implementation of this EM because, the TR states that the use of a detailed methodology for mechanistically determining specific leak conditions, such as sodium and gas leak rates and timing, has not yet been developed. As such, any applicants or licensees referencing the sodium liquid and gas leak methodology described in section 5.5 of this TR must appropriately justify that its selected sodium and gas leakage rates and timing are suitably conservative. This limitation is captured by limitation and condition 5, below.

5.6 EM Conservatisms

TR section 6.2, "EM Conservatism Summary," provides an overview of the conservative approaches TerraPower is taking for the different EMs included in this TR. For in-vessel transients with radiological release and partial flow blockages, the TR states that it will use conservatisms similar to those discussed in NAT-9390, including selecting conservative plant initial and boundary conditions within the operating band, assessing hot pin PCT within the sub-assemblies **[[ ]]**, and relying on conservatisms in the TATNF screening criteria and the DSAW process. TerraPower states that fuel misload analyses are adjusted for uncertainties in final temperature distribution **[[ ]]** to determine potential for fuel failures. For FHAs, **[[ ]]**. TR section 6.2 states that an effort is underway to determine that these approaches are sufficiently conservative for the Natrium design.

The NRC staff determined that TerraPower's approach to ensuring conservatism for the EMs discussed in this TR is reasonable. The NRC staff notes that the ultimate means of determining whether the EM is sufficiently conservative is to compare the prediction of the EM with applicable experimental data. Therefore, the NRC staff has not made a determination with respect to the final appropriateness of TerraPower's EM conservatisms. As discussed in limitation and condition 5, future licensing submittals referencing this TR must appropriately justify that the initial and boundary conditions and other input modeling parameter values are conservative.

LIMITATIONS AND CONDITIONS

The NRC staff imposes the following limitations and conditions on the use of this TR:

1. The NRC staff's determinations in this SE are limited to the Natrium design described in section 2.2 of the TR and this SE. An applicant or licensee referencing the EMs developed in this TR must justify that any departures from these design features do not affect the conclusions of the TR and this SE. Additionally, this methodology was developed to analyze certain DBAs as discussed in TR section 1.0 and this SE (and as defined in NEI 18-04); use of this methodology for other kinds of analyses must be justified.
    a. For the FHA EM, the NRC staff's determinations are limited to the Natrium design **[[ ]]**.
    b. For the in-vessel transients with radiological release, partial flow blockage, FHA, and fuel misload EMs, the NRC staff's determinations in this SE are limited to the Natrium design using Natrium Type 1 fuel.
2. As discussed in section 2.4 of the TR, the DBAs with radiological release methodology does not contain event-specific EMs for events associated with excessive sodium-water reaction in the PIC, loss of EVST cooling while storing fuel assemblies, and leakage from the gaseous radwaste processing system (RWG). Use of this methodology for these events requires further justification.
3. Section 2.4 of the TR states that "DBAs which are not in-vessel are evaluated using the appropriate methodology in the DBA with release EM, an appropriate event-specific method, or evaluated with the source term EM using conservative assumptions." As such, applications involving ex-vessel release analyses referencing this TR for their basis must provide sufficient detail to demonstrate that the methodology used is suitable.
4. An applicant or licensee referencing this methodology must submit documentation and justify that code qualification, verification, and validation activities have been completed to a state that is appropriate for the intended licensing application for each of the EMs discussed in the TR.
5. Consistent with section 6.2 of the TR, applicants or licensees referencing this methodology must appropriately justify that the initial and boundary conditions and other input modeling parameter values are conservatively selected. This includes the selection of **[[ ]]**.
6. As discussed in section 5.1.3 of the TR, the applicability of the in-vessel transients with radiological release methodology for licensing analyses is restricted to those events that do not experience severe accident phenomena (e.g., coolant boiling, gross cladding failure, significant fuel melting and relocation).
7. An applicant or licensee referencing the methodology described in TR section 5.4 for performing detailed mechanical analysis for FHAs must submit documentation and justify that the development and assessment of this methodology has been completed to a state appropriate for the intended licensing application.

CONCLUSION

The NRC staff has determined that TerraPower's TR NAT-9394, "Design Basis Accident Methodology for Events with Radiological Release," Revision 0, provides an acceptable approach to develop a methodology for use by future applicants utilizing the Natrium design as described in the TR and this SE to evaluate DBA events with radiological release because the assumptions made for each EM, EM development plans, selected calculational devices, planned conservatisms, and EM assessment plans are appropriate for analyzing the Natrium design, as discussed in this SE. This approval is subject to the limitations and conditions discussed in the previous section of this SE.

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