January 11, 2024

TP-LIC-LET-0117
Project Number 99902100

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Document Control Desk

Subject: Submittal of Approved TerraPower, LLC Topical Report: Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces

References:

1. U.S. Nuclear Regulatory Commission, TerraPower, LLC – Final Safety Evaluation for Topical Report NATD-LIC-RPRT-0001, "Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces," Revision 0 (ML23257A259)

The U.S. Nuclear Regulatory Commission (NRC) provided the final safety evaluation for the TerraPower, LLC Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces Topical Report in Reference 1.

Enclosure 1 to this letter provides the accepted version of the Topical Report with the additional content incorporated per the NRC staff request, designated NATD-LIC-RPRT-0001-A.

This letter and the associated enclosures make no new or revised regulatory commitments.

If you have any questions regarding this submittal, please contact Ryan Sprengel at rsprengel@terrapower.com or (425) 324-2888.

Sincerely,

Ryan Sprengel
Director of Licensing
TerraPower, LLC

Enclosures: TerraPower, LLC Topical Report NATD-LIC-RPRT-0001-A, Revision 0, Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces

cc: Mallecia Sutton, NRC

ENCLOSURE 1

TerraPower, LLC Topical Report NATD-LIC-RPRT-0001-A, Revision 0, Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces

TerraPower, LLC
15800 Northup Way
Bellevue, WA 98008

Natrium™ - A TerraPower & GE-Hitachi Technology

Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces

NATD-LIC-RPRT-0001-A

Revision 0

January 10, 2024

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 28, 2023

Mr. George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC
15800 Northup Way
Bellevue, WA 98008

SUBJECT: TERRAPOWER, LLC – FINAL SAFETY EVALUATION FOR TOPICAL REPORT NATD-LIC-RPRT-0001, “REGULATORY MANAGEMENT OF NATRIUM NUCLEAR ISLAND AND ENERGY ISLAND DESIGN INTERFACES,” REVISION 0 (CAC: 000431 / EPID: L-2022-TOP-0045)

Dear Mr. Wilson:

By letter dated October 4, 2022, TerraPower, LLC (TerraPower) submitted topical report (TR) NATD-LIC-RPRT-0001, “Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces,” Revision 0 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22277A824), for the U.S. Nuclear Regulatory Commission (NRC) staff’s review. The TR describes “an evaluation of the NRC regulations pertaining to the design interface of the Nuclear Island [NI] and Energy Island [EI] systems for the Natrium™ design” and includes a supporting summary of the Natrium reactor plant design, interfaces, safety features, and basic plant transient analysis, and TerraPower’s process for classifying structures, systems, and components (SSCs).

By email dated November 17, 2022 (ML22319A153), the NRC staff informed TerraPower that the TR provided sufficient information for the NRC staff to conduct a detailed technical review. On January 5, 2023, the NRC staff submitted an audit plan to TerraPower (ML22353A642) and subsequently conducted an audit of materials related to the TR from January 23, 2023, to March 10, 2023. The audit summary was issued on June 16, 2023 (ML23167A478).

The NRC staff has found that TR NATD-LIC-RPRT-0001, Revision 0, is acceptable for referencing in licensing actions to the extent specified and under the limitations and conditions delineated in the TR and in the enclosed final Safety Evaluation (SE). The final SE defines the basis for the NRC staff’s acceptance of the TR.

The NRC staff requests that TerraPower publish an approved version of this topical report within 3 months of receipt of this letter. The approved version should incorporate this letter and the enclosed SE after the title page. The approved version should include a “-A” (designating approved) following the TR identification symbol.

If you have any questions or comments concerning this matter, please contact Mallecia Sutton at (301) 415-0673 or via email at Mallecia.Sutton@nrc.gov.

Sincerely,

Signed by Jessup, William
on 09/28/23

William Jessup, Chief
Advanced Reactor Licensing Branch 1
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No.: 99902100

Enclosure:
As stated

cc: TerraPower Natrium via GovDelivery

SUBJECT: TERRAPOWER, LLC – FINAL SAFETY EVALUATION FOR TOPICAL REPORT NATD-LIC-RPRT-0001, "REGULATORY MANAGEMENT OF NATRIUM NUCLEAR ISLAND AND ENERGY ISLAND DESIGN INTERFACES," REVISION 0 (CAC: 000431 / EPID: L-2022-TOP-0045)
DATED: SEPTEMBER 28, 2023

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ADAMS Accession Nos.:
Package: ML23257A260
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Safety Evaluation: ML23257A258 **NRR-043**

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UNITED STATES
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WASHINGTON, D.C. 20555-0001

TERRAPOWER, LLC – FINAL SAFETY EVALUATION OF TOPICAL REPORT NATD-LIC-RPRT-0001, "REGULATORY MANAGEMENT OF NATRIUM NUCLEAR ISLAND AND ENERGY ISLAND DESIGN INTERFACES," REVISION 0 (EPID: L-2022-TOP-0045)

SPONSOR AND SUBMITTAL INFORMATION

Sponsor:	TerraPower, LLC
Sponsor Address:	15800 Northup Way, Bellevue, WA 98008
Project No.:	99902100

Submittal Date: October 4, 2022

Submittal Agencywide Documents Access and Management System (ADAMS) Accession No.: ML22277A824 [1]

Brief Description of the Topical Report: The topical report (TR) describes "an evaluation of the NRC regulations pertaining to the design interface of the Nuclear Island [NI] and Energy Island [EI] systems for the Natrium™ design" and includes a supporting summary of the Natrium reactor plant design, interfaces, safety features, and basic plant transient analysis, and TerraPower's process for classifying structures, systems, and components (SSCs). TerraPower stated in the TR that "[t]he independence of operation between the systems contained within the NI and the plant systems composing the EI is a key aspect of the Natrium design philosophy."

To support the implementation of this philosophy, "[t]he NI boundary conditions have been intentionally designed so the interrelationship with the EI does not impact the NI safety case." In the TR, TerraPower evaluated four regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) as they "pertain to the interfaces between the NI and EI systems and the design of the Natrium reactor." For two of the regulations, TerraPower intends to request exemptions and provided brief discussions of the planned technical and regulatory rationales for these exemptions. However, the exemptions themselves were not requested in the TR.

TerraPower requested NRC staff review and approval of the subject TR to "serve as a means, via reference, for Natrium reactor licensees to utilize the regulatory evaluation."

By email dated November 17, 2022 [2], (ML22319A153), the NRC staff informed TerraPower that the TR provided sufficient information for the NRC staff to conduct a detailed technical review [3]. On January 5, 2023, the NRC staff submitted an audit plan to TerraPower [4], (ML22353A642), and subsequently conducted an audit of materials related to the TR from January 23, 2023, to March 10, 2023. The audit summary was issued on June 16, 2023 [5], (ML23167A478).

REGULATORY EVALUATION

The NRC staff reviewed whether the regulatory analysis provided in the TR was consistent with the relevant statutes, regulations, guidance, and Commission policy, and supportable by the Natrium design in its current state as discussed in the TR.

The statute with requirements relevant to the operator and operator license issues discussed in the TR is the Atomic Energy Act of 1954, as amended (AEA), which states, in part, the following:

- Section 11, “Definitions,” Paragraph “r.”: “The term ‘operator’ means any individual who manipulates the controls of a utilization of production facility.”
- Section 107, “Operators’ Licenses”: “The Commission shall… prescribe uniform conditions for licensing individuals as operators of any of the various classes of production and utilization facilities licensed in this Act….”

The regulations evaluated by TerraPower in the TR and assessed by the NRC staff include:

- 10 CFR 50.10, “License required; limited work authorization,” which in the pertinent part discusses the types of activities that constitute construction and may not be conducted without a construction permit (CP) or limited work authorization;
- 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants” (also known as the Maintenance Rule), which provides requirements for monitoring nuclear power plant maintenance activities, including requirements regarding the scope of monitoring programs;
- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” which establishes quality assurance requirements relevant to nuclear power plants; and
- 10 CFR Part 55, “Operators’ Licenses,” which establishes requirements for operators’ licenses.

The NRC staff also considered the following regulations:

- 10 CFR 50.2, “Definitions,” which provides definitions for:
    - *Safety-related structures, systems and components*
    - *Controls* for nuclear reactors
- 10 CFR 50.12, “Specific exemptions,” which provides requirements associated with exemptions from the regulations
- 10 CFR 50.54, “Conditions of licenses,” specifically paragraphs (i) and (j). Paragraph 50.54(i) allows only licensed operators or senior operators to manipulate the controls of a reactor. Paragraph 50.54(j) provides requirements for the manipulation of “[a]pparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor[.]”

The guidance considered by the NRC staff includes:

- Regulatory Guide (RG) 1.206, Revision 1, “Applications for Nuclear Power Plants” [6]. Though this document primarily relates to applications submitted under the processes in 10 CFR Part 52, Section C.2.18 provides guidance on limited work authorizations (LWAs) and the types of activities constituting or not constituting “construction” under 10 CFR 50.10, with examples.

- RG 1.160, Revision 4, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” [7], which provides guidance on implementation of the Maintenance Rule (10 CFR 50.65).
- RG 1.233, Revision 0, “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” [8], which provides guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-light-water reactors (non-LWRs). This RG endorses NEI 18-04, Revision 1, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development” [9], which provides an approach to selecting licensing basis events (LBEs), classifying structures, systems and components (SSCs), and assessing defense-in-depth (DID).

TECHNICAL EVALUATION

1.0 NATRIUM PLANT DESIGN AND TRANSIENTS

The plant’s design is fundamental to the transients that may occur at the facility, the plant response to those transients, and the safety classification of the SSCs that prevent and mitigate those transients. These are significant considerations in evaluating TerraPower’s regulatory analyses seeking to demonstrate the independence of the Natrium NI and EI. Sections 1.1 through 1.8 of this safety evaluation (SE) provide context for the regulatory analyses assessed in Section 3.0 of this SE. The NRC staff considered the plant design, as discussed in Sections 3, 4, and 5 of the TR, and the transient response of the reactor, as discussed in Section 7 of the TR, and highlighted those design features that are significant to the independence of the Natrium NI and EI. The NRC staff has placed a condition on the use of this TR, as outlined in the Limitations and Conditions section of this SE, to ensure that these key design features are appropriately addressed in future licensing applications or correspondence referencing this TR.

1.1 Fuel and core

As discussed in Sections 2, 3, 4, and 5 of the TR, the proposed Natrium reactor is a metal-fueled, pool-type sodium fast reactor (SFR). The fuel and core are located on the NI. The proposed initial Natrium core uses fuel composed of metallic uranium alloyed with 10 weight percent zirconium (U-10Zr). The fuel is formed in slugs and inserted into fuel rods made of HT9, a martensitic stainless steel alloy. The fuel is bonded to the fuel rod cladding with sodium, which fills the gap between the fuel and cladding and improves thermal conductivity from the fuel out to the coolant. These fuel rods are arranged into assemblies and inserted into the core. Other types of assemblies in the core include control assemblies, reflector assemblies, and shield assemblies. The fuel operates in the fast neutron spectrum. The core is cooled by liquid metallic sodium, which circulates between cold and hot pools within the vessel as discussed in more detail in Section 1.2 below.

The NRC staff determined that the designs of the fuel and core, as described in the TR, are key aspects of the plant for NI-EI independence. The metallic fuel has a high thermal conductivity and thus peak fuel temperatures are expected to be close to coolant temperatures and below the coolant boiling point. The NRC staff expects the fuel system to have substantial margin to safety limits, even during transients, and beneficial reactivity feedback effects; these features are necessary to avoid damage to the fuel system because of transients. Additionally, the fuel system has beneficial characteristics for retention of radionuclides in the event of an accident. All these aspects of the fuel system's behavior are necessary for NI-EI independence because they improve safety margins and prolong transients, allowing NI systems to respond to mitigate them without reliance on the EI.

1.2 Primary heat transport system and reactor vessel

Section 5.1 of the TR describes the reactor vessel and the Natrium primary heat transport (PHT) system. The reactor vessel and PHT are located on the NI. In the PHT, liquid sodium is circulated around the reactor vessel by mechanical primary sodium pumps (PSPs), which take suction from the cold pool and discharge into the core inlet. From there, the sodium is heated by the core and flows into the hot pool. Hot sodium then flows downward through the shell side of the intermediate heat exchangers (IHXs) where its energy transfers into the intermediate heat transport (IHT) system. The cooled sodium then enters the cold pool and completes the flow circuit. An illustration of the PHT system is in Figure 1 of the TR.

The reactor vessel is surrounded by a guard vessel, which is designed to contain sodium leaks in the event of a breach in the reactor vessel. Between the reactor vessel and the guard vessel is a small gap filled with inert gas.

The NRC staff determined that the design of the primary heat transport system, as described in the TR, is a key aspect of the plant for NI-EI independence because (a) it contains the coolant and provides the geometry necessary for coolant to flow past the core, (b) it provides the primary pathway for heat to be removed from the fuel by the reactor air cooling (RAC) and intermediate air cooling (IAC) systems (discussed below in Section 1.6), and (c) it provides thermal inertia that insulates the reactor core from upsets on the EI.

1.3 Intermediate heat transport system

Section 5.1 of the TR also discusses the IHT system. The IHT is located on the NI. In the IHT system, liquid sodium is circulated by mechanical intermediate sodium pumps (ISPs) from the IHX tube side to the tube side of the sodium-salt heat exchangers (SHXs). There are two IHT loops, each of which is connected to one IHX and two SHXs. Between the IHX and the SHX, the sodium passes through sodium-air heat exchangers (AHXs), which are a part of the IAC system. The IHT also contains expansion and drain tanks to handle the sodium. The IHT system can be seen in TR Figure 2.

The NRC staff determined that the design of the IHT system, as described in the TR, is a key aspect of the plant for NI-EI independence because (a) it provides a pathway for heat to be removed from the fuel by the IAC (discussed below in Section 1.6) and (b) it provides additional thermal inertia that reduces the effects on the reactor core resulting from transients on the EI.

1.4 Thermal salt system and boundary between NI and EI

Section 5.1 of the TR states that the four SHXs allow heat to be transferred from the IHT system, which contains sodium, to the thermal salt system (TSS), which contains a molten salt similar to that used in concentrated solar systems. In the TSS, variable speed mechanical pumps take suction from the cold salt tank and pump molten salt through the shell side of the SHXs, where it is heated by the IHT, and into a hot salt storage tank. In the SHXs, salt pressure is higher than sodium pressure, so leakage across the SHX tubes would be salt from the TSS into the sodium in the IHT. The TSS is on both the NI and EI. As discussed in Sections 5.1 and 7 of the TR, the boundary between the EI and NI is provided by drain isolation valves on the inlet and outlet of the salt side of the SHXs. The SHXs, a small amount of salt piping, and the salt isolation valves are therefore considered to be part of the NI, while the salt tanks and the salt piping on the EI-side of the isolation valves are considered to be part of the EI. As discussed in Section 7 of the TR, the drain isolation valves close in response to transients, isolating the NI from the EI.

The NRC staff determined that the TSS, as described in the TR, is a key aspect of the plant for NI-EI independence because it plays a key role in allowing the NI and EI to operate independently. The NI portion of the TSS, which includes the SHXs, the salt drain isolation valves, and the piping in between, may be relied on for safety; this portion of the TSS allows the NI to be isolated from the EI to mitigate effects of failures in the EI portion of the salt system on the SHXs. Also, the large volume in the thermal salt tanks in the EI portion of the TSS reduces the effects on the NI resulting from transients in the steam generation, power conversion, or other EI systems (e.g., power output changes, turbine trips). However, the NRC staff notes that while the TSS plays a key role in the independent operation of the NI and EI, the EI portion of the TSS is not relied on for the safety of the reactor. This is because the NI can effectively respond to transients regardless of the condition of the EI portion of the TSS, as discussed in Section 7 of the TR and Section 1.8 of this SE.

1.5 EI systems

EI systems are discussed in Section 5.2 of the TR. These include the TSS, the steam generation system, the condensate and feedwater system, the turbine and generator and associated systems, and the heat rejection system. A diagram of some of these systems is provided in Figure 5 of the TR.

Salt is pumped from the hot and cold salt tanks (mixed to maintain desired temperatures) through an economizer, evaporator, superheater, and reheater, to generate steam to run the turbine. Salt that has deposited energy in the steam generation system is returned to the cold salt tank so it may be reheated by the SHXs.

Steam used to drive the turbine goes through a condenser, where it is subcooled, and passes through feedwater heaters and a deaerator before being pumped back to the steam generation system. The turbine itself is planned to be a commercially available steam turbine. Steam can bypass the turbine to the condenser. The condenser is cooled by a circulating water system that rejects heat to the atmosphere via a mechanical forced draft cooling tower.

The NRC staff determined that EI systems other than the TSS are not significant to NI-EI independence, because their effects on the core are reduced by the thermal inertia provided by the TSS, IHT system, and PHT system.

1.6 Reactor air cooling and intermediate air cooling systems

The reactor air cooling (RAC) and intermediate air cooling (IAC) systems are discussed in Section 5.1 of the TR and illustrated in TR Figure 3. These systems provide residual heat removal capability for the Natrium reactor.

The RAC is a natural draft system that utilizes ducts to guide outside air past the guard vessel. Heat from the core is transported to the reactor vessel by a mix of convection (either forced or natural depending on the state of the plant) and conduction, then from the reactor vessel to the guard vessel primarily by radiation[1]. The exterior surface of the guard vessel is cooled by a mixture of convection from the RAC air flow and radiation to surrounding structures (primarily the collector cylinder), which are also cooled by the RAC air flow. The RAC is a fully passive system with no dampers and is therefore always in operation.

The IAC, as mentioned previously in Section 1.3 of this SE, consists of an AHX on the IHT system along with blowers and dampers on the air side of the AHX. In active mode, the IAC provides controlled, forced flow of air across the AHX that allows heat to be transferred from the IHT to the atmosphere. In passive mode, the dampers are designed to fail open so that natural draft flow of outside air can remove heat from the IHT without active systems, as a supplement to the RAC.

As discussed in Section 4.2 of the TR, the IAC (in forced flow mode) is intended to serve as the normal shutdown cooling system, while the RAC is designed to be able to remove all decay heat from the reactor using purely passive means. The NRC staff notes that because radiation heat transfer is a key contributor to RAC performance, the reactor may need to heat up for the RAC to remove heat effectively. The NRC staff audited preliminary analyses performed by TerraPower that included these systems and verified that they appear to be capable of performing the necessary heat removal function, based on the current design. The NRC staff stresses, however, that the analyses audited were preliminary and are subject to confirmation as design details are developed and finalized.

The NRC staff determined that the RAC and IAC, as described in the TR, are a key aspect for NI-EI independence because they provide the capability to remove all decay heat using systems located solely on the NI.

1.7 Reactor control and protection systems

Reactivity control is accomplished via control rods and inherent feedback mechanisms, as discussed in part in Section 1.1 of this SE. Shutdown is accomplished via control rods. As stated in Section 4.1 of the TR, all sensors monitoring reactor trip parameters, except for some seismic sensors, are located on the NI. A non-safety related anticipatory power runback feature is also included in the design which inserts the control rods automatically to reduce power and attempt to avoid a scram, providing additional time and margin to respond to transients.
However, as discussed in Section 4.1 of the TR, if a reactor runback or scram fails, the core is designed such that many anticipated transients can be accommodated without scram.

The NRC staff determined that the reactor control and protection systems, as described in the TR, are a key aspect for NI-EI independence because they are designed to be able to shut down the reactor without relying on SSCs, including sensors, located on the EI.

1.8 Natrium transients

TerraPower stated in Section 7 of the TR that no EI SSCs are required to (1) respond to mitigate any events impacting the NI, (2) support safety-related (SR) SSCs, or (3) ensure DID adequacy. The response to transients impacting the NI can be accomplished solely with

[1] The inert gas in the gap between the reactor vessel and guard vessel may also provide a minor contribution to heat transfer through conduction or natural convection. However, radiation is expected to be the dominant means of heat transfer across the gap.

equipment located on the NI. Section 7 of the TR also describes the response to transients initiated on the EI.

Specifically, from the NI perspective, all transients on the EI can be grouped into either increases in heat removal by the salt system (increased salt flow, low salt temperature) or decreases in heat removal by the salt system (decreased salt flow, high salt temperature, low salt pressure). These manifest as a decrease or increase, respectively, in the IHT cold leg temperatures. With no action, this eventually causes an increase or decrease, respectively, in the core inlet temperature. TerraPower stated the thermal inertia of the IHT and PHT is such that changes in salt conditions can be adequately responded to using only signals monitored within the NI; this was verified by the NRC staff in the audit. As discussed in Section 7.1 of the TR, the plant isolates the NI from the EI after a scram or runback by closing the SHX drain isolation valves, so there is no prolonged effect on the NI from any EI systems. As discussed in Section 1.6 of this SE, TerraPower's preliminary analyses further show that removal of all decay heat can be accomplished solely with NI systems (RAC and IAC).

The NRC staff therefore determined that, based on the summary of Natrium design and analyses presented in the TR, the NI has the capacity to effectively respond safely to transients, regardless of whether they are initiated on the NI or EI, using only NI systems. This plays a key role in the safety classification of SSCs on the EI, which is discussed in further detail in Section 2.0, below.

2.0 NATRIUM SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

As discussed in Section 1 of the TR, the Natrium reactor licensing approach follows NEI 18-04 [9], which was endorsed by the NRC staff in RG 1.233 [8]. NEI 18-04 provides a risk-informed and performance-based process for determining the safety classification of SSCs at nuclear reactor facilities. The safety classification process in NEI 18-04 is highly integrated with the rest of NEI 18-04, which also provides processes for selecting LBEs to be included in the plant safety analysis[2] and ensuring adequate DID. These other processes will only be discussed here to the extent that they have a direct nexus to the SSC classification.

The safety classification of an SSC is used to determine the standards to which it is designed, fabricated, and maintained. Additionally, certain regulatory requirements apply to SR SSCs and other SSCs that are deemed important to safety; some of these requirements will be discussed later in this SE.

Figure 6 of the TR provides a flow chart with the NEI 18-04 SSC classification process. Additionally, a high-level overview of the ultimate safety-classification categories is provided in Section 4 of NEI 18-04. The categories are:

[2] The LBE selection and evaluation process described in NEI 18-04 identifies the transients that must be analyzed for the facility based on the plant design and a probabilistic risk assessment (PRA). Transients are assigned to LBE categories based on frequency. Anticipated operational occurrences are events with frequency greater than 10-2 per year; design basis events (DBEs) are those with frequency between 10-4 and 10-2 per year; and beyond design basis events (BDBEs) are those with frequency between 5*10-7 and 10-4 per year. The LBEs are evaluated against a frequency-consequence curve derived from regulatory requirements for dose. Design-basis accidents (DBAs) are derived from the DBEs by analyzing them to ensure that they meet the 10 CFR 50.34 dose limits, applying conservative assumptions and only crediting equipment selected to be safety-related in the analysis.

- SR, which includes:
    - SSCs selected[3] to perform safety functions required to mitigate the consequences of DBEs to within the frequency-consequence target,
    - SSCs selected to mitigate DBAs (which rely solely on SR SSCs) to meet the dose limits of 10 CFR 50.34 using conservative assumptions, and
    - SSCs selected to prevent the frequency of high-consequence BDBEs[4] from increasing into the DBE region and beyond the frequency-consequence target curve
- Non-safety related with special treatment (NSRST), which includes:
    - Non-safety-related SSCs relied on to perform risk-significant functions; that is, those functions that prevent or mitigate any LBE from exceeding the frequency-consequence target or make significant contributions to the cumulative risk metrics (discussed in more detail in Section 3.3.6 of NEI 18-04), and
    - Non-safety-related SSCs relied on for DID adequacy
- Non-safety related with no special treatment (NST), which includes all other SSCs.

In addition to these definitions provided in NEI 18-04, the NRC staff clarified in RG 1.233 that the NRC expects that “SSCs that provide essential support (including required human actions) for SR or NSRST SSCs will be classified in a manner consistent with the higher-level function, even if the supporting SSC is not explicitly modeled in the PRA.”

TerraPower determined that all SSCs on the EI would be NST according to the NEI 18-04 process because they are not needed to meet the required safety functions, do not provide risk-significant functions, and are not needed for DID. As described in Section 1.0 of this SE, the NRC staff reviewed TR summaries of proposed Natrium design features and basic plant transient responses as well as associated supporting analysis during a regulatory audit. With respect to DID considerations, TerraPower indicated in TR Sections 4.1 through 4.3 that there are multiple NI SSCs capable of accomplishing the plant safety functions and providing adequate DID without involving the EI. Based on the NRC staff’s review of this information, the NRC staff determined that a future applicant could be able to justify characterizing EI systems as NST for a Natrium design with these specified high-level features. The need for future applicants and licensees referencing the TR to demonstrate that EI SSCs are appropriately classified as NST is listed as one of the limitations and conditions in the Limitations and Conditions section of this SE.

Additionally, the NRC staff noted in RG 1.233 that the definition of “safety-related” SSCs in NEI 18-04 differs from that provided in 10 CFR 50.2. While RG 1.233 determined that the process provided in NEI 18-04 was acceptable, it also noted that “[a]pplicants referencing this RG are

[3] The term “selected” is used because multiple, overlapping sets of SSCs may be able to carry out the necessary safety functions. Thus, one set of SSCs that is necessary and sufficient to perform these functions is selected by the designer, and other SSCs that are capable of performing the same safety functions may be considered for their risk significance or if they are needed for DID.

[4] “High-consequence BDBEs” are BDBEs whose consequences exceed the 10 CFR 50.34 dose limits.

expected to use the terminology in NEI 18-04 and as needed, identify exceptions to and exemptions needed from NRC regulations." If TerraPower does not comply with 10 CFR 50.2 as stated, they must request an exemption; however, neither compliance with, nor an exemption from, the definition of safety-related SSCs in 10 CFR 50.2 was discussed in the TR. This issue is discussed further in the Limitations and Conditions section of this SE.

3.0 TERRAPOWER REGULATORY ANALYSES

3.1 Analyses of 10 CFR 50.10

Pursuant to 10 CFR 50.10(c), "[n]o person may begin the construction of a production or utilization facility on a site on which the facility is to be operated until that person has been issued either a CP… or a limited work authorization [LWA] …." The 10 CFR 50.10(a) definition of "construction" is divided in two parts: 10 CFR 50.10(a)(1) specifies activities deemed to constitute "construction," and 10 CFR 50.10(a)(2) specifies activities which are excluded from the definition.

Paragraph 10 CFR 50.10(a)(1) specifies the following definition of construction:

> (1) Activities constituting construction are the driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of foundations, or in-place assembly, erection, fabrication, or testing, which are for:
>
> (i) Safety-related structures, systems, or components (SSCs) of a facility, as defined in 10 CFR 50.2;
> (ii) SSCs relied upon to mitigate accidents or transients or used in plant emergency operating procedures;
> (iii) SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function;
> (iv) SSCs whose failure could cause a reactor scram or actuation of a safety-related system;
> (v) SSCs necessary to comply with 10 CFR part 73;
> (vi) SSCs necessary to comply with 10 CFR 50.48 and criterion 3 of 10 CFR part 50, appendix A; and
> (vii) Onsite emergency facilities, that is, technical support and operations support centers, necessary to comply with 10 CFR 50.47 and 10 CFR part 50, appendix E.

In Section 8.1 of the TR, TerraPower evaluated the EI SSCs against the criteria in 10 CFR 50.10(a)(1) to determine whether construction of EI SSCs would constitute "construction" under the regulation and would therefore require NRC authorization through a CP or LWA. The NRC staff notes that the analyses provided in the TR only relate to the 10 CFR 50.10 definition of "construction" and do not address definitions of "construction" elsewhere in NRC regulations (e.g., 10 CFR Part 51). This is highlighted as a limitation and condition in the Limitations and Conditions section of this SE.

10 CFR 50.10(a)(1)(i) - (iii)

TerraPower stated in the TR that, since the Natrium design includes only NST SSCs on the EI, does not rely on any NST SSCs to provide mitigation for an accident or transient, and does not have any NST SSCs whose failure could prevent SR SSCs from fulfilling their safety-related function, the 10 CFR 50.10(a)(1)(i) through (iii) criteria do not apply to SSCs on the EI.

Regarding 10 CFR 50.10(a)(1)(i), since the criterion applies only to SR SSCs the NRC staff finds it reasonable in principle to exclude NST SSCs (with no special treatment) from consideration. However, the regulation specifically states that “construction” includes safety-related SSCs of a facility “as defined in 10 CFR 50.2”. As noted previously in Section 2.0 of this SE, the TR does not address compliance with or exemption from the 10 CFR 50.2 definition of “safety-related SSCs.” If TerraPower seeks an exemption from the 10 CFR 50.2 definition of “safety-related SSCs,” TerraPower must also seek an exemption from 10 CFR 50.10(a)(1)(i). If compliance with the 10 CFR 50.2 definition is demonstrated, an exemption may not be needed. This is highlighted in the Limitations and Conditions section of this SE.

With respect to 10 CFR 50.10(a)(1)(ii), TerraPower stated that the Natrium plant design does not rely on any NST SSC – which includes all SSCs on the EI, as discussed above – to provide mitigation for an accident or transient. The NRC staff considers this reasonable to partially address criterion (ii), provided that the EI SSCs remain appropriately categorized as NST as the design matures; this is addressed in the discussion provided in Section 2.0 of this SE and in the Limitations and Conditions section of this SE. TerraPower’s logic is also consistent with the design and transient response of the reactor as discussed in SE Sections 1.1 through 1.8. However, TerraPower did not address plant emergency operating procedures (EOPs), which are included in the regulation. Since the NRC staff has not yet reviewed EOPs for the Natrium facility, future licensing applications or correspondence referencing this TR must demonstrate that EI SSCs are not used in the EOPs to support a conclusion that no EI SSCs meet this aspect of the 10 CFR 50.10(a)(1)(ii) criterion; this is included in the Limitations and Conditions section of this SE.

TerraPower dispositioned 10 CFR 50.10(a)(1)(iii) by stating that there are no NST SSCs in the Natrium design whose failure could prevent SR SSCs from fulfilling their safety-related function. This is reasonable, and the NRC staff finds it to be logically consistent with the definition of “safety-related” from NEI 18-04, in that any SSCs whose failure could prevent an SR SSC from fulfilling its safety-related function would not be characterized as NST under the risk-informed process. However, as stated in SE Section 2.0, the definition of “safety-related SSCs” in NEI 18-04 differs from the definition in 10 CFR 50.2. Compliance with, or an exemption from, the 10 CFR 50.2 definition is addressed in the Limitations and Conditions section of this SE.

10 CFR 50.10(a)(1)(v) – (vii)

TerraPower determined that criterion (v) is not applicable to EI SSCs. With respect to physical security program SSCs, TerraPower stated in the TR that: “[c]onstruction activities for SSCs necessary to comply with 10 CFR 73 include the preparation and building of physical barriers and structures and associated hardware and detection systems for the physical security program. None of these physical security program SSCs are located on the EI for the Natrium reactor.” With respect to cyber security program SSCs, TerraPower stated in the TR that no digital components or control systems identified as critical digital assets (CDAs) would be installed or activated prior to receipt of a CP. While the NRC staff finds this to be plausible based on the safety significance of SSCs on the EI, TerraPower did not provide detailed information on how the physical and cyber security programs would be implemented in such a way that they would deliberately avoid involving the EI. Thus, additional justification must be provided in future licensing applications or correspondence referencing this TR to support a conclusion that no EI SSCs meet the 10 CFR 50.10(a)(1)(v) criterion; this is included in the Limitations and Conditions section of this SE. The NRC staff also determined that, contrary to TerraPower’s overall conclusion, criterion (v) is applicable to CDAs on the EI; however, TerraPower’s proposed disposition of this issue, which is to defer installation of CDAs on the EI until after the CP is issued, is appropriate.

TerraPower determined that criterion (vi) is not applicable to EI SSCs because fires on the EI would not prevent the ability to achieve and maintain safe shutdown of the reactor. The NRC staff reviewed TerraPower's assessment of 10 CFR 50.10(a)(1)(iv) and determined that it was adequate based on the classification of the EI systems as NST and the NRC staff's review and conclusions documented in Section 1.0 of this SE that the Natrium reactor has the capability for safe shutdown using only NI SSCs.

TerraPower also evaluated criterion (vii) as not applicable to the EI because onsite emergency facilities are not located on the EI. This is consistent with plant facilities layouts provided to the NRC staff in public meetings (e.g., in [10]) and could reasonably be a design objective. However, as with criterion (v), additional justification must be provided in future licensing applications or correspondence referencing this TR to support a conclusion that no EI SSCs meet the 10 CFR 50.10(a)(1)(vii) criterion, as discussed in the Limitations and Conditions section of this SE.

10 CFR 50.10(a)(1)(iv)

Contrary to the other criteria, TerraPower determined that the 10 CFR 50.10(a)(1)(iv) criterion is applicable to SSCs on the EI. The NRC staff agrees with this assessment. The analysis provided by TerraPower in the TR indicates that failures in the TSS can cause a reactor scram, and this may be the case for other EI SSC failures if the failures are not resolved within a certain period (which is defined by the amount of thermal salt available in the cold salt tank). Since this criterion is applicable, TerraPower intends to submit a request for an exemption from 10 CFR 50.10(a)(1)(iv).

TerraPower stated in Section 8.2 of the TR that the rationale for requesting an exemption from 10 CFR 50.10(a)(1)(iv) would be the same as that for requesting an exemption from 10 CFR 50.65(b)(2)(iii). TerraPower based this determination on discussion in the LWA rule issuance (72 FR 57415), which stated that the criteria in 10 CFR 50.10(a)(1) were intended to encompass those SSCs which "have a reasonable nexus to radiological health and safety or common defense and security." As further stated in the rule issuance, the NRC chose to base the criteria in 10 CFR 50.10(a)(1)(i) through (iv) on the scoping criteria used in 10 CFR 50.65(b) in part because "[the] definition is well understood and there is good agreement on its implementation." RG 1.206 [6], additionally indicates that it is acceptable to apply guidance developed for the maintenance rule to determine which SSCs are within the scope of the definition of construction, for criteria (i) through (iv). While the NRC staff understands that an exemption from these regulations could have the same basis, due to the similar language and underlying basis of the regulations, no such exemption has yet been submitted. The NRC staff is not taking a position in this SE on any prospective exemption request the NRC might receive.

3.2 Analyses of 10 CFR 50.65

Section 50.65 of 10 CFR requires licensees to have a program that monitors the performance or condition of certain SSCs or demonstrates the performance or condition of these SSCs through appropriate preventative maintenance to provide reasonable assurance that they are capable of fulfilling their intended functions. The scope of SSCs that must be subject to this program is defined in 10 CFR 50.65(b):

> (b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related [SSCs], as follows:

(1) Safety-related [SSCs] that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

(2) Non-safety related structures, systems, or components:
   (i) That are relied upon to mitigate accidents or transients or are used in plant [EOPs]; or
   (ii) Whose failure could prevent safety-related [SSCs] from fulfilling their safety-related function; or
   (iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

The requirements of 10 CFR 50.65(b)(1) are outside the scope of the NRC staff's evaluation of TerraPower's regulatory analysis, which focused on 10 CFR 50.65(b)(2).

TerraPower determined that criteria (i) and (ii) do not apply to SSCs located on the EI for the same reasons 10 CFR 50.10(a)(1)(ii) and (iii) were determined to be not applicable. The conclusions drawn by the NRC staff in SE Section 3.1 are applicable to these criteria as well. TerraPower also stated in the TR that the Natrium reactor does not use any NST SSC in EOPs. However, as previously discussed, the NRC staff has not yet reviewed EOPs for the Natrium facility; therefore, future licensing applications or correspondence referencing this TR must provide additional information to justify a conclusion that EI SSCs are not used in the EOPs, as listed in the Limitations and Conditions section of this SE.

TerraPower also determined that criterion (iii), which is similar to 10 CFR 50.10(a)(1)(iv), is applicable at least to the portion of the TSS located on the EI. Under this regulation, the portion of the TSS on the EI would be subject to the maintenance rule because a failure in this system would lead to a runback that, if failed, would cause a scram. The NRC staff agrees with this assessment, which is the same as that discussed in SE Section 3.1. Also, as discussed in Section 3.1 of this SE, TerraPower intends to seek an exemption from 10 CFR 50.65(b)(2)(iii). The NRC staff is not taking a position in this SE on any prospective exemption request the NRC might receive.

3.3 Analyses of 10 CFR Part 50, Appendix B

Section 8.4 of the TR provides TerraPower's evaluation of 10 CFR Part 50, Appendix B, (hereafter referred to as Appendix B). TerraPower states that Appendix B provides quality assurance requirements for the design, manufacture, construction, and operation of SSCs, and that the requirements of Appendix B apply to all activities affecting the SR functions of those SSCs. TerraPower determined that, since SSCs located on the EI will be classified as NST according to the NEI 18-04 process, they do not affect the SR functions of SSCs used for mitigation and are therefore not subject to the requirements of Appendix B.

The introduction to Appendix B states, in part:

> Nuclear power plants… include SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, manufacture, construction, and operation of those SSCs. The pertinent requirements of this appendix apply to all activities affecting the safety-related

> functions of those SSCs; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

Accordingly, Appendix B is applicable to “all activities affecting the safety-related functions” of SSCs that “prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.” The NRC staff considered that systems classified as NST under the NEI 18-04 process would not be involved in preventing or mitigating accidents, based on the risk-informed safety classification process, and would furthermore not be involved in supporting the SR functions of SSCs that are used in the prevention or mitigation of accidents. However, as stated in SE Section 2.0, the definition of “safety-related SSCs” in NEI 18-04 differs from the definition in 10 CFR 50.2. Compliance with, or an exemption from, the 10 CFR 50.2 definition is addressed in the Limitations and Conditions section of this SE. Thus, the NRC staff determined that TerraPower’s evaluation of the requirements of Appendix B, with respect to NST SSCs on the EI is acceptable, subject to this limitation and condition. The NRC staff’s determination in this regard is based on TerraPower’s summary of its preliminary implementation of the NEI 18-04 SSC classification process and is subject to confirmation; this is also documented in the Limitations and Conditions section of this SE.

3.4 Analyses of 10 CFR Part 55

TerraPower states in TR Section 8.2 that the Natrium design removes direct interaction between the nuclear reactor and the main turbine generator and that, due to the lack of direct interaction, operation of the main turbine generator is not an apparatus or mechanism whose manipulation directly affects the reactivity or power level of the reactor. TerraPower goes on to state that, on this basis, the Natrium design allows for a non-licensed individual to fully operate and control the main turbine generator.

The NRC staff evaluated TerraPower’s regulatory analysis in TR Section 8.2 within the context of the regulations of 10 CFR Parts 50 and 55, the associated regulatory history, and relevant statutory requirements. The NRC staff notes that the AEA defines operators under Section 11 as individuals who manipulate the controls of utilization facilities and that it does not define what those controls consist of, thereby leaving that definition to be made by the NRC. The AEA further mandates under Section 107 that individuals who operate utilization facility controls must be licensed by the NRC.

From the inception of operator licensing in 1956 (21 FR 359), manipulation of the controls of a utilization facility was restricted to licensed operators under 10 CFR 50.54(i). This specific regulation is closely linked to the AEA Section 107 mandate that is discussed above. The original 1956 definition of “controls” (21 FR 6) was very broad and encompassed “mechanisms which by manipulation or failure to manipulate singly or in combination could result in the release of atomic energy or radioactive material in amounts determined by the Commission to be sufficient to cause danger to the health and safety of the public.”

However, in 1963 (28 FR 3197), the agency narrowed this definition of “controls,” stating that “[t]his narrower interpretation… is in accord with the original Commission intent.” This amended definition of “controls” (which remains unchanged to the present day) is limited in scope to “apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor.” A separate 1963 rulemaking (28 FR 3196) also introduced 10 CFR 50.54(j) which, in contrast with 10 CFR 50.54(i), does not limit the manipulation of apparatus and mechanisms other than controls to performance by licensed operators and senior operators, but states, “Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to part 55 of this chapter present at the controls.” Thus, the regulations recognize a distinction between an apparatus or mechanism whose manipulation directly affects the reactivity or power level of the reactor, and therefore is a control, and a non-control apparatus or mechanism whose operation may affect the reactivity or power level of the reactor and is subject to 10 CFR 50.54(j).

Thus, the word “direct,” as used in the definition of “controls,” is central to understanding the meaning of the related requirements of 10 CFR 50.54(i) and (j). In analyzing this, the NRC staff recognized that substituting wording of equivalent meaning for “direct” yielded a working interpretation of these requirements that was suitable for use in evaluating the TR. Specifically, "controls" can be interpreted to mean apparatus and mechanisms that, when manipulated, affect the reactor power level or reactivity without also needing something intermediate to make that happen. Manipulations of this type fall under the scope of 10 CFR 50.54(i) and their performance is restricted to licensed operators and senior operators. Thus, the presence, or absence, of a significant intermediary between any given manipulation and the reactivity or power level effects on the reactor is the key factor that the NRC staff, in its judgement, identified as being the essential determinant of whether that manipulation falls under the scope of 10 CFR 50.54(i). The NRC staff note this overall characterization is consistent with the Commission perspective expressed under 28 FR 3197 which disfavored an excessively broad definition of what constitutes the “controls” of a facility.

The NRC staff noted that the TSS provides thermal energy storage capacity equivalent to several hours of full electrical generation, such that reactor power is not directly correlated to EI steam generation over significant timeframes. The Natrium design thus enables the reactor and EI steam loads (such as a turbine used for electric power generation) to operate at power levels that are different from one another. The NRC staff evaluated the implications of this design configuration and determined that the TSS acts as a significant intermediary between manipulations involving EI steam loads and reactivity effects on the reactor. Based upon this, the NRC staff concluded that manipulations of Natrium apparatus and mechanisms that affect EI steam loads do not directly affect the reactivity or power level of the reactor and, therefore, do not fall under the scope of 10 CFR 50.54(i). TerraPower did not discuss compliance with 10 CFR 50.54(j) in its TR; the NRC staff addresses this regulation in the Limitations and Conditions section of this SE.

LIMITATIONS AND CONDITIONS

The NRC staff identified the following limitations and conditions, applicable to any licensee or applicant referencing this TR:

1. The NRC staff’s review identified key aspects of the Natrium plant design and transient response that play a significant role in the independence of the Natrium NI and EI, as discussed in Sections 1.1 through 1.8 of this SE. Applicants or licensees referencing this TR must propose a Natrium design with these key aspects or justify that any departures from these key aspects do not affect the conclusions in the TR and this SE.

2. The conclusions reached in this SE are not valid if a process other than that described in NEI 18-04 is used to perform the safety classification or to the extent that any SSCs on the Natrium EI are found to have a safety classification other than NST. Thus, applicants or licensees referencing this TR must confirm, with appropriate justification, that all EI SSCs are classified as NST using NEI 18-04. Such justification must provide, or reference and make available for audit, the analyses supporting EI SSC classifications.

3. The TR does not address differences between the 10 CFR 50.2 and NEI 18-04 definitions of “safety-related SSCs” and does not discuss whether TerraPower plans to demonstrate compliance with the 10 CFR 50.2 definition or seek an exemption. Applicants or licensees referencing this TR must comply with the 10 CFR 50.2 definition or propose an exemption. If an applicant or licensee referencing this TR proposes an exemption from 10 CFR 50.2, an exemption to 10 CFR 50.10(a)(1)(i) must also be proposed, since that regulation references the 10 CFR 50.2 definition.

4. The NRC staff’s review of what constitutes “construction” is limited to the provisions of 10 CFR 50.10 and does not address other definitions of “construction” (e.g., those in the environmental regulations in 10 CFR Part 51 or other non-NRC regulations).

5. Applicants or licensees using this TR as a basis for non-applicability of 10 CFR 50.10(a)(1)(ii) or 10 CFR 50.65(b)(2)(i) to the EI must provide additional information demonstrating that EI SSCs are not used in the EOPs.

6. Applicants or licensees using this TR as a basis for non-applicability of 10 CFR 50.10(a)(1)(v) to the EI must provide detailed information demonstrating that the physical and cyber security programs would be implemented in such a way that the EI would not include any SSCs that meet the 10 CFR 50.10(a)(1)(v) criterion (i.e., not include any “SSCs necessary to comply with 10 CFR part 73,” or otherwise not construct the EI SSCs necessary to comply with 10 CFR Part 73 until a CP has been issued).

7. TerraPower asserted in the TR that onsite emergency facilities necessary to comply with 10 CFR 50.47 or 10 CFR Part 50 Appendix E and facilities for providing onsite emergency first aid and decontamination are not located on the EI. Applicants or licensees using this TR as a basis for non-applicability of 10 CFR 50.10(a)(1)(vii) must provide detailed information to demonstrate this assertion.

8. This TR does not address the requirements of 10 CFR 50.54(j), and the NRC staff does not provide any evaluation in this SE of the implications of the Natrium design as it relates to this specific regulation. Thus, any Natrium facility licensee or applicant for an operating license (OL) or combined license (COL) referencing this TR must, in the absence of receiving an exemption, ensure that manipulation of any EI apparatus or mechanism which may affect the reactivity or power level of the reactor is only permitted to occur with the knowledge and consent of a licensed operator or senior operator.

9. Under 10 CFR 55.31(a)(5), reactivity manipulations for operator license applicant experience requirements must involve operating “controls” which, as discussed in Section 3.4 of this SE, are associated with direct reactivity or power changes. Therefore, any apparatus or mechanism determined to not be a “control” must, logically, also be excluded from being acceptable for applicant experience credit under 10 CFR 55.31(a)(5). Consistent with this, applicants for operator and senior operator licenses at a Natrium facility where the facility licensee (or applicant for an OL or COL)

references this TR cannot rely upon the manipulation of apparatus and mechanisms that affect EI steam loads (to include a turbine used for electrical generation) for the purposes of satisfying the operator license applicant experience requirements of 10 CFR 55.31(a)(5).

10. The NRC staff is not taking a position in this SE on any prospective exemption request that the NRC might receive on matters discussed in the TR.

CONCLUSION

This TR provides an evaluation of the NRC regulations pertaining to the design interface between the Natrium NI and EI and includes a supporting summary of key aspects of the Natrium reactor plant design, interfaces, safety features, and basic plant transient analysis, and TerraPower's process for classifying SSCs. The NRC staff reviewed the key aspects and TerraPower's evaluation of select regulatory requirements and concluded that the TR is acceptable for use in licensing applications subject to the Limitations and Conditions in this SE.

REFERENCES

[1] R. Sprengel, letter to U.S. Nuclear Regulatory Commission, “Submittal of TerraPower, LLC Topical Report, Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces,” dated Oct. 04, 2022 (ML22277A824).

[2] M. Sutton, email to G. Wilson, “TerraPower: Completeness Determination for the Natrium Nuclear Island and Energy Island Design Interfaces Topical Report,” dated Nov. 17, 2022 (ML22319A153).

[3] U.S. Nuclear Regulatory Commission, “NRC 898 - Topical Report Completeness Determination for TerraPower, LLC Natrium Nuclear Island and Energy Island Topical Report,” dated Nov. 16, 2022 (ML22319A131).

[4] M. Sutton, “‘Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces’ Audit Plan,” dated Jan. 05, 2023 (ML22353A642).

[5] M. Sutton, email to G. Wilson, "TerraPower: Natrium EI-NI Separation TR Audit Summary," dated June 16, 2023 (ML23167A478).

[6] U.S. Nuclear Regulatory Commission, “Applications for Nuclear Power Plants, Revision 1,” RG 1.206, dated Oct. 3, 2018 (ML18131A181).

[7] U.S. Nuclear Regulatory Commission, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 4,” Regulatory Guide 1.160, dated Aug. 31, 2018 (ML18220B281).

[8] U.S. Nuclear Regulatory Commission, “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,” Regulatory Guide 1.233, Revision 0, dated June 30, 2020 (ML20091L698).

[9] Nuclear Energy Institute, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, Revision 1,” NEI 18-04, dated Aug. 29, 2019 (ML19241A472).

[10] R. Sprengel, letter to the U.S. Nuclear Regulatory Commission, “TerraPower Plant and Licensing Strategy Overview Presentations to the Advisory Comittee on Reactor Safeguards,” dated Mar. 31, 2023 (ML23090A228).

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Approval

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EXECUTIVE SUMMARY

This Topical Report describes an evaluation of the Nuclear Regulatory Commission (NRC) regulations pertaining to the design interface of the Nuclear Island (NI) and Energy Island (EI) systems for the Natrium™ reactor design, a TerraPower, LLC (TerraPower) and GE-Hitachi technology. TerraPower is requesting NRC review and approval of this regulatory assessment for use by Natrium reactor applicants.

Acronyms

Acronym	Definition
ARDP	Advanced Reactor Demonstration Program
CFR	Code of Federal Regulations
CRD	Control Rod Drive Mechanism System
CSS	Core Support Structure
DID	Defense in Depth
DOE	Department of Energy
EBR	Experimental Breeder Reactor
EI	Energy Island
EOP	Emergency Operating Procedure
ESS	Energy Island Salt Transport System
FFTF	Fast Flux Test Facility
FSAR	Final Safety Analysis Report
GEH	GE-Hitachi Nuclear Energy Americas, LLC
GV	Guard Vessel
LMP	Licensing Modernization Project
IAC	Intermediate Air Cooling
IHT	Intermediate Heat Transfer System
IHX	Intermediate Heat Exchanger
IPE	Individual Plant Examination
ISP	Intermediate Sodium Pump
IVS	In-Vessel Storage
LBE	Licensing Basis Event
LWA	Limited Work Authorization
NEI	Nuclear Energy Institute
NI	Nuclear Island
NRC	Nuclear Regulatory Commission
NST	No Special Treatment
NSRST	Non-Safety-Related with Special Treatment
PHT	Primary Heat Transport System
PSP	Primary Sodium Pump
RAC	Reactor Air Cooling
RCC	Reactor Core Components
RES	Reactor Enclosure System
RG	Regulatory Guide
RI	Reactor Internals
RIS	Reactor Instrumentation System
RIPB	Risk-Informed and Performance-Based
RPS	Reactor Protection System
RXB	Reactor Building
RSA	Reactor Support Assembly
RV	Reactor Vessel
RVH	Reactor Vessel Head
SFR	Sodium Fast Reactor

SGS	Steam Generation System
SHX	Sodium-Salt Heat Exchanger
SR	Safety-Related
SSC	Structure, System, and Component
TSS	Thermal Salt Storage System

1 INTRODUCTION

In October 2020, the Department of Energy (DOE) selected the Natrium Sodium Fast Reactor (SFR) design for the Advanced Reactor Demonstration Program (ARDP).

TerraPower is utilizing the methodology described in Nuclear Energy Institute (NEI) 18-04 Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," as endorsed by 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," for the selection of licensing basis events (LBEs); classification and special treatments of structures, systems, and components (SSCs); and assessment of defense-in-depth (DID).

The independence of operation between the systems contained within the NI and the plant systems composing the EI is a key aspect of the Natrium design philosophy. The NI boundary conditions have been intentionally designed so the interrelationship with the EI does not impact the NI safety case. Thermal energy storage and steam generation operations are independent from reactor power operations due to the Thermal Salt Storage System (TSS). This separation of plant areas allows the power production systems on the EI to be separate entities with respect to site design and quality standards while being integrated through the TSS. The isolation point for the thermal salt storage system (EI to NI isolation) is at the inlet valve (input from the cold salt storage tank) and outlet valve (output to the hot salt storage tank) of the sodium to salt heat exchangers.

TerraPower evaluated the regulatory impacts of the Natrium design interfaces with respect to the interaction of NI and EI systems and identified four regulations of interest. These regulations are described below in the Regulatory Analyses section of the report.

TerraPower requests NRC review and approval of this TR, which contains an evaluation of these four regulatory requirements as it pertains to the interfaces between the NI and EI systems and the design of the Natrium reactor. This Topical Report will serve as a means, via reference, for Natrium reactor licensees to utilize the regulatory evaluation.

2 NATRIUM REACTOR DESCRIPTION

The Natrium reactor is a metal fueled, pool-type Sodium Fast Reactor (SFR) that takes advantage of the simple and robust safety profile of SFRs to reduce the complexities associated with nuclear design and construction. Safety functions are made integral to the reactor vessel, and support equipment is moved to separate structures, resulting in a simplified reactor building. The superior heat transfer characteristics of sodium and low-pressure plant operations permit the use of compact and lightweight equipment, unlike in other reactor types cooled with pressurized water or gas.

The higher operating temperatures and constant thermal output provide an ideal match for thermal energy storage using molten salt, a mature and proven technology that is commercially deployed in the concentrated solar power industry. Hot sodium from the reactor transfers its heat to an intermediate sodium loop and eventually to a molten salt loop, which carries heat off the NI to the EI where it can be stored, converted into electricity, or used for industrial process heating. This design architecture minimizes the size of the NI and allows the reactor to operate at constant conditions while the EI meets variable energy demands. The rating for the Natrium reactor is 840 MW thermal. The EI has the capability to produce up to 500 MW net electric.

3 BASIC PLANT DESIGN AND INTERFACES

The Natrium reactor was developed using insights from decades of research, design, and development leveraging the legacy of 40 reactor-years from the Experimental Breeder Reactor (EBR) -II and Fast Flux Test Facility (FFTF) operations. Insights were also gained from more recent designs such as GEH's PRISM technology and TerraPower's Traveling Wave Reactor technology. Previous SFR operating experience has demonstrated that the SFR technology can passively accommodate severe transients which challenge traditional Light Water Reactor technology. These inherent safety characteristics are leveraged in the Natrium design to reduce the envelope of safety-related (SR) SSCs.

The EI was designed specifically for energy storage to allow the Natrium plant to vary its supply of energy provided based on overall grid conditions. This feature allows the Natrium plant to provide a utility-scale carbon free solution that can make a meaningful impact on efforts to mitigate climate change and complement the increased use of intermittent renewable energy technology (e.g., solar, wind).

The reactor operates at near atmospheric pressure, circulating sodium through its core by pumps. Heat is transferred from the hot primary sodium pool to the intermediate sodium piping loop by means of two intermediate heat exchangers, which are located inside the reactor vessel. The Intermediate Heat Transfer System (IHT) uses two IHT pumps to move sodium through the two intermediate heat exchangers, via piping penetrations running through the reactor vessel head, to two sodium/salt heat exchangers per loop. The sodium/salt heat exchangers and IHT pumps are located in the reactor auxiliary building. Cold salt is pumped from the thermal salt storage cold tank through the four sodium/salt heat exchangers (two per loop) and is returned as hot salt back to the thermal salt storage hot tank. The thermal salt storage hot tank serves as thermal energy storage and is located within the EI systems.

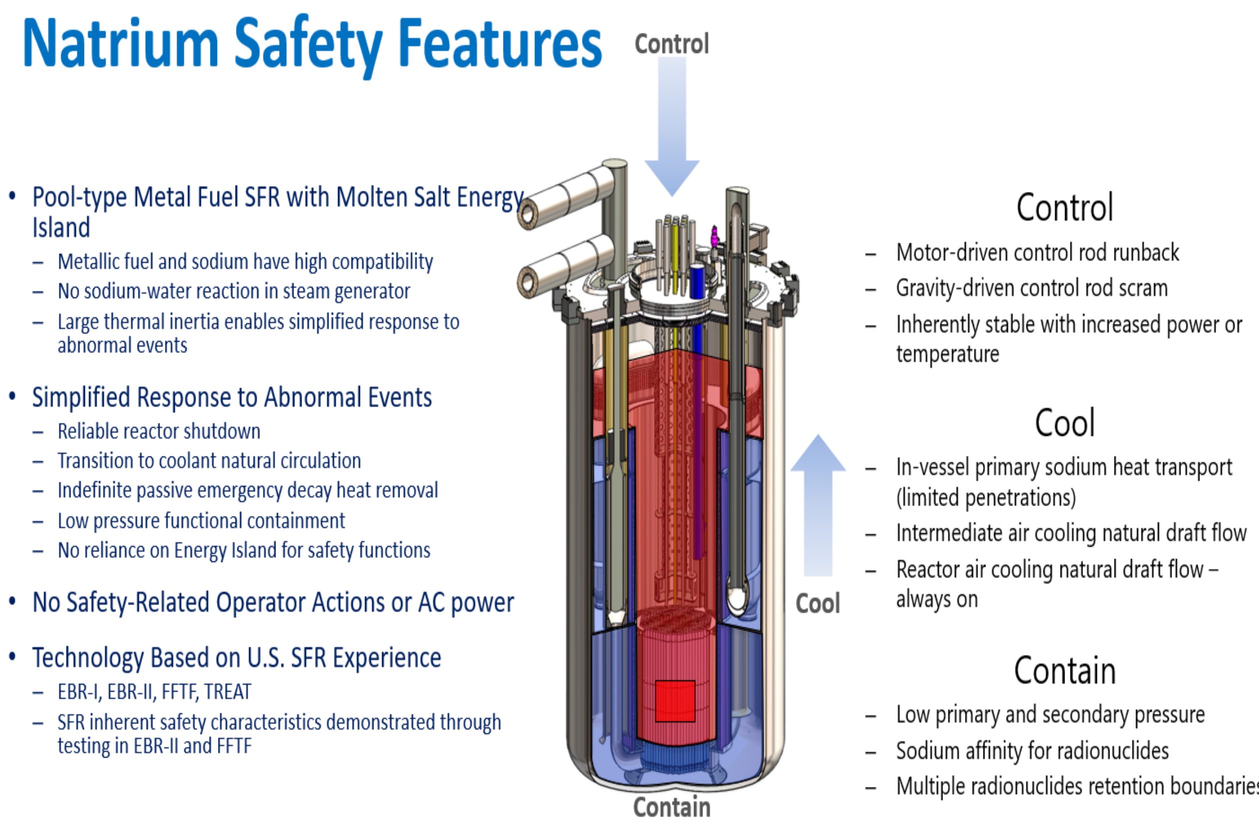
The salt stored in the thermal salt storage hot tank is then used to generate steam for use in commercially available steam turbine generators or industrial process heating. This is accomplished by pumping the hot salt from the thermal salt storage hot tank through the steam generators and returning the salt to the thermal salt storage cold tank. The steam generator equipment converts water into steam by passing the hot molten salt through an economizer (water preheater), evaporator, superheater, and reheater and provides that steam to the Turbine/Generator. This technology is essentially the same as molten salt systems used in the concentrated solar power industry.

The operational flexibility to move the output of the NI separately to the energy storage and electric generation portions of the EI enables the steam turbine and reactor to operate at different power levels. The duty cycle of the reactor is low because it is isolated from grid demand changes. This enables the NI to operate at 100 percent, 24/7 at a capacity factor greater than 90 percent. Meanwhile, this operational flexibility allows the EI to adjust power output to meet real-time grid demands and accommodate for the variability of renewable technologies.

As described below in the SSC classification and plant transient analyses sections, the NI is considered separate from the EI from a safety case perspective. Parameters such as salt pressure, temperature, and flow, are bounded in the safety analysis and are designed to be insignificant contributors to overall nuclear safety risk.

4 NATRIUM SAFETY FEATURES

Figure 1: Natrium Safety Features



4.1 Reactivity Control

The Natrium reactor has been designed to accomplish the control of reactivity function with multiple layers of protection (i.e., DID); the first is active, the second is passive, and the third is inherent.

The reactor control system acts as a buffer to prevent the need for a scram. It detects abnormal operation and initiates a runback via motor driven insertion of neutron absorbing control rods to achieve a softer shutdown than a scram.

The reactor protection system (RPS) exists to initiate a scram should the reactor control system fail, or a properly initiated runback fails to prevent the reactor from reaching a scram setpoint. The scram function results in passive gravity insertion of control rods into the reactor core. All sensors monitoring Natrium reactor trip parameters, except for the seismic sensors, are located on the NI.

The core is designed such that the reactor can accommodate anticipated transients without scram for events such as loss of primary flow, loss of heat sink, and uncontrolled rod withdrawal. The natural feedbacks are self-regulating (inherent in the fuel design) and find a low power level at which heat production and heat removal are in balance.

4.2 Cooling - Residual Heat Removal

The Natrium reactor has been designed to accomplish the residual heat removal function with multiple layers of protection (i.e., DID); the first is active, the second is passive, and the third is inherent. The Natrium reactor utilizes two diverse residual heat removal methods in the Reactor Air Cooling (RAC) and Intermediate Air Cooling (IAC) systems. This design has been selected due to the passive, reliable, and sustainably safe nature of RAC with DID provided by IAC. RAC and IAC provide for passive, reliable, and safe air cooling of the reactor vessel.

Forced flow heat removal via IAC serves as the normal shutdown cooling system for outages. There are two trains, one for each IHT system loop. IAC leverages the existing intermediate heat exchanger heat transfer area rather than adding a shutdown cooling-only heat exchanger inside the reactor vessel. For the final heat sink, it rejects heat to a sodium-air heat exchanger leveraging the IHT system piping.

Natural circulation flow through both trains of IAC, actuated by passive damper opening, serves the heat removal function if power is not available to run the blowers feeding the sodium-air heat exchangers. The passive nature of this mode assures high reliability of the function.

RAC can remove all decay heat using natural circulation of air around the exterior of the guard vessel. Heat removal from the reactor vessel is dominated by radiative heat transfer with a small contribution by convection through argon in the gap between the reactor and guard vessels. The guard vessel, in turn, is cooled by the natural convection of air and by radiative heat transfer to the outside air duct wall, referred to as the collector cylinder. Hot air then rises, which establishes a natural circulation path using the atmosphere as a heat sink. In contrast with the IAC, RAC does not have any dampers to turn the system on or off. RAC is always operating and there are no actions required for the system to perform its function.

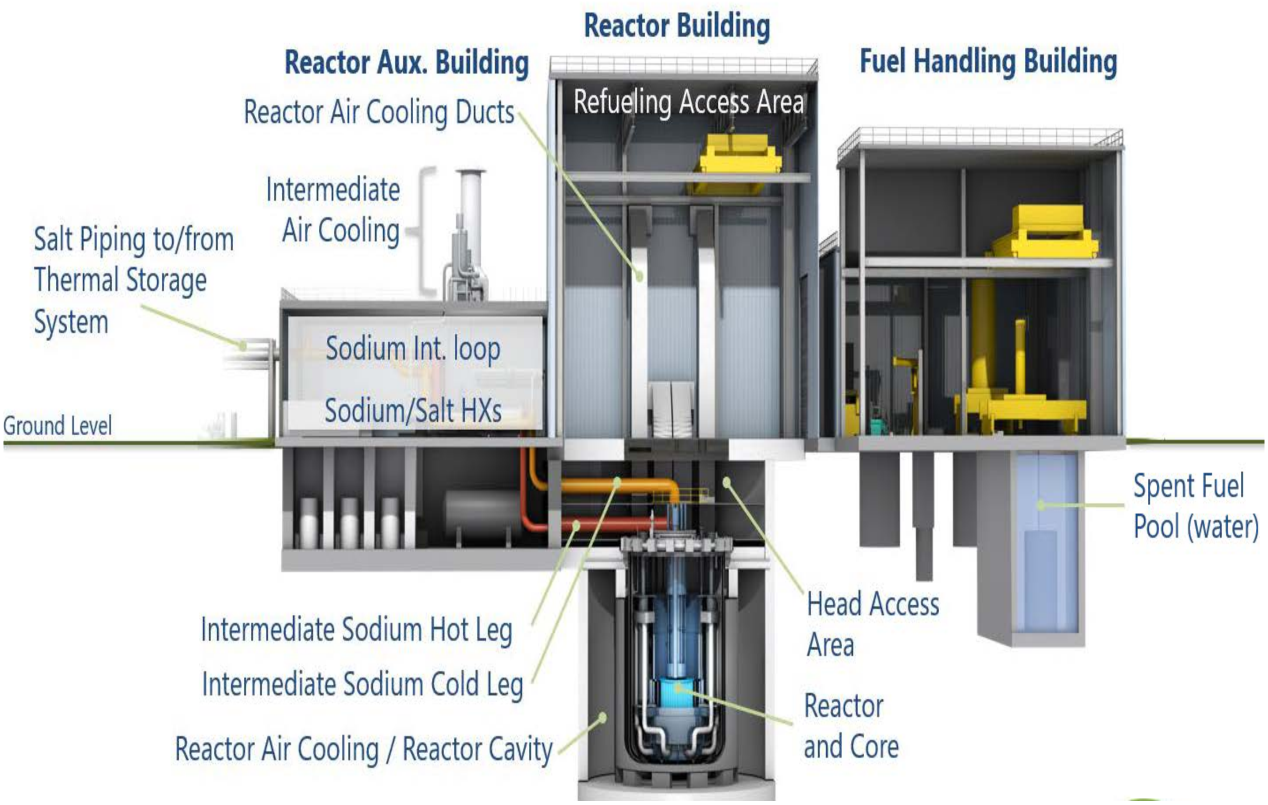
4.3 Contain - Low Reactor Pressure

The high boiling point of sodium allows reactor operation at near atmospheric pressure resulting in thinner reactor structures. The Natrium design is a low-pressure pool type reactor with no piping or fittings below the surface of the pool. A close-fitting guard vessel prevents a loss of coolant should a leak develop in the reactor vessel. The reactor cover gas system also operates at essentially atmospheric pressure. The intermediate coolant is, by static head alone, at a slightly higher pressure than the primary coolant.

Due to the absence of pressure driving forces in accidents and by utilizing a combination of different systems and components of the Natrium design a functional containment approach is utilized to meet the onsite and offsite radionuclide release limits for the event categories.

5 NATRIUM PLANT DESIGN

Figure 2: Natrium Nuclear Island



5.1 NI Systems

Reactor Core Components System (RCC)

The Natrium reactor is a sodium fast reactor that provides 840 MW thermal of heat generation. The coolant flows upward through the core which is composed of fuel, control, reflector, and shield assemblies. The fuel assembly produces heat and provides the neutron flux environment. Natrium fuel features a U-10Zr fuel column with a sodium bond to HT9 cladding. The RCC contains control assemblies that function to position neutron absorber material to control and terminate the nuclear reaction. These are positioned by the control rod drive mechanism system discussed further below. The reflector assemblies surround the active fuel assemblies radially, improving neutron efficiency and limiting radiation damage to permanent reactor structures. A single row of shield assemblies makes up the outermost row of the reactor core, directly adjacent to the reflector assemblies. These function to absorb neutron leakage outside the reflector assemblies, limiting activation of the intermediate sodium system while also contributing to prevent radiation damage to permanent reactor structures.

Reactor Enclosure System (RES)

The RES contains and supports the reactor core, the primary sodium coolant, and all supporting equipment and structures. The RES is divided into five subsystems: Reactor Vessel (RV), Reactor Internals (RI), Reactor Vessel Head (RVH), Guard Vessel (GV), and Reactor Support Assemblies (RSA). All subsystems are located in, and are either directly or indirectly supported by, the Reactor Building (RXB). The RV, along with the RVH, form the majority of the primary coolant and primary cover gas boundaries. Additionally, the RVH locates and supports additional systems and equipment that interfaces with the core and primary coolant. Finally, the RVH and RV provides support for the RI as well as the Core Support Structure (CSS).

The CSS is welded to and considered part of the RV. The RI is a collection of internal structures which perform various passive functions in reactor operation. The major components of the Reactor Internals include the Vessel Liner, Primary Sodium Pump (PSP) and Intermediate Heat Exchanger (IHX) supporting structures, primary coolant supply piping, and the Upper Internal Structure (UIS), which houses instrumentation conduits and locates the Control Rod Drivelines. The RVH supports and seals the RV. The RVH provides structural support and interfaces for a number of systems, including the IHX and PSP components of the Primary Heat Transport System (PHT), and elements of the Fuel Handling System (In-Vessel and Ex-Vessel), Reactor Instrumentation System (RIS), and Control Rod Drive Mechanism System. The RVH also houses and supports a number of other subsystems and components, such as the Rotating Plug, Head Thermal Shields, Fuel Transfer Ports and Access Ports, and Reactor Head Temperature Control. Of particular interest, the RVH supports and locates the rotating plug for refueling operation. This plug is essential for the initial fueling of the reactor and for all subsequent fuel transfer operations during refueling and decommissioning. The plug is configured such that the In-Vessel Transfer Machine can access all core components, the In-Vessel Storage (IVS) locations, and the fuel elevator. The plug rotates via a bearing and drive assembly and is equipped with sealing mechanisms to isolate the primary fluid and cover gas from atmosphere during normal, accident, and refueling operations. The GV is supported either by the RXB or the RVH, or a combination of both. The GV surrounds the RV and is designed to contain any sodium leak in the unlikely event of a breach in the RV, ensuring sufficient coolant inventory is maintained in the RV for residual heat removal through level equalization and preventing a sodium reaction with the surrounding reactor building concrete. The GV is surrounded by a guided pathway for the RAC system, which is always in operation and functions to remove decay heat in the case of an emergency.

Primary Heat Transport System (PHT)

The PHT is entirely contained within the RV and consists of the reactor core, the Intermediate Heat Exchangers (IHX), the PSPs, the hot pool, and the cold pool. The PHT sodium flows up through the core where the fuel assemblies heat the sodium. The hot sodium enters the hot pool and flows downward through the shell side of the two IHXs. The sodium, cooled by the IHT sodium coolant, exits the bottom of the IHXs and enters the cold pool. Cold pool sodium flows downward to the PSP inlet plenums which are located very near the bottom of the vessel to maximize coolant inertia. PSP pumps drive the cold pool sodium downward from the inlet and discharge it into a series of core supply pipes, which return the sodium to the core inlet. The sodium then enters the core through the core support and distribution structure completing this flow circuit.

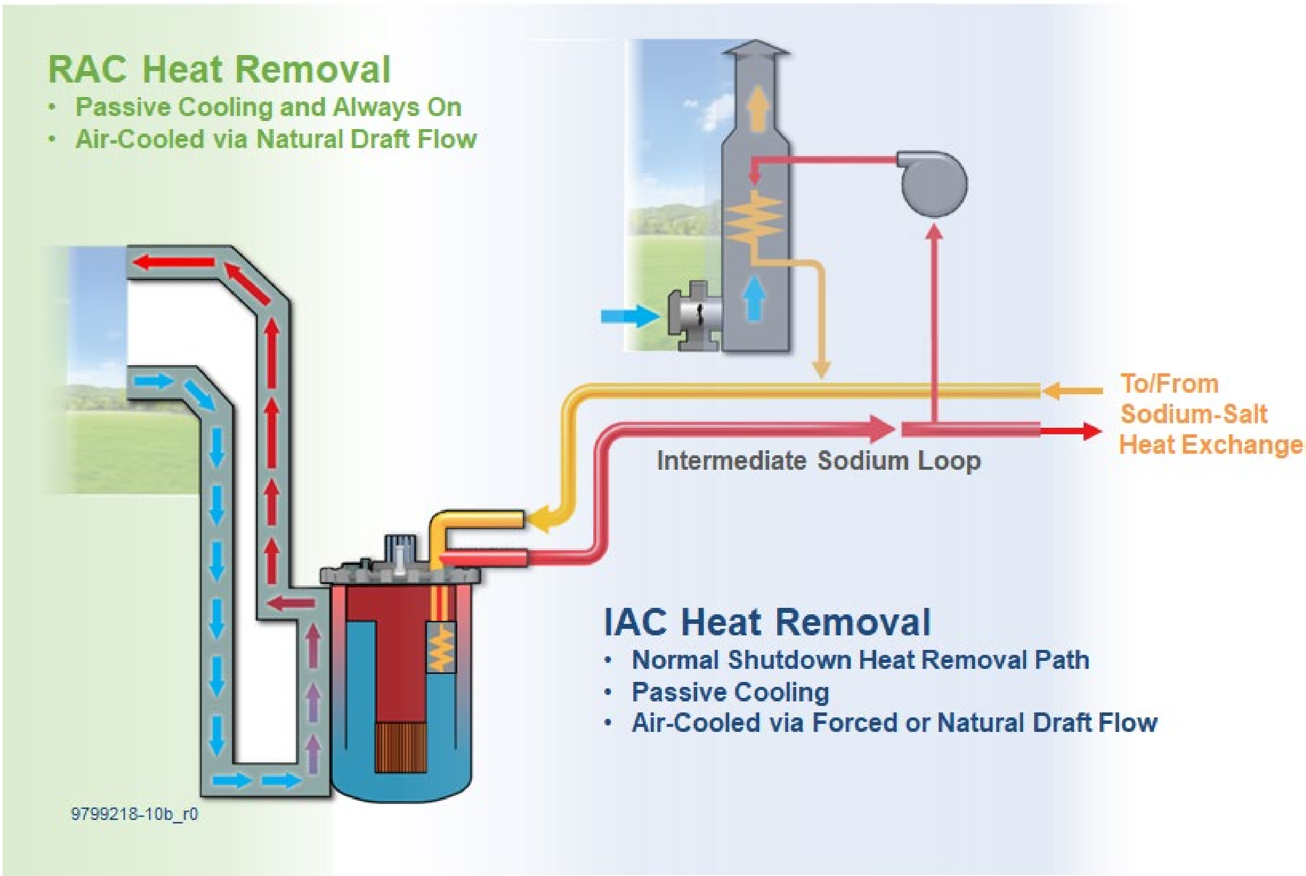
Intermediate Heat Transport System (IHT)

The IHT transfers heat from the PHT to the Thermal Salt Storage System (TSS). The IHT performs this function during normal power operation and transient conditions. There are two IHT piping loops for each reactor module. Each intermediate loop is thermally coupled to the reactor PHT by an IHX. IHT non-radioactive sodium is circulated via the Intermediate Sodium Pumps (ISPs) which transport heat from the IHXs to the Sodium-Salt Heat Exchangers (SHXs). The IHT pumps are located in the cold leg to reduce their operating temperature. The main components of the IHT loops are the ISPs, SHXs, intermediate sodium hot and cold leg piping, expansion tanks, and the sodium drain tank. The salt system pumps cold salt from the cold salt tank to NI above ground level as a single supply line. The salt stream splits into four parallel pipes within the NI to provide individual streams to each of the four SHXs. Cold salt enters each SHX unit where it is heated via the IHT sodium flow. Once heated in the SHX, the hot salt stream exits the SHX, joins into a single stream with the other SHX salt outlet streams, and flows out to the EI hot salt storage tank. Salt flow is controlled by the variable speed cold salt pumps and a temperature control system. A set of drain isolation valves on each stream ensures isolation of the salt systems and rapid drain of the SHXs during selected transient events.

Reactor Air Cooling System (RAC)

The RAC is the residual heat removal cooling system for the reactor and protects the fission product boundaries from the most severe spectrum of plant events. RAC supplies natural draft outside ambient air for reactor cooling. RAC relies on the natural circulation performance of the primary sodium and the conductive/convective heat transfer to the reactor vessel wall. Thermal radiative heat transfer then dominates heat transfer to the guard vessel. From there, natural draft air inlets provide ambient outside air to cool the guard vessel wall via a combination of radiative and convective heat transfer. RAC is an open passive design that is always in operation and does not require equipment alignment, power, operator action, or support systems to perform at peak performance.

Figure 3: Reactor Air Cooling and Intermediate Air Cooling



Intermediate Air Cooling (IAC)

IAC serves as the normal shutdown cooling system for outages. It has two cooling modes: forced flow and passive flow. Each IHT loop is composed of two trains. For the final heat sink, it transfers heat to the atmosphere from the Sodium-Air Heat Exchangers (AHXs). Simple operation of a fail-open electromagnetic damper on the air side of the AHX initiates passive cooling. Active operations support normal controlled cooling operations (such as during a refueling outage) and in response to anticipated transient events. Forced flow is provided by air blowers on the air side of the AHXs and the ISPs on the sodium side. The IAC's natural draft arrangement, similar to RAC, permits passive operation of the system as a diverse alternative to the RAC cooling capabilities if power to support forced cooling is not available. These functions supplement the RAC system.

Control Rod Drive Mechanism System (CRD)

The CRD controls reactor power by positioning the neutron absorber bundle within the core. This system also allows for rapid control rod insertion in response to manual or automatic signals (scram). When a scram is initiated, the CRD shuts down the reactor. Control rods are normally positioned by an electric motor. However, when a scram signal is received, the latch releases the absorber bundle from the driveline. The absorber bundle then falls by gravity into the core which shuts down the reactor. As the alternate and preferred emergency shutdown method, each drive mechanism has a drive-in motor to rapidly drive the control rod into the core shutting down the reactor. In the unlikely event that the control rod does not fully insert, the drive-in motor pushes the control rod into the core.

Reactor Protection System (RPS)

The RPS is designed to ensure the safety of the reactor during plant operations. RPS monitors the plant conditions and initiates a sequence of appropriate functions as necessary to protect the public, plant workers, plant equipment, and the environment when an abnormal operating condition is detected. The RPS has multiple function-identical redundant divisions included in the configuration. Each division is located in a dedicated room physically separated from other divisions such that potential hazards within the design basis such as fire or flood would not damage multiple divisions simultaneously.

5.2 EI Systems

Figure 4: Energy Island Photo

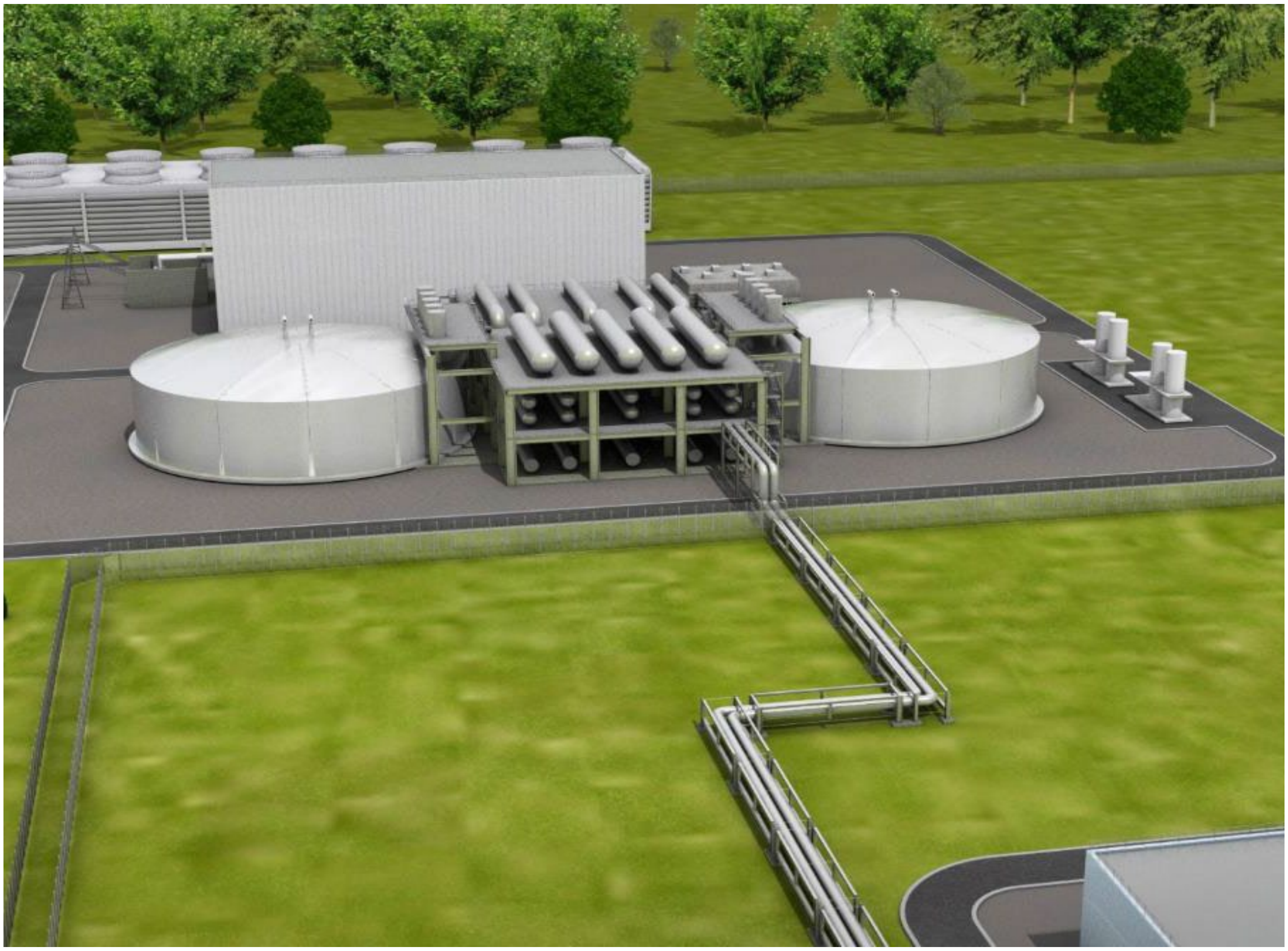
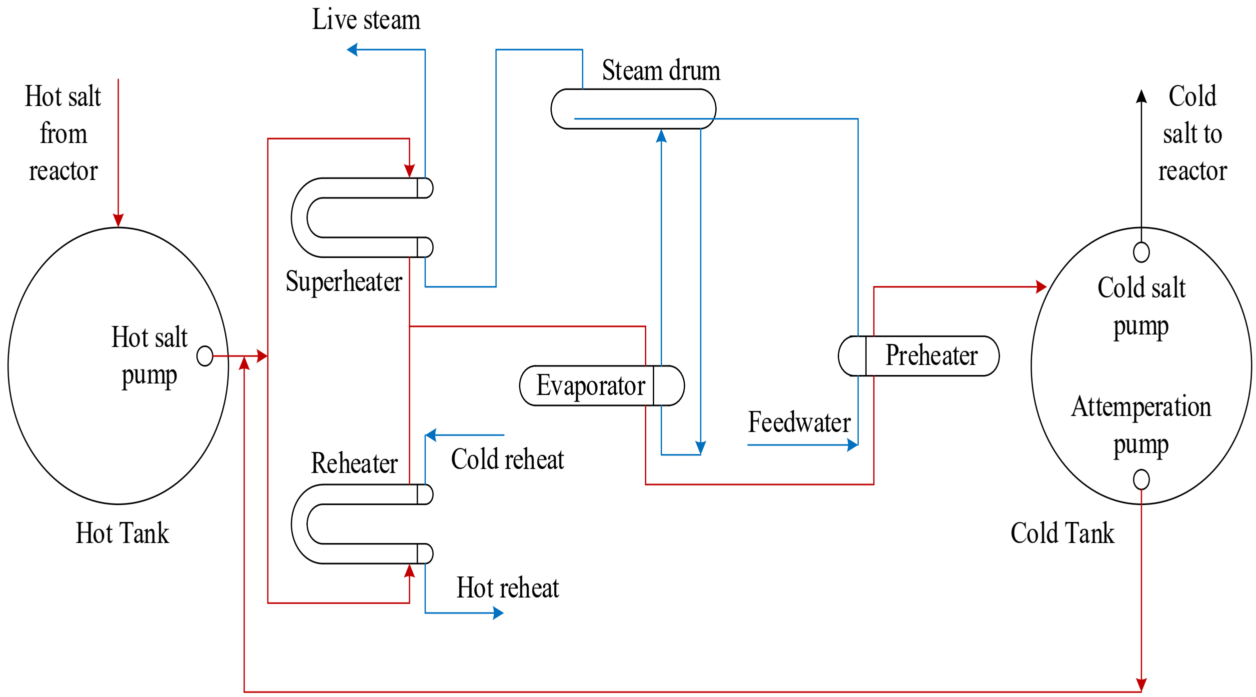


Figure 5: Energy Island Diagram



Thermal Energy Storage System (TSS)

Cold salt is transported from the cold salt tank, via the cold salt tank pump, to the NI intermediate sodium/salt heat exchanger, where it is heated, and returns to the hot salt tank. The sodium to salt heat exchangers are designed such that the thermal salt side pressure (shell side) is higher than the intermediate sodium pressure (tube side), to ensure that if a leak were to occur in the intermediate sodium system inside the sodium to salt heat exchanger the thermal salt system would flow into the IHT system. The system is composed of salt pumps and salt storage tanks, interfacing with salt heat transport loops to send and receive salt. The salt tanks are used for energy storage and the reactor is the charging system. Its architecture is essentially the same as the molten salt systems designed for concentrated solar power except the heat source is the Natrium reactor.

Salt from the hot salt tank is transported through the Steam Generation System (SGS) and returns to the cold tank. The energy storage capacity for the tank pair is equivalent to approximately 4 hours at 500-MWe net load. Energy storage maximizes the NI capacity factor while the EI can ramp power up and down based on grid conditions. An attemperation pump delivers cold salt to the SGS to control the hot salt temperature at the steam generator inlet during start-up.

Steam Generation System (SGS)

The SGS generates high-pressure, superheated steam for conversion to electrical power by the Turbine/Generator. The SGS converts water into steam by passing the hot molten salt through an economizer (water preheater), evaporator, superheater, and reheater and provides that steam to the Turbine/Generator. Use of hot molten salt in the steam generator re-heaters improves overall plant thermodynamic efficiency. The relatively colder salt returning from the SGS is stored in the cold salt tank prior to flowing back to the NI for re-heat, thus completing the heat transport circuit.

Condensate and Feedwater System

For power generation, the steam supplied to the turbine condenses and collects in a condenser. The condensate system pumps this sub-cooled water through several low-pressure feedwater heaters. Heat for these heaters is supplied by extraction steam from the turbine. After passing through the feedwater heaters, the water collects in the deaerator. The deaerator acts as the water supply for the feedwater system. The feedwater pumps supply high pressure water to the SGS system.

Steam Turbine and Generator Systems

The steam turbine generator is a commercially available modular platform General Electric STF-D453 or similar steam turbine. The system includes the main steam, reheat steam, and turbine bypass steam sub-systems. The main steam supplies the High-Pressure turbine and the reheat steam supplies the Intermediate- Pressure and Low-Pressure turbines. The bypass steam allows the steam to bypass the turbine in the event of a turbine trip which maintains the primary heat sink for the salt tanks. Operating in thermal equilibrium with the reactor, the main turbine generator can produce a net output of 336 MWe consistently. Using the additional stored energy capacity in the hot salt tank the main turbine generator can increase its electrical output to 500 MWe net without the NI adjusting its thermal output.

Heat Rejection Systems

The condenser/heat rejection system utilizes circulating water to condense and sub-cool the steam exiting the main turbine generator steam turbine. Circulating water pumps take water from the mechanical draft cooling tower basin passes it through the condenser and returns it to the cooling tower. The heat is removed from the tower by the water passing through a forced draft air flow.

6 PROCESS FOR SSC CLASSIFICATION

TerraPower developed a set of instructions for classifying SSCs of nuclear reactor facilities in accordance with their function, design & licensing basis, and safety significance. These classifications determine the codes and quality standards to which the SSCs shall be designed, fabricated, erected, maintained, and tested. The procedure is designed to be used under a licensing process that follows 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” which endorses, while providing clarification and points of emphasis, the methodology described in NEI 18-04, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development”. This process was used for the selection of LBEs; safety classification of SSCs and associated risk-informed special treatments; and determination of DID adequacy for the Natrium reactor.

The SSC classification process flowchart shown below, as Figure 6, was used for the classifications of the SSCs in the Natrium reactor.

Figure 6: SSC Classification Process (Figure 4-1, SSC Function Safety Classification Process from NEI 18-04)

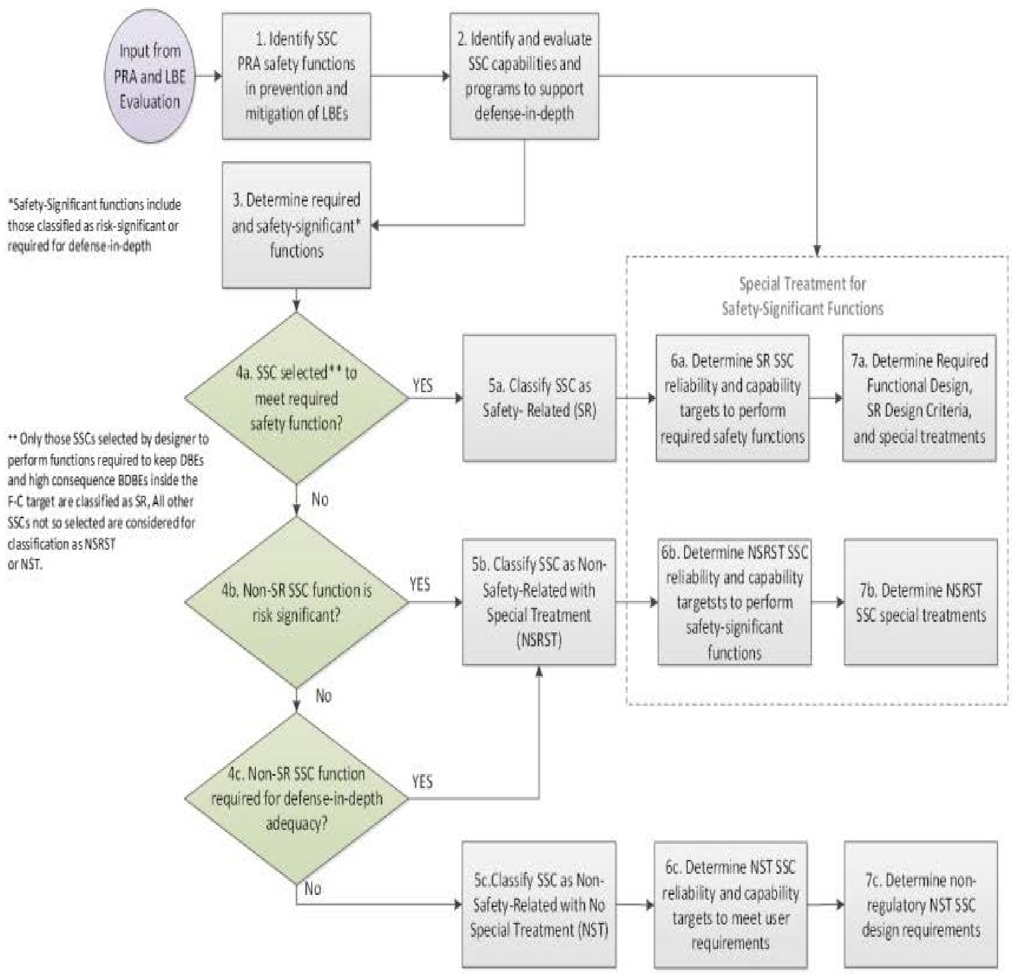


Figure 4-1. SSC Function Safety Classification Process

Using the above process, there are no SSCs on the EI necessary to meet the fundamental safety functions of the NI for controlling reactivity, cooling the core, or containing the release of radioactivity. There are no SSCs on the EI that provide a function classified as risk-significant. Additionally, no SSCs on the EI are used to provide a required DID function for mitigation. Based on this, all SSCs located on the EI are classified as non-safety-related with no special treatment (NST). This analysis shows that none of the SSCs on the EI are utilized for any accident mitigation, safety system support, or DID, showing a separation from the NI for accident mitigation and the safety bases for the plant.

7 BASIC PLANT TRANSIENT ANALYSES

The Natrium reactor ensures that no EI SSCs are required to respond to mitigate any events impacting the NI, support SR SSCs, or ensure DID adequacy. This establishes a separation between the NI and EI from a safety perspective. The isolation point for the thermal salt storage system (EI to NI isolation) is at the inlet valve (input from the cold salt storage tank) and outlet valve (output to the hot salt storage tank) of the sodium to salt heat exchangers. The valves are located as close as possible to the exterior reactor auxiliary building wall (located on the NI) in the thermal salt system piping.

To fully appreciate the separation and interfaces between the NI and EI, it is important to understand the Natrium reactor response to the failure of any EI SSCs. From the NI perspective, all failures associated with the EI are grouped into loss of salt flow, high salt temperature, increased salt flow, low salt temperature and low salt pressure initiating events. The transient analyses from the EI groups these five events into two categories that are modeled either as a decrease in heat removal (loss of salt flow, high salt temp, low salt pressure) or an increase in heat removal (increase in salt flow, low salt temp).

For each of these event categories, the initial effects are seen in the IHT system and do not directly impact the conditions in the PHT system. A decrease in heat removal results in an increase in temperature in the IHT system while an increase in heat removal results in a decrease in IHT sodium temperature. The thermal inertia of the IHT and PHT systems are such that any changes in salt conditions can be adequately responded to using only signals monitored within the NI. Heat removal is ensured by the RAC and IAC systems of the NI.

Parameters in the salt system are continuously monitored such as salt flow, salt pressure, and salt tank levels. The architecture of the plant allows the use of different signals, including IHT temperature signals, to actuate an anticipatory power runback to provide asset protection, reduce thermal stresses on the NI systems, and improve plant availability without compromising plant safety.

7.1 Plant Transient Response

A. Reactor scram:

- RPS receives an input that exceeds a scram setpoint and initiates a scram signal.
- IAC Forced Circulation initiates coincident with the scram signal.
- Once reactor power has lowered to below the low neutron setpoint, ramp down of PSPs and ISPs commences.
- Once the ramp down sequence is complete, the ISPs and PSPs are controlled to establish PHT and IHT flow.
- After initiation of the scram signal and once reactor power has lowered to the low neutron setpoint, the cold salt flow ramps down, followed by isolation of the SHXs.

- As cold salt flow is reduced, cooling is transferred to IAC by adjusting the IAC damper position and fan speed to maintain cold leg IHT temperature to hot standby parameters.
- The plant is now in a stable hot standby condition.

B. Anticipatory reactor runback:

- Reactor power is decreased at a pre-determined rate by inserting control rods.
- PSP, ISP, and salt pump flows are decreased to the targeted low flow settings.
- The IAC/AHX system controls IHT temperature at a specified temperature.
- Once the IHT temperature is within the controlled setpoint the salt flows are reduced at a predetermined rate.
- The SHXs isolate and the cold salt pumps shut down.
- The reactor is now controlled at low power.

8 REGULATORY ANALYSES

TerraPower evaluated the impact of the Natrium design interfaces between the NI and EI on the following regulations:

- 10 CFR 50.10 - License required; limited work authorization
- 10 CFR 55 – Operators' Licenses
- 10 CFR 50.65 – Requirements for monitoring the effectiveness of maintenance at nuclear power plants
- 10 CFR 50 Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

8.1 Regulatory analyses associated with 10 CFR 50.10, "License required; limited work authorization."

The purpose of 10 CFR 50.10 is to ensure that activities constituting construction of a production or utilization facility on a site on which the facility is to be operated do not begin until a license or authorization is issued by the NRC. This ensures there is a safety analysis report that demonstrates the activities conducted under the limited work authorization (LWA) are conducted in compliance with the technically-relevant NRC requirements in 10 CFR Chapter 1 that are applicable to the design of those portions of the facility within the scope of the LWA and that the NRC staff issues a final environmental impact statement for the LWA in accordance with subpart A of Part 51 of Chapter 1. These requirements ensure that an NRC authorization is obtained before undertaking activities that have a reasonable nexus to radiological health and safety and/or common defense and security.

Activities constituting construction are described in 10 CFR 50.10(a)(1):

> 50.10 License required; limited work authorization.
>
> (a) Definitions. As used in this section, construction means the activities in paragraph (a)(1) of this section, and does not mean the activities in paragraph (a)(2) of this section.

> (1) Activities constituting construction are the driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of foundations, or in-place assembly, erection, fabrication, or testing, which are for:
>
> (i) Safety-related structures, systems, or components (SSCs) of a facility, as defined in 10 CFR 50.2;
>
> (ii) SSCs relied upon to mitigate accidents or transients or used in plant emergency operating procedures;
>
> (iii) SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function;
>
> (iv) SSCs whose failure could cause a reactor scram or actuation of a safety-related system;
>
> (v) SSCs necessary to comply with 10 CFR part 73;
>
> (vi) SSCs necessary to comply with 10 CFR 50.48 and criterion 3 of 10 CFR part 50, appendix A; and
>
> (vii) Onsite emergency facilities, that is, technical support and operations support centers, necessary to comply with 10 CFR 50.47 and 10 CFR part 50, appendix E.

The Natrium reactor safety classification of SSCs is being performed utilizing the LMP as defined in NEI 18-04, Rev. 1. The output of the SSC classification process results in all SSCs located on the EI being classified as non-safety-related with no special treatment (NST). The Natrium design does not rely on any NST SSCs to provide mitigation for an accident or transient. The Natrium design does not have any NST SSCs whose failure could prevent SR SSCs from fulfilling their safety-related function. Based on this assessment, criteria i, ii, and iii of 10 CFR 50.10(a)(1) do not apply to SSCs on the EI.

The language in 10 CFR 50.10(a)(1)(iv) is consistent with the language currently in 10 CFR 50.65(b)(2)(iii). The language alignment of 10 CFR 50.10(a)(1)(ii, iii, iv) to that of 10 CFR 50.65(b)(2) (i, ii, iii) was further clarified in RG 1.206, Rev. 1, “Applications for Nuclear Power Plants.” RG 1.206 states that, in the LWA rule, the scope of SSCs falling within the definition of construction was derived from the scope of SSCs that are included in the program for monitoring the effectiveness of maintenance at nuclear power plants, as defined in 10 CFR 50.65(b). As discussed in the supplementary information, the NRC selected the criteria used in the definition of construction to take advantage of work done during development and implementation of the maintenance rule (10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”). Like the LWA rule, the maintenance rule defines a scope of SSCs that have some nexus to radiological health and safety (i.e., safety significance). The NRC selected the maintenance rule criteria for use in the definition of construction, in part, because the criteria are well understood and there is good agreement on their implementation. In addition, the NRC prepared guidance that has been used extensively in the industry for implementing the maintenance rule in RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” issued in March 1997. RG 1.160 endorses industry guidance provided in

NUMARC 93-01, Rev. 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," issued in April 1996. For these reasons, the NRC decided that the maintenance rule guidance can also be applied to determinations of SSCs that are within the scope of the definition of construction. The NRC also recognizes that determinations of which SSCs fall within the definition of construction depends on the design of the facility.

TerraPower plans to seek an exemption from 10 CFR 50.10(a)(1)(iv) and 10 CFR 50.65(b)(2)(iii) for the construction and operation of the EI of the Natrium reactor. The rationale for seeking an exemption from 10 CFR 50.10(a)(1)(iv) is discussed in Section 8.3 with the discussion on 10 CFR 50.65(b)(2)(iii).

Construction activities for SSCs necessary to comply with 10 CFR 73 include the preparation and building of physical barriers and structures and associated hardware and detection systems for the physical security program. None of these physical security program SSCs are located on the EI for the Natrium reactor. To meet the requirements of the cyber security programs included in 10 CFR 73.55, digital components and control systems identified as Critical Digital Assets (CDAs) are not installed or activated prior to receiving a construction permit. Based on this assessment, criterion v of 10 CFR 50.10(a)(1) does not apply to SSCs on the EI.

Construction activities for SSCs necessary to comply with 10 CFR 50.48 and Advanced Reactor Design Criterion 3 in Regulatory Guide 1.232, Rev. 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," include fire protection equipment and components to achieve and maintain safe shutdown. This includes all the fire protection equipment for protection of the equipment necessary to achieve and maintain safe shutdown, including the fire barriers in walls for penetrations to minimize spread and maintain separation. Potential fires on the EI would not prevent the ability to achieve and maintain safe shutdown of the reactor. Based on this assessment, criterion vi of 10 CFR 50.10(a)(1) does not apply to SSCs on the EI.

Onsite emergency facilities (i.e., technical support and operations support centers) necessary to comply with 10 CFR 50.47 and 10 CFR 50, appendix E are not located on the EI. The facility for providing onsite emergency first aid and decontamination are not located on the EI. Based on this assessment, criterion vii of 10 CFR 50.10(a)(1) does not apply to SSCs on the EI.

Based on these analyses and contingent on approval of the planned exemption request for criterion iv, construction of the EI would not constitute construction in accordance with 10 CFR 50.10. Therefore, an LWA would not be needed for EI construction activities.

8.2 Regulatory analyses associated with 10 CFR 55, "Operators' Licenses".

10 CFR 55 provides the requirements, scope and regulations associated with operators' licenses. In accordance with 10 CFR 55, operators' licenses are for activities constituting the operation of controls. Controls are manipulations which directly affect the reactivity or power level of the reactor of utilization facilities licensed under the Atomic Energy Act of 1954, as amended, or Section 202 of the Energy Reorganization Act of 1974, as amended, and part 50.

The following are excerpts from 10 CFR 55:

> 55.1 Purpose states that regulations in this part:
>
> (a) Establish procedures and criteria for the issuance of licenses to operators and senior operators of utilization facilities licensed under the Atomic Energy Act of 1954, as amended, or Section 202 of the Energy Reorganization Act of 1974, as amended, and part 50, part 52, or part 54 of this chapter.
>
> 55.2 Scope states that the regulations in this part apply to:
>
> (a) Any individual who manipulates the controls of any utilization facility licensed under parts 50, 52, or 54 of this chapter,
>
> (b) Any individual designated by a facility licensee to be responsible for directing any licensed activity of a licensed operator.
>
> (c) Any facility license.
>
> 55.3 License Requirements states that a person must be authorized by a license issued by the Commission to perform the function of an Operator, or a Senior Operator as defined in this part.
>
> 55.4 Definitions states the following definitions:
>
> *Controls* when used with respect to a nuclear reactor means apparatus and mechanisms the manipulation of which directly affects the reactivity or power level of the reactor.
>
> *Operator* means any individual licensed under this part to manipulate a control of a facility.
>
> *Senior operator* means any individual licensed under this part to manipulate the controls of a facility and to direct the licensed activities of licensed operators.

The Natrium reactor removes direct interaction between the nuclear reactor and the main turbine generator. Due to the lack of direct interaction, operation of the main turbine generator is not an apparatus or mechanism whose manipulation directly affects the reactivity or power level of the reactor. Based on this design feature, operation of the main turbine generator is not considered a control by the definition in 10 CFR 55.4, which states, "with respect to a nuclear reactor means apparatus and mechanisms the manipulation of which directly affects the reactivity or power level of the reactor." The scope of 10 CFR 55.2 states that the regulations in this part apply to "any individual who manipulates the controls of any utilization facility licensed under part 50." Since main turbine generator operation for the Natrium design is not included in the definition of controls in 10 CFR 55.4, the regulations associated with 10 CFR 55 do not apply to operation of the main turbine generator of the Natrium plant. Because of this, the Natrium design allows for a non-licensed individual to fully operate and control the main turbine generator. Utilizing these principles allows the grid operator to control the main turbine generator as another asset for electrical grid management similar to conventional power sources.

8.3 Regulatory analyses associated with 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants”.

The purpose of 10 CFR 50.65, commonly referred to as the Maintenance Rule, is to monitor the effectiveness of maintenance at nuclear power plants. Holders of an operating license must monitor the overall performance or condition of SSCs and demonstrate that the SSCs are being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended functions.

In accordance with 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants”, the scope of the program shall monitor the performance or condition of SSCs, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these SSCs, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience.

As stated in Table 4-1 of NEI 18-04 Revision 1, it is a basic requirement for all safety-significant SSCs that a Maintenance Program assures targets for SSC availability and effectiveness of maintenance to meet SSC reliability targets. This is essentially the same as 10 CFR 50.65 Maintenance Rule and is consistent with 10 CFR 50.69 for RISC-1 (SR) and RISC-2 (NSRST) SSCs. In addition, RG 1.233 states that “the framework incorporates an integrated review approach by using the satisfaction of selected requirements to provide reasonable assurance of some aspects of SSC performance (for example, performance-based acceptance criteria related to SSC capability, reliability, and availability). Examples of requirements that could be applied for this purpose include 10 CFR Part 50, Appendix A (general design criteria, overall requirements, criteria 1 through 5), 10 CFR Part 50, Appendix B (quality assurance program), 10 CFR 50.49 (electric equipment environmental qualification program), 10 CFR 50.55a (code design, Inservice testing and Inservice inspection programs), 10 CFR 50.65 (maintenance rule)[...]." Based on this statement, the process can be used to inform the SSCs included in the maintenance rule.

The following are excerpts from 10 CFR 50.65:

> 50.65, “Requirements For Monitoring The Effectiveness Of Maintenance At Nuclear Power Plants”:
>
> (b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:
>
> (2) Nonsafety related structures, systems, or components:
>
> (i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or
>
> (ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

(iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

The Natrium reactor safety classification of SSCs is being performed utilizing the LMP. The output of the process results in all SSCs located on the EI being classified as NST. The Natrium reactor does not rely on any NST SSC to provide any mitigation for any accident or transient or that are used in EOPs. The Natrium reactor does not have any NST SSCs whose failure could prevent any SR SSC from fulfilling their safety-related function. Based on this assessment, criteria i and ii of 10 CFR 50.65(b)(2) do not apply to SSCs located on the EI.

RG 1.160, “Monitoring the Effectiveness Of Maintenance At Nuclear Power Plants,” Revision 4, and NUMARC 93-01, “Industry Guideline For Monitoring The Effectiveness Of Maintenance At Nuclear Power Plants,” Revision 4 were evaluated for potential additional clarification for criterion iii, especially on the implementation of the maintenance rule program.

NUMARC 93-01 provides additional clarification by stating licensees should consider the following SSCs to be within the scope of the rule for criterion (iii):

1. SSCs whose failure has caused a reactor scram or actuation of a safety-related system at their site.
2. SSCs whose failure has caused a reactor scram or actuation of a safety-related system at a site with a similar configuration.
3. SSCs identified in the licensee’s analysis (e.g., FSAR, IPE) whose failure would cause a reactor scram or actuation of a safety-related system.

A licensee may exclude SSCs that meet criteria 2 or 3 if they have demonstrated by analysis (e.g., FSAR, IPE) and by operational experience that the design or configuration of an SSC is fault-tolerant through redundancy or installed standby spares such that a reactor scram or actuation of a SR system is implausible.

The additional clarifications provided in 1 and 2 are not applicable to the Natrium design, since there is no previous operating experience and the configuration for the Natrium design is unique based on the thermal salt storage system. The clarification provided in 3 is applicable, since analysis shows that failures in the thermal storage salt system would lead to a runback, which is not an SR actuation. If the non-safety runback did not happen, the plant would eventually reach a scram set point. Since clarification 3 applies, 10CFR50.65(b)(2)(iii) also applies to the thermal salt storage system.

The NRC proposed Part 53 rulemaking language for a risk-informed, technology-inclusive regulatory framework for advanced reactors, specifically the NRC’s proposed language in Section 53.715, “Maintenance, repair, and inspection programs” (ADAMS Accession No. ML22165A265, dated June 2022), states:

a) A program to control maintenance activities and monitor the performance or condition of SR and NSRSS SSCs must be developed, implemented, and maintained to provide reasonable assurance that the safety criteria defined in Sections 53.210 and 53.220 of this part will be met.

Section 53.715 proposed language is consistent with the safety classification of SSCs described in NEI 18-04. Please note that TerraPower is using the SSC classifications as defined in NEI 18-04 for safety-related (SR), non-safety-related with special treatment (NSRST), and non-safety-related with no special treatment (NST). These categories correspond to those in Part 53: safety related (SR), non-safety related but safety significant (NSRSS), and non-safety related (NSR). In utilizing the new NRC proposed draft process to address the current language associated with 50.65(b)(2)(iii), no SSCs on the EI would be part of the 53.715 rule as they would not be classified as SR or NSRSS.

Based on this, TerraPower plans to seek an exemption from 10 CFR 50.65(b)(2)(iii) by utilizing the NRC's proposed draft 10 CFR 53.715 language. This exemption would allow a Natrium plant's licensing basis to be consistent with the safety classification of SSCs described in NEI 18-04, in that only SR and NSRST SSCs are part of the maintenance rule scope. The output of the SSC classification process results in all SSCs located on the EI being classified as NST.

Based on these analyses and contingent on approval of the planned exemption request for criterion iii, EI SSCs would not be in the scope of the maintenance rule in accordance with 10 CFR 50.65(b)(2).

8.4 Regulatory analyses associated with 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plant".

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, manufacture, construction, and operation of structures, systems, and components and states, "[t]he pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying."

The Natrium reactor safety classification of SSCs is being performed utilizing the LMP as defined in NEI 18-04. The output of the SSC classification process results in all SSCs located on the EI being classified as NST. The Natrium reactor does not rely on any NST SSCs to provide any mitigation for any accident or transient. The Natrium design does not have any NST SSCs whose failure could prevent any SR SSCs from fulfilling their safety-related function. Since the SSCs located on the EI are NST and do not affect the SR functions of the SSCs used for mitigation, the requirements associated with 10 CFR 50 Appendix B do not apply to SSCs which are located on the EI.

9 CONCLUSION

The NI boundary conditions have been intentionally designed so the interface with the EI does not impact the Natrium reactor's safety case. Similarly, steam generation and thermal energy storage operations (e.g., ramp rates) are independent from reactor power operations due to the presence of the molten salt energy storage tanks. Several regulations were analyzed with these design principles in mind including: 10 CFR 50.10 - License Required;

Limited Work Authorization; 10 CFR 55 – Operators' Licenses; 10 CFR 50.65 – Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants; and 10 CFR 50 Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

TerraPower requests NRC review and approval of the evaluations provided in this TR, as they pertain to the four regulatory requirements discussed above.

Exemption requests from portions of the requirements of 10 CFR 50.10(a)(1) and 10 CFR 50.65(b) for SSC's not classified as SR or NSRST in accordance with NEI 18-04 are planned to be submitted for the NRC's review. If the exemptions are granted for these two criteria, discussed in Sections 8.1 and 8.3, the four regulations discussed in Section 8 of this Topical Report would not apply to any SSCs located on the EI.

10 REFERENCES

1) 10 CFR 50

2) 10 CFR 73

3) NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development"

4) NRC proposed Part 53 rulemaking

5) NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

6) Regulatory Guide 1.160, Revision 4, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

7) Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants"

8) Regulatory Guide 1.232, Revision 0, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactor"

9) Regulatory Guide 1.233, Revision 0, "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"

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