5 SAFETY FUNCTIONS, DESIGN CRITERIA, AND SSC SAFETY CLASSIFICATION

This chapter of the SE describes the U.S. Nuclear Regulatory Commission staff's (the staff's) review and evaluation of Kemmerer Unit 1 (KU1) PSAR chapter 5, which describes the safety classification of structures, systems, and components (SSCs), the safety-significant probabilistic risk assessment (PRA) safety functions (PSFs), and the principal design criteria (PDCs).

The applicable regulatory requirements for the staff's evaluation of the SSC safety classification, PSFs, and PDCs, are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(1), (3), (4), and (5) – Contents of Applications; Technical Information
- 10 CFR 50.35, "Issuance of Construction Permits"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

The applicable guidance for the staff's evaluation of the SSC safety classification, PSFs, and PDCs, are as follows:

- 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," (Agencywide Documents Access and Management System (ADAMS) Accession No.: ML20091L698) which endorsed NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A472), with clarifications.
- RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML23269A222) which endorsed NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology," Revision 1 (ML22060A190), with clarifications and additions.
- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," section 1.1.4, "Developing Proposed Principal Design Criteria (PDC) for Those Aspects of the Facility Design not Informed by the [Licensing Modernization Project] LMP Process (e.g., Normal Operations)" (ML23277A139).
- RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," (ML17325A611).

5.1 Safety Classification of SSCs

The regulations in 10 CFR 50.34(a)(1) and (4) together require that each CP applicant supply a safety assessment with a preliminary analysis and evaluation of the design and performance of facility SSCs. Specifically, 10 CFR 50.34(a)(1)(ii)(D) requires consideration be given to the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. The requirements of 10 CFR 50.34(a)(4) specify that the objective of the preliminary analysis and evaluation of the design and performance of SSCs is to assess the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and mitigation of consequences of accidents. These required assessments of the facility provide an understanding of those SSCs that are most important to radiological health and safety because they play a key role in preventing and mitigating accidents. Throughout the regulations in 10 CFR Part 50, it is established that certain SSCs are safety-related (SR) (as defined in 10 CFR 50.2) and these SSCs must be provided with augmented design standards, quality requirements, etc. as necessary to ensure their reliability and capability to prevent and mitigate accidents; other requirements may apply to SSCs that are safety-significant but not SR.

The guidance in NEI 18-04, as endorsed with clarifications in RG 1.233 provides an acceptable process for using the PRA and licensing basis event (LBE) analyses to classify an SSC based on the role it plays in prevention and mitigation of accidents, as described in more detail in chapter 3 of the PSAR and this SE. Under this process, the most safety-significant SSCs are classified as SR, SSCs that perform safety-significant functions but are less safety-significant are classified as non-safety-related with special treatment (NSRST), and other SSCs are classified as non-safety-related with no special treatment[1]. These classifications are an integral part of the licensing basis development process described in NEI 18-04, because they are used to assess SSCs according to their safety- or risk-significance and identify if special treatments beyond normal industrial practices are needed to ensure SSC performance of safety functions in the prevention and mitigation of licensing basis events.

Section 5.1 of the KU1 PSAR states that the safety classification of SSCs was performed in accordance with NEI 18-04 as endorsed in RG 1.233. There are multiple steps in the SSC classification process as described in sections 3 and 4 of NEI 18-04, and each one results in a different combination of safety-significance and safety classification. Table 5.1-1 provides a summary of these steps, including the scope of LBEs to be evaluated, the parameters used in the evaluation, and the resulting risk importance category and classification for the functions and SSCs that perform those functions.

The staff reviewed the applicant's SSC safety classification process implementation to verify it was consistent with NEI 18-04 as endorsed in RG 1.233. As stated in section 5.1 of NEI 21-07, the applicant's PSAR should include specifics of how the LMP methodology was applied within the range of options specified in NEI 18-04. USO did not provide the specifics of how the LMP methodology was applied within PSAR section 5.1. The staff reviewed USO's implementation of

[1] The staff notes that because the definition of safety-related in NEI 18-04 is not identical to that provided in 10 CFR 50.2, the staff initiated an exemption to apply the NEI 18-04 definition, as described in appendix X of this SE.

the LMP methodology through an audit of internal documentation. The staff found this review approach acceptable for the USO's CP application review because USO is the first applicant to the NRC to use the LMP methodology, however these specifics should be clearly described within section 5.1 of the FSAR to meet the guidance in NEI 21-07, as endorsed in RG 1.253.

The staff conducted an audit of USO's implementation of the LMP methodology as documented in the General Audit Report (MLXXXXXXXXX). The staff gained an understanding of the process used for SSC classification through audit questions, audit discussions, and review of USO's internal documentation. The staff audited relevant design basis documents to gain an appropriate level of understanding of the functions and the SSCs used to perform them. During the audit the staff was provided with a demonstration of the software used to evaluate the impact of each function or SSC on the LBEs. The staff also reviewed the inputs and outputs of the classification process to evaluate whether the process was being executed as defined. The staff considered the radiological material in the various plant systems that would be available for release when evaluating the mitigating impact of functions relative to the radiological consequences presented in chapter 3. This information also supported a risk-informed approach to the review, allowing the staff to focus on areas where failure of mitigating features could result in the highest consequences.

During its review of the USO CP, the staff identified certain areas of the NEI 18-04 guidance and corresponding endorsement in RG 1.233 that could benefit from additional clarification. In addition, the staff identified areas where USO deviated from the LMP methodology. During the audit, these deviations were assessed and USO implemented changes to its process to align more closely with the LMP methodology as discussed in the staff's request for confirmation of information (RCI) dated September 16, 2025 (ML25259A180). Additional discussion is provided below to document those deviations and provide explanation of how the LMP methodology was implemented by USO and evaluated by the staff.

<u>Identification of risk-significant PSFs</u>

In response to the staff's RCI, USO confirmed, with one clarification as described in the RCI response, the following regarding its implementation of the LMP methodology:

> The LMP methodology documented in NEI 18-04, as endorsed by RG 1.233, states that, "An SSC is classified as risk-significant if… a prevention or mitigation function of the SSC is necessary to meet the design objective of keeping all LBEs within the F-C Target." The methodology further clarifies that, "An LBE is considered within the F-C Target when a point defined by the upper 95th percentile uncertainty on both the LBE frequency and dose is within the F-C target." This evaluation is performed by assuming failure of the SSC in performing a prevention or mitigation function and checking how the resulting LBE risks compare with the F-C target.
>
> These risk-significant functions are designated as required safety functions (RSFs) and result in safety related (SR) classifications if they are necessary to meet the F-C target for DBEs or high-consequence BDBEs when evaluated using mean risk values. The remaining risk-significant functions identified through the evaluation described above are classified as risk-significant NSRST functions. USO unintentionally deviated from the LMP methodology in two ways while performing these risk-significant function determinations for the CPA:

1. Identification of RSFs was performed using the 95th percentile versus mean risk values.

2. Identification of risk-significant NSRST functions needed to keep LBEs within the F-C target was not performed.

The first deviation is conservative and may have resulted in a small number of SSCs that should have been classified as risk-significant NSRST being classified as SR, reducing the impacts of the second deviation. Impacts of the second deviation are further reduced by an additional quantitative assessment step performed by USO as part of the defense in depth (DID) evaluation, beyond what is outlined in the LMP methodology. In this additional step, various sets of non-SR functions were evaluated, collectively with the SR functions, to identify the minimum set of additional non-SR functions that were needed to ensure that all LBEs were below the F-C target when assessed at the mean risk values. The set of additional non-SR functions were classified as NSRST for DID.

With this additional assessment step performed assuming all non-SR functions not in the evaluation set failed, this resulted in many of the SSCs that were not identified as risk-significant NSRST because of the second deviation being classified as NSRST for DID. With this and the use of the 95th percentile in identifying RSF, the impact of this second deviation on the overall design and facility risk at the CP stage is small. USO has modified their process and workflow to ensure that this step for identifying risk-significant NSRST functions is performed moving forward, in a manner consistent with the LMP methodology.

<u>Safety classification for SSCs needed to reduce the likelihood of initiating events (IEs)</u>

NEI 18-04 section 3.3.4 states that, "[a] PSF, as used in the LMP, is any function by any SSC modeled in the PRA that is responsible for preventing or mitigating a release of radioactive material from any radioactive material source within the plant." Throughout NEI 18-04, where functions are discussed, phrases like "prevent or mitigate" or "prevention and mitigation" are used, indicating that both types of functions are important to ensuring safety and both should be evaluated at each step of the SSC classification process.

During its review of the KU1 PSAR, the staff identified that some preventative functions were not being evaluated through the SSC safety classification process. These preventative functions included interlocks on fuel handling movements, **[[** [redacted] **]]** USO noted in a supplement to its PSAR dated September 10, 2025 (ML25253A386) that these preventative functions were not viewed as PSFs within its process because they were incorporated into the PRA within the initiating event fault tree or not explicitly modeled within the PRA. This approach limited the evaluations that could be performed in assessing the impact on risk.

Following audit discussions, USO made two modifications to its process in order to ensure that preventative controls were addressed in the classification process. USO provided descriptions of these process changes in the supplement dated September 10, 2025. The first was to move preventative functions with a failure on demand from the initiating event fault tree into a top gate to be evaluated as a PSF within the USO process. This modification primarily impacted on the interlock functions. The second was to add the following steps to the USO DID evaluation process:

[[[redacted]]]

As identified by USO in the supplement, a preliminary assessment of the process indicated that these new steps would lead to SSCs, **[[** [redacted] **]]** being classified as NSRST for DID adequacy.

This alternative process for the SSC classification of preventative functions deviates from what is outlined in NEI 18-04. The process in NEI 18-04 indicates that preventative functions serving to reduce the likelihood of IEs are PSFs that should go through the same SSC classification process as mitigative functions. USO's alternative process has the potential to result in some preventative SSCs that could have been classified as risk-significant being classified as NSRST for DID because evaluations of whether the function is needed to keep LBEs below the F-C target are not performed. While the alternative process does have the potential to lead to less conservative safety classifications, the staff determined that it provides a reasonable method of ensuring safety based on the following:

- The threshold of **[[** [redacted] **]]** is low relative to the F-C target in the DBE and BDBE region, which is likely to result in a conservative number of events being identified as requiring NSRST preventative controls.
- With functions that have a probability of failure on demand being moved out of the fault trees and fully evaluated for the SSC safety classification, the number of potentially risk-significant SSCs impacted by this deviation is expected to be small.
- The integrated decision-making process (IDP) and IDP panel will review the roles of these preventative SSCs and evaluate where additional controls may be needed to meet DID adequacy or SSC classifications may need to be elevated for risk significance.

Based on these considerations, the staff determined that the alternative process provides a reasonable method of ensuring safety and additional review of the process, its implementation, and the resulting impact on SSC classifications is appropriate to consider at the OL stage.

<u>NSRST SSCs credited at degraded performance level in DBA analyses</u>

NEI 18-04 contains the following statements in sections 5.3 and 5.6.2:

a. DBAs are then constructed, starting with each DBE, and then assuming that only the SR SSCs perform their associated RSFs.

b. It is appropriate that for the inherent capabilities of passive functions, degradation of the passive function is considered, as opposed to complete failure (i.e., a physical non-existence of that function).

For SSCs with active functions, USO followed statement (a). However, based on USO's interpretation of statement (a) that NSRST SSCs may continue to function but not perform RSFs, and statement (b), USO incorporated some passive NSRST SSCs performing at a degraded level in their DBA analyses. The NSRST SSCs credited at a degraded level are generally large passive structures such as confinement barriers and buildings, where the ability to perform the NSRST function would not be impacted by the IE. Examples of this are discussed in section 3.8.3.4 of this SE for releases from ex-vessel systems. The staff considered if USO's approach to ascribing some amount of degraded performance to passive NSRST SSCs in DBA analyses was acceptable and determined it was reasonable, but is dependent on the design aspects of the SSCs and the level of degraded performance credited. Since the design is preliminary, a final determination on the acceptability of this approach to DBA analysis will be completed at the OL stage when the final design and degraded performance will be provided.

Seismic interaction aspects of non-SR SSCs

With respect to the following statement in NEI 18-04:

> SR-classified SSCs are required to perform their RSFs following a Safe Shutdown Earthquake, NSRST and [non-safety-related with no special treatment] NST SSCs required to meet Seismic II/I requirements (required not to interfere with the performance of SR SSC RSFs following a Safe Shutdown Earthquake).

The following statement in RG 1.233:

> None of the non-SR with no special treatment SSCs are classified as safety significant, but they may have requirements to ensure that failures following a design-basis internal or external event do not adversely impact SR or NSRST SSCs in their performance of safety-significant functions.

And the following statement in NEI 21-07:

> When a non-safety-related SSC is required to protect the SR SSCs in their ability to perform their RSFs, such non-safety-related SSCs are not necessarily NSRST. The NSRST classifications are based on the PRA Safety Functions these SSCs perform to prevent or mitigate event sequences and not functions that are focused on protecting the SR SSCs.

While the above statement from NEI 18-04 could imply that only SR SSCs need to be protected from a seismic event, the statement from RG 1.233 notes that requirements may also be needed to protect the ability of NSRST SSCs to perform their safety-significant functions. Furthermore, RG 1.253, Item C.3(e) clarifies that NSRST SSCs credited in non-DBA LBEs or relied on to establish adequate DID may need to be specially designed to withstand or be protected from the design basis hazard levels (DBHLs) or beyond-design-basis hazards. Because the LMP methodology reduces the use of SR equipment by using NSRST items to mitigate events in the AOO, BDBE, and 95th percentile consequence regions as well as provide DID, widespread failure of NSRST from a seismic event could result in an unacceptably high consequence. For this reason, seismic interactions that may interfere with an NSRST SSC's ability to perform its safety-significant function are evaluated and addressed appropriately based on seismic risk significance. USO describes how seismic risk significance is determined for NSRST SSCs in section 6.4.1.1 of the PSAR and addresses consideration of NSRST SSCs as seismic interaction targets in section 6.4.1.5.2 of the PSAR, providing reasonable assurance that the safety-significant functions needed to ensure safety during and following a design basis seismic event can be performed. This is consistent with the guidance in RG 1.233 provided above.

Note, while USO is identifying SSCs that are a source of safety concerns with respect to seismic interaction, USO is not including these considerations in their SSC safety classification which is consistent with the statements from RG 1.233 and NEI 21-07 provided above. However, NEI 21-07 also includes discussion for the consideration of "special safety functions" for NST SSCs. Therefore, USO has expanded the scope of seismic interaction beyond support for systems or components located above or near SR and NSRST SSCs to buildings, including the reactor building (RXB) superstructure, and any component containing sodium.

NEI 21-07 section 6.1.3 indicates that design information for non-SR SSCs having special treatment for seismic interaction should be identified and described in the FSAR, including identification of the SSC being protected, the DBHL associated with the requirement, and the specific design requirement. To address the safety aspects of these "special safety function" requirements for NST SSCs, the staff requested the applicant describe the extent to which programs are applied to ensure the safety aspects are met. Details of this were audited, and USO added statements on how programmatic controls are applied to seismic interaction sources to PSAR section 6.1.3.1. The staff also requested additional design information regarding the RXB superstructure, which was added to PSAR section 7.8.

5.1.1 Conclusion

Based on its review as documented above and supported by the General Audit, the staff determined USO's SSC safety classification process meets the requirements in 10 CFR 50.34(a)(1), (4), and 10 CFR 50.35 and is acceptable for the CP application considering the following:

- Apart from the deviations discussed above, USO's implementation of the LMP methodology appears to be consistent with the guidance in NEI 18-04, as endorsed in RG 1.233. This methodology provides reasonable assurance that safety-significant SSCs and their associated PRA safety functions will be identified and appropriately classified to support the application of the necessary design, construction, and operational requirements needed to ensure safety.

- The impacts of deviations from the LMP methodology within the preliminary design have been assessed and found to be small relative to the overall facility risk. Based on this, correction of these deviations can reasonably be left for later consideration and review at the OL stage.

- The process USO will be implementing to develop the OL provides reasonable assurance that the final design will conform to the LMP methodology and provide adequate protection of the health and safety of the public.

Table 5.1-1. SSC Classification Steps in NEI 18-04 Methodology

Task*	LBEs Evaluated	Evaluation	Criteria Compared Against	Values Used	Resulting SSC Classification and risk importance category
7a.1	DBEs and AOOs and BDBEs with uncertainty bands that extend into the DBE region	Remove individual functions and evaluate impact on individual LBEs	F-C target curve	Mean	SR required safety function (RSF), risk-significant and safety-significant
7a.2	BDBEs with mean consequences that exceed 25 rem TEDE	Remove individual functions and evaluate impact on individual LBEs	F-C target curve	Mean	SR RSF, risk-significant and safety-significant
7d	DBAs	Conservative consequences evaluation in which only available SR SSCs perform their function	10 CFR 50.34 dose limits	95^{th} percentile or greater	SR RSF, safety-significant
7a.3	All non-DBA LBEs	Remove individual functions and evaluate impact on individual LBEs	F-C target curve	95^{th} percentile	Non-safety-related with special treatment (NSRST), risk-significant and safety-significant
7b	All non-DBA LBEs	Remove individual functions and evaluate total frequency of all associated LBEs	1% of cumulative risk metrics	Mean	NSRST, risk-significant and safety-significant
7e	All non-DBA LBEs	Perform DID adequacy evaluation using the integrated decision-making process (IDP) and IDP panel (IDPP)	Table 5-2 and other DID concepts described in Section 5 of NEI 18-04	N/A	NSRST, safety-significant

** From section 3.2.2 of NEI 18-04. Note, task 7c of determining risk significance is an assessment of the results determined in steps 7a and 7b*

5.2 Safety-Significant PRA Safety Functions

As discussed in section 5.1, 10 CFR 50.34(4) requires a preliminary analysis and evaluation of SSCs and their adequacy in the prevention and mitigation of accidents. USO performs this evaluation using the methodology in NEI 18-04, as described in section 5.1 of the SE. In the NEI 18-04 process, as shown in Figure 1.3-1 of this SE, the safety-significant PSFs are identified and then SSCs are classified based on the PSFs they perform. The category of the function and the SSC classification are based on which step of the process the safety-significant function was identified. See Table 5.1-1 for a summary of these steps.

The information in section 5.2 of the PSAR is the output of the SSC classification process: the safety-significant functions, the SSCs performing those functions, and the classifications of those SSCs. The technical evaluation of this information has been divided in this SE to more closely align with the recommended structure in NEI 21-07, with RSFs and SR SSCs (Table 5.2-1 through 5.2-3 of the PSAR) evaluated in section 5.4 and NSRST functions and SSCs (Table 5.2-4 of the PSAR) evaluated in section 5.5 of this SE. Section 5.5 of this SE also includes a summary of radionuclide retention barriers (SR and NSRST). This section of the SE focuses on the summary information provided outside the tables in PSAR section 5.2 and overall conclusions.

The staff reviewed PSAR section 5.2 to evaluate if the preliminary list of safety-significant PSFs, in PSAR tables 5.2-1 through 5.2-4, are consistent with NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253; or, if deviations were used, that the applicant provided a reasonable basis. As part of its evaluation, the staff audited relevant plant design basis documents and information associated with how probabilities, mechanistic source terms (MSTs), and radiological consequences were calculated. The staff's review was supported by the applicant's responses to audit questions pertaining to the PSFs as documented in the General Audit Report, and audit of underlying SSC classification documentation.

The PRA described in section 3.1 of the PSAR was used in the preliminary determination of safety-significant PSFs. As described in section 3.1 of the SE, due to the preliminary nature of the design, some SSCs are not currently modeled within the PRA. Because the SSC safety classification process relies on the results of the PRA to determine classification and risk significance, these SSCs will need to be incorporated prior to the OL application to confirm that the appropriate SSC safety classification and risk significance was assigned. At this preliminary design stage, USO identified these SSCs through the IDP and included them in PSAR table 5.2-4. In section 5.2.2 of the PSAR, USO committed to performing safety analysis evaluations to confirm the risk-significance of all non-SR PSFs at the OL stage.

Based on its review of the preliminary safety-significant PSFs described in PSAR section 5.2, the staff determined that the results are reasonable based on the preliminary PRA, LBEs described in chapter 3 of the PSAR, the SSC safety classification process described in section 5.1 of the PSAR and section 5.1 of this SE, and the preliminary design available at this time. In addition, the staff determined that the safety-significant PSFs were developed using the process described in NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253, with exception of the deviations noted in section 5.1 of this SE.

5.3 Principal Design Criteria

The regulations in 10 CFR 50.34(a)(3)(i), require a CP applicant to include in the PSAR the PDC for the facility. The PDC for KU1 are incorporated by reference into the PSAR via topical report (TR), NATD-LIC-RPRT-0002-A, Rev 1, "Principal Design Criteria for the Natrium Advanced Reactor," (ML24283A066). PSAR section 5.3, lists the PDC from the TR and provides a brief description of how each PDC is addressed by the design. The staff found the PDC for the Natrium plant to be acceptable, as documented in NATD-LIC-RPRT-0002-A, Rev 1. The staff's conclusions in the SE were subject to two limitations and conditions (L&Cs):

1. An applicant or licensee referencing this TR must propose a design that is substantially similar to the Natrium design as discussed in SE Section 1, or otherwise justify that any departures from these design features do not affect the conclusions of the TR and this SE.

2. The use of this TR is restricted to those applicants using the risk-informed, performance-based licensing process described in NEI 18-04, Revision 1, as endorsed by RG 1.233. Because the proposed PDCs may not fully address all performance requirements for SSCs defined as safety-significant under the NEI 18-04 process, applicants or licensees referencing this TR must augment the PDC in the TR with appropriate PDC for any SR or NSRST SSCs whose safety function relates to BDBEs, or NSRST SSCs needed for DID adequacy, or otherwise justify that the Natrium PDCs as described in the subject TR are adequate.

The staff reviewed the design information presented in the PSAR, particularly in chapters 1, 3, and 7, and determined it is consistent with the design as described in NATD-LIC-RPRT-0002, Rev. 1. Based on this, the staff determined L&C 1 is adequately addressed. The staff also identified that the KU1 application uses the NEI 18-04 process as described throughout this chapter of the SE and elsewhere, so this aspect of L&C 2 is adequately addressed.

The second aspect of L&C 2 relates to defining PDCs associated with performance of safety functions for BDBEs or as needed for DID adequacy. The staff reviewed the PDCs in section 5.3 of the PSAR and noted that USO did not identify any additional PDCs. However, the staff reviewed PSAR tables 5.2-1 through 5.2-3 and found that they identify SR functions needed to mitigate BDBEs and link them to performance of safety-related design criteria (SRDC) for these events.

With respect to NSRST functions, PSAR table 5.2-4 identifies them and indicates whether each is risk-significant or required for DID adequacy. It also identifies the SSCs needed to perform each function. The staff reviewed the safety-significant SSC descriptions in chapter 7 of the PSAR and identified that each section appropriately references the safety-significant functions supported by the SSC, including those that are NSRST (regardless of whether they are risk-significant or needed for DID adequacy), and provides performance requirements for the SSC to provide support for each function.

The staff's SE for NATD-LIC-RPRT-0002-A, Rev. 1, stated that "the NRC staff expects that any additional PDCs needed would be established with a minimum scope and content similar to that discussed in NEI 21-07, Section C, Section 5.6". This section of NEI 21-07 states that the complementary design criteria, which are intended to provide performance requirements similar to the PDCs for NSRST SSCs, may "be defined at the functional level (related to the PRA safety functions that are satisfied by the NSRST SSCs)". Because the performance requirements provided in chapter 7 describe functional performance necessary to support the NSRST functions, the staff determined that these performance requirements are consistent with the concept of complementary design criteria as discussed in NEI 21-07, and therefore meet the staff's expectations articulated in the SE. Because the staff determined above that performance of SSCs in BDBEs are appropriately tied to safety-related design criteria in PSAR tables 5.2-1 through 5.2-3 and performance of NSRST SSCs are appropriately tied to additional criteria in PSAR chapter 7, the staff determined the second aspect of L&C 2 is appropriately addressed.

Based on staff determinations above, and as incorporated by reference from NATD-LIC-RPRT-0002, Rev 1, the staff determined that the information regarding the PDC in PSAR section 5.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter. The table below describes where in the SE SSCs are evaluated relative to each PDC described in PSAR section 5.3.

Table 5.3.2-1

Criterion	Title	SE Section(s)
I. Overall Requirements		
1	Quality Standards and Records	1.4.4, 8.1
2	Design Basis for Protection Against Natural Phenomena	chapter 2, 6.1.1., 6.1.3, 6.4 and chapter 7
3	Fire Protection	7.5.3
4	Environmental and Dynamic Effects Design Basis	6.1.1, chapter 7
5	Sharing of Structures, Systems, and Components	Not Applicable – KU1 is a single unit site
II. Multiple Barriers		
10	Reactor Design	3.11, 3.12, 7.1.1, 7.1.2, 7.1.3, 7.1.4, 7.2.2, 7.6.2 and 7.6.3
11	Reactor Inherent Design	3.11, 7.1.1, and 7.1.2
12	Suppression of Reactor Power Oscillations	3.11, 7.1.1, 7.1.2 and 7.2.5
13	Instrumentation and Control	7.6
14	Primary Coolant Boundary	6.4.3, 7.1.2, 7.1.3, 7.2.3, 7.2.4, 7.2.5 and 7.2.6
15	Primary Coolant System Design	3.12, 7.1.2., 7.1.3, 7.2.3, 7.2.4, 7.2.5, 7.6
16	Containment Design	1.3.2.1, chapter 7
17	Electrical Systems	7.7.1
18	Inspection and Testing of Electric Power Systems	7.7.1
19	Control Room	7.5.1, 7.6.7, 7.8.4
III. Reactivity Control		

Criterion	Title	SE Section(s)
20	Protection System Functions	7.6.3, 7.6.4, and 7.6.5
21	Protection System Reliability and Testability	7.6.3, 7.6.4, and 7.6.5
22	Protection System Independence	7.6.3, 7.6.4, and 7.6.5
23	Protection System Failure Modes	7.6.3, and 7.6.4
24	Separation of Protection and Control Systems	7.6.2, 7.6.3, 7.6.4, 7.6.5, and 7.6.8
25	Protection System Requirements for Reactivity Control Malfunctions	7.6.2, 7.6.3, and 7.6.4
26	Reactivity Control Systems	3.11, 7.1.1, 7.1.2, 7.2.5, 7.6.2, 7.6.3
27	Combined Reactivity Control Systems Capability	In accordance with RG 1.232, this PDC was deleted and incorporated into PDC 26
28	Reactivity Limits	7.1.1, 7.2.5, 7.6.2
29	Protection Against Anticipated Operational Occurrences	7.1.1, 7.2.5, 7.6.2, 7.6.3, 7.6.4, and 7.6.5
IV. Fluid Systems		
30	Quality of Primary Coolant Boundary	7.1.2, 7.1.3. 7.2.3, 7.2.4, 7.2.5
31	Fracture Prevention of Primary Coolant Boundary	6.4.3, 7.1.2, 7.1.3, 7.2.3, 7.2.4, 7.2.5
32	Inspection of Primary Coolant Boundary	7.1.2, 7.1.3, 7.2.3, 7.2.4, 7.2.5 and chapter 8
33	Primary Coolant Inventory Maintenance	7.1.2, 7.2.4, 7.6.3
34	Residual Heat Removal	3.12, 7.1.2, 7.1.3, 7.1.4, 7.2.2
35	Emergency Core Cooling	7.1.2, 7.1.3, 7.1.4, 7.2.1
36	Inspection of Emergency Core Cooling System	7.1.2, 7.1.3, 7.2.1 and chapter 8
37	Testing of Emergency Core Cooling System	7.1.2, 7.1.3, 7.2.1
38	Containment Heat Removal	As described in TR NATD-LIC-RPRT-0002, Rev 0, the staff concluded that these PDC are not applicable to the Natrium design because the design does not include a pressure containing reactor containment structure. The functional containment basis is evaluated as part of PDC 16.
39	Inspection of Containment Heat Removal System	
40	Testing of Containment Heat Removal System	
41	Containment Atmosphere Cleanup	
42	Inspection of Containment Atmosphere Cleanup Systems	
43	Testing of Containment Atmosphere Cleanup Systems	
44	Structural and Equipment Cooling	7.5.1

Criterion	Title	SE Section(s)
45	Inspection of Structural and Equipment Cooling Systems	7.5.1
46	Testing of Structural and Equipment Cooling Systems	7.5.1
V. Reactor Containment		
50	Containment Design Basis	As described in TR NATD-LIC-RPRT-0002, Rev 0, the staff concluded that these PDC are not applicable to the Natrium design because the design does not include a pressure containing reactor containment structure. The functional containment basis is evaluated as part of PDC 16.
51	Fracture Prevention of Containment Pressure Boundary	
52	Capability for Containment Leakage Rate Testing	
53	Provisions for Containment Testing and Inspection	
54	Piping Systems Penetrating Containment	
55	Reactor Coolant Pressure Boundary Penetrating Containment	
56	Primary Containment Isolation	
57	Closed System	
VI. Fuel and Radioactivity Control		
60	Control of Releases of Radioactive Materials to the Environment	7.2.3, 7.4.1, 7.6.2, 7.6.6, 9.1, and 9.3
61	Fuel Storage and Handling and Radioactivity Control	7.1.2, 7.2.4, 7.3.1, 7.3.2, 7.3.3, 7.4.1, 7.6.2, chapter 8, 9.1.1, 9.3.1
62	Prevention of Criticality in Fuel Storage and Handling	3.13, 7.1.2, 7.3
63	Monitoring Fuel and Waste Storage	7.2.3, 7.2.4, 7.3, 7.4.1, 7.6.2, 7.6.6, and chapter 9
64	Monitoring Radioactivity Releases	7.2.1, 7.4.1, 7.5.1, 7.6.2, and 7.6.6
VII. Additional Sodium-Cooled Fast Reactor Design Criteria[2]		
70	Intermediate Coolant System	7.1.4, 7.2.2, 7.2.3, and 7.2.4
71	Primary Coolant and Cover Gas Purity	7.1.3, 7.2.3, 7.2.4
72	Sodium Heating Systems	7.1.2, 7.1.3, 7.1.4, 7.2.2, 7.2.3, 7.2.4, 7.3.2, and 7.6.2
73	Sodium Leakage Detection and Reaction Prevention and Mitigation	7.1.2, 7.1.4, 7.2.2, 7.2.3, 7.2.4, 7.3.2, 7.5.3

[2] PDC 70 through 79 are found in RG 1.232 appendix B. These PDC apply to both pool- and loop-type sodium-cooled fast reactors. As noted in the RG applicants/designers may also develop entirely new PDC as needed to address unique design features in their respective designs.

Criterion	Title	SE Section(s)
74	Sodium/Water Reaction Prevention/Mitigation	7.1.2, 7.1.3, 7.1.4, 7.2.2, 7.2.3, 7.2.4, 7.3.1, 7.3.2, and 7.5.2
75	Quality of the Intermediate Coolant Boundary	7.1.4, 7.2.2, 7.2.3, and 7.2.4
76	Fracture Prevention of the Intermediate Coolant Boundary	7.1.4, 7.2.2, 7.2.3, and 7.2.4
77	Inspection of the Intermediate Coolant Boundary	7.1.4, 7.2.2, 7.2.3, 7.2.4, and chapter 8
78	Primary Coolant Interfaces	7.1.2, 7.1.3, and 7.1.4
79	Cover Gas Maintenance	7.1.2, and 7.2.3
VIII. Additional Criteria Unique to the KU1 Design[3]		
80	Reactor Vessel and Reactor System Structural Design Basis	7.1.1, 7.1.2
81	Reactor Building Design Basis	7.8.1
82	Provisions for Periodic Reactor Building Inspection	7.8.1

5.4 Safety-Related SSCs

This section of the SE documents the staff's review of the SR functions and SR SSCs identified in PSAR section 5.2 and documented in PSAR table 5.2-1 through table 5.2-3. The staff reviewed PSAR section 5.2 to determine if the preliminary list of SR functions and SR SSCs was developed consistently with the methodology in NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253, with the exceptions noted in section 5.1 of this SE. To support its review, the staff audited relevant design documents to assess the details of the supporting data associated with the calculation of probabilities, MSTs, and radiological consequences that are part of the PRA. The staff also notes that an overview of all three fundamental safety functions and the SSCs necessary to support them with adequate defense-in-depth (including both SR and NSRST SSCs) is provided in section 1.3.1 of this SE.

Two SR functions for control of heat generation were identified for the preliminary design in PSAR table 5.2-1. These are the scram – gravity driven absorber insertion by latch release and reactor scram on loss of power. The former is the primary scram mechanism for the reactor and credited in most LBEs. The latter is a SR function within the SR control system to initiate a scam on loss of power, "in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at safe shutdown condition." Additional information on this SR function and the SSCs performing this SR function is provided in NAT-4950-A, Revision 2, "Instrumentation & Control Architecture and Design Basis Topical Report," (ML24305A009), PSAR section 7.6, and sections 7.2.1 of this SE. The staff determined this preliminary set of SR functions for the control of heat generation is reasonable because it provides at least one SR shutdown method for the LBEs requiring shutdown to not exceed 10 CFR 50.34 dose limits or the F-C target.

[3] In accordance with RG 1.232 appendix B, TerraPower identified PDC 80 and PDC 81 based on the unique features in the KU1 design.

Fourteen SR functions for heat removal were identified for the preliminary design in PSAR table 5.2-2. Of these, ten are passive SR functions: one for coastdown of the intermediate sodium pumps (ISPs), two for natural circulation in the reactor air cooling system (RAC) and the primary heat transport (PHT) system, and seven for passive heat removal aspects of various plant systems including fuel handling systems. The remaining four SR functions are pump trips for situations that would indicate failure or hazardous conditions in the systems that remove heat from the core, including high primary sodium temperature, low primary sodium level, low primary sodium level, and high primary sodium level.

Both the primary sodium pumps (PSPs) and ISPs trip on high primary sodium temperature to limit heat addition to the primary coolant. As described in PSAR section 7.6.3.2, the PSP and ISP trip on high primary coolant temperature only occurs if a scram signal exists in the RPS and neutron flux is below a setpoint, in addition to primary sodium cold pool temperature being above the setpoint. There is also a SR trip of the ISPs on high primary sodium level, which would indicate a leak in the intermediate heat exchanger (IHX). The final SR pump trip is a trip of the primary sodium processing system (SPS-P) pumps on low primary sodium level, which would indicate a leak within the system, possibly the SPS-P. This trip prevents the reactor vessel sodium from draining below its safe level.

To summarize, RAC is the SR means of heat removal from the reactor core, while other systems containing fuel (like the ex-vessel storage tank and ex-vessel handling machine) have similar SR passive heat removal functions. These functions are supported by SR functions related to natural circulation flow and pump trips necessary to limit heat addition or maintain coolant inventory. The staff determined that this preliminary set of SR functions providing control of heat removal is reasonable based on the preliminary set of LBEs described in chapter 3 of the PSAR and this SE. For a limited number of LBEs, the staff found that the analysis supporting sufficient heat removal was inconsistent with the description in chapter 3 of the PSAR. However, this did not impact the staff's determination for these SR functions and, as noted in chapter 3 of the SE, the event descriptions and analyses should be updated as needed at the OL stage to ensure consistency with the final LBEs.

Ten SR functions for radionuclide retention were identified for the preliminary design in PSAR table 5.2-3. These controls are radionuclide barriers, with the exception of the reactor enclosure system (RES) pressure relief which is located in the SCG and prevents pressure in the RES and supporting system functional containment barriers from exceeding their design pressure. The pressure relief is directed into the SCG vapor trap cell which has an NSRST function to isolate based on the position indication of the reactor vessel relief valves, as described in PSAR section 7.2.3.

The SR radionuclide retention barriers include the fuel cladding, RES, vessels used for fuel handling (e.g., ex-vessel storage tank (EVST) and bottom loading transfer cask (BLTC) barriers), and the pin removal cell (PRC) cell barrier. Due to the preliminary nature of the design, many details for these barriers are unavailable, preventing the staff from making any determinations related to whether the mitigation factors assumed for these barriers used in the generation of source terms in section 3.2 are acceptable. Additional design information will need to be provided at the OL stage to support findings on the acceptability of these barriers and their mitigating effects, including information on heating, ventilation and air conditioning (HVAC), and other connections and the isolations needed to achieve the target leak rates.

The staff notes that failure of SR and NSRST radionuclide barriers were generally treated as a degraded performance condition, rather than complete failure. In most cases, the degraded performance was assumed to be 100% leakage per day or approximately 4% per hour. Justification for degraded performance assumed post-failure will be reviewed at the OL stage.

The SCG isolation radionuclide retention SR function, DL3-RR10, was identified as necessary during the staff's review. This function provides a SR isolation of the normal process flow path out of the reactor vessel. Without this isolation, radionuclide releases into the reactor vessel would be exhausted through this pathway, leading to EAB TEDE values that could potentially exceed the 10 CFR 50.34(a)(1) dose criteria.

The staff notes that the need for an SR SCG isolation was not originally identified from the LBE results in PSAR chapter 3 because the preliminary PRA did not include this SR function as a PSF or as a potential bypass path for the RES barrier. The bypass path of an SCG pipe break was included, but with the low probability of a pipe break occurring concurrently with a release into the reactor vessel, the SR function to automatically close the SCG valve on leak detection (DL4-RR4a) to mitigate this bypass is classified as an NSRST function. The detection for this isolation is also NSRST and located outside the SCG piping, meaning it would not be effective in detecting a release in the reactor vessel without a concurrent pipe break.

LBEs involving radionuclide releases into the RES were evaluated with MSTs that assumed the RES barrier was isolated and could maintain 1% leakage per day. By comparison, the normal process flow path that was not modeled would move the release from the RES into the SCG and subsequently into the RWG within hours based on preliminary design flowrates, effectively bypassing the RES function. The addition of the SCG isolation SR function, including the necessary detection and actuation signals, allows the RES barrier (DL3-RR1) to perform its function.

Statements were added to the chapter 3 radiological consequence sections associated with the seven LBEs impacted, documenting that this pathway was not modeled and that LBE analyses that include this pathway and the mitigation provided by the SR SCG isolation will be provided at the OL stage. The addition of this flow path to the analysis is expected to result in some increases to the associated source terms due to the radionuclides that exit through the SCG prior to detection and isolation. Because the time prior to isolation is expected to be short relative to the overall duration of the event and there is margin between the current consequence results and the F-C target, it is reasonable to support the conclusions at the CP stage and the full impacts of this change will be reviewed at the OL stage.

The staff notes that one of the most risk significant barriers that will require additional information in the OL application is the PRC, a hot cell where pins are removed from lead test assemblies (LTAs) by operators using manipulators, to be sent for further analysis. The failure of a pin during this operation has a frequency in the AOO range. In addition, the LTAs generally have less decay time than other fuel assemblies handled outside of the RES, meaning there is a higher radionuclide inventory available for release.

The staff gained an understanding of the current design for this cell through audit discussions. Since the design is preliminary and not yet fully justified, the staff could not determine if the consequence evaluation for the AOO involving the PRC (RFH-OERC-BL) was conservative. USO indicated that a more accurate design assessment including detailed design information, consistent with the level of associated risk, would be provided and staff will review the PRC at the OL stage. The staff determined that this information was reasonable to leave to the OL because the PRC barrier is already identified as SR and reasonable special treatments associated with the design and construction of the PRC were provided, as discussed in section 7.3.2 of this SE.

5.4.1 Conclusion

The staff reviewed the SR functions and SR SSCs described in PSAR section 5.2 and determined that the SR SSCs meet the requirements in 10 CFR 50.34 and 10 CFR 50.35, and are consistent with guidance in NEI 18-04, as endorsed in RG 1.233. The SR SSCs identified provide reasonable assurance of protection for the LBEs described in PSAR chapter 3 and were appropriately identified using the SSC safety classification process described in section 5.1 of the PSAR and SE based on preliminary design. SR SSCs will need to be reevaluated at the OL stage and additional design information provided. Justification should be provided for any radionuclide retention barrier leak rates and isolation functions needed to achieve those leak rates.

5.5 Non-Safety-Related with Special Treatments SSCs

This section of the SE documents the staff's review of the NSRST functions and SSCs identified in PSAR section 5.2 and documented in PSAR table 5.2-4. The staff reviewed PSAR section 5.2 to determine that the preliminary list of NSRST functions and NSRST SSCs is consistent with NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253. To support its review, the staff audited relevant design documents to assess the details of the supporting data associated with the calculation of probabilities, MSTs, and radiological consequences that are part of the PRA documentation.

PSAR table 5.2-4 provides a preliminary list of all NSRST PSFs. This table includes whether the NSRST function is risk-significant or was identified through the DID evaluation. Risk-significant NSRST functions are those needed to maintain the 95th percentile LBE results below the F-C target and those that have a greater than 1% impact on the cumulative risk metrics. As described in SE section 5.1, USO deviated from the first of these two steps, however the use of the 95th percentile when identifying RSFs and the additional step within the DID evaluation compensated for the missed classification step. As described in SE section 5.1, USO has committed to perform the step as described in NEI 18-04 for the OL.

As discussed in PSAR section 5.2, SSCs that have no associated LBEs have been identified as DID but may be risk-significant once modeled within the PRA. As stated in PSAR section 5.2, USO has committed to performing analyses to confirm the risk significance of all non-SR PSFs at the OL stage. The following are NSRST actions that are redundant to SR functions. The following NSRST functions are not credited in the PRA and are identified as necessary for DID adequacy:

- Manual reactor scram
- Manual PSP trip
- Manual ISP trip
- Manual sodium processing system (SPS) pump trip on leak indication
- PSP trip automatic backup
- ISP trip automatic backup

The following are NSRST alternative shunt trips that are identified as risk-significant defense line functions in the most recent version of the PRA, but are not described in detail within the PSAR:

- Alternative shunt trip on high core exit temperature
- Alternative shunt trip on low primary sodium outlet pressure
- Alternative shunt trip on high IHT level
- Alternative shunt trip on low IHT level

The staff determined that this preliminary set of NSRST trip functions for control of heat generation and heat removal is reasonable based on the preliminary set of LBEs described in chapter 3 of the PSAR and this SE because they provide redundant trip capability to the SR functions discussed in SE section 5.4. Risk significance determinations will depend on the reliability of the primary trip function and potential consequences of failure. These will be reviewed at the OL stage.

Other NSRST functions identified as risk-significant include:

- CRD driveline scram follow
- FHP emergency makeup

Additional NSRST DID functions include the automatic seismic trip, ISP trip on low IHT level, IAC passive mode operation, post-accident monitoring, and intermediate leak guard piping. With the exception of the IAC passive mode operation, these functions do not have associated LBEs evaluated in the PSAR.

The staff determined that these additional NSRST functions for controlling heat generation and heat removal are reasonable and consistent with the preliminary set of LBEs described in chapter 3. As the PRA continues to mature and LBEs are developed to evaluate the risk significance of these functions, SSC and risk significance classifications may change. The staff notes that the SSC classification for the intermediate leak guard piping might change as a result of the fire PRA which will be provided for review at the OL stage.

There are two guard vessels listed in table 5.2-4 of the PSAR as NSRST for DID adequacy: the guard vessel for the reactor vessel and the EVST guard tank. These do not have an associated LBE in the PSAR, which is due to the low likelihood of failure associated with the reactor vessel and the EVST, causing events involving these functions to be below the 5×10^{-7} frequency cut-off for LBEs. Further evaluation of the basis for reactor vessel leak frequencies is provided in section 3.4.3 of this SE. Functions and SSCs with very low failure rates will be reviewed by the staff at the OL stage to ensure that the bases support the frequencies being used within the PRA.

The staff developed SE table 5.5-1 based on the source term and LBE descriptions in PSAR chapters 3 and the tables in PSAR chapter 5, which provides a summary of the radionuclide retention functions mapped to the material-at-risk (MAR) they are mitigating and the LBE scenarios they are associated with. The staff noted that all NSRST radionuclide retention functions in PSAR table 5.2-4 are listed as required for DID adequacy, however the staff notes some of these functions could be classified as risk-significant at the OL stage once the risk significance determination has been performed.

SE table 5.5-1 identifies where evaluations were performed, assuming degraded performance of the radionuclide retention barrier. USO generally credited degraded performance of NSRST radionuclide barriers in LBEs where that barrier was assumed to be failed, or for NSRST barriers within a DBA analysis, assuming that even in a failed state there would be some amount of mitigation provided by these barriers. For the SPS cell barrier, the nominal leakage was assumed to be 10% per day and the degraded performance was assumed to be 50% per day. Due to the short half-lives of sodium isotopes, the degraded performance of the SPS cell has a significant impact on consequences. As noted in SE section 5.1, USO will need to provide bases for the nominal and degraded performances of radionuclide retention barriers used in the source term analyses at the OL stage.

The staff makes the following observations when evaluating SE table 5.5-1 regarding the role of NSRST radionuclides barriers in LBE mitigation for KU1:

- For events with the largest MAR (i.e., fuel releases during operation), an SR barrier (in addition to the SR fuel cladding) and an NSRST barrier is needed to provide sufficient margin and defense-in-depth.
- For fuel release after shutdown, an SR barrier (in addition to the SR fuel cladding) is generally sufficient unless the frequency of the initiating event plus SR barrier failure is high enough that an LBE involving the SR barrier failure is included. In this case, an NSRST barrier is needed to provide mitigation when the SR barrier fails.
- For releases from auxiliary systems, the MAR is generally low enough that only one NSRST barrier is needed, in addition to the process barrier, to reach acceptable risk.
- The gaseous radwaste processing system (RWG) inventory and tritium MAR are low enough that no mitigation is needed to reach acceptable risk when the process barrier fails.

The staff determined that this preliminary set of NSRST radionuclide retention barriers is reasonable based on the preliminary set of LBEs described in chapter 3 of the PSAR and this SE because they provide sufficient mitigation and DID to maintain LBEs below the F-C target and within 10 CFR 50.34 limits.

5.5.1 Conclusion

The staff reviewed the NSRST functions and NSRST SSCs described in PSAR section 5.2 and determined that the NSRST SSCs meet the requirements in 10 CFR 50.34 and 10 CFR 50.35, and are consistent with guidance in NEI 18-04, as endorsed in RG 1.233. The NSRST SSCs identified provide reasonable assurance of protection for the LBEs described in PSAR chapter 3 and were identified using the SSC safety classification process described in PSAR section 5.1, with the clarifications in SE section 5.1, based on the preliminary design. The staff will review the risk significance of all NSRST functions at the OL stage based on the impact of the function on the LBE results relative to the F-C target and the total impact on the facility risk based on the cumulative risk metrics.

Table 5.5-1 Radionuclide retention functions mapped to MAR and LBEs

Material-at-risk (MAR)	Primary Barrier	Secondary Barrier	Tertiary Barrier	Licensing Basis Event (LBE)
Fuel Inventory	Fuel Cladding (SR)[i]	RES boundary (SR) SCG Isolation (SR)	Head Access Area (HAA) barrier (NSRST)	LFF-SAO-BL (DBE) DHP-LOOP-3 (BDBE) DHP-LOOP-4 (BDBE)
			HAA barrier (NSRST), degraded performance	LFF-SAO-1 (BDBE) LFF-SAO-CN (DBA) RFH-FDIV-CN (DBA)
		RES boundary (SR) SCG Isolation (SR) Ex-Vessel Handling Machine (EVHM) cask barrier (SR)		RFH-FDIV-1 (DBE) RFH-FDIV-3 (DBE)
		RES boundary (SR), degraded performance SCG Isolation (SR)	Reactor building (RXB), degraded performance (NST)	RFH-FDIV-2 (BDBE) RFH-FDIV-4 (BDBE)
			HAA barrier (NSRST)	LFF-SAO-2 (BDBE)[ii]
		EVHM cask barrier (SR)		RFH-LMCA-1 (BDBE) RFH-FDEM-1 (BDBE)
		EVHM cask barrier (SR), degraded performance		RFH-LMCA-2 (BDBE) RFH-FDBL-2 (BDBE) RFH-FDEM-2 (BDBE)
		PRC cell barrier (SR)		RFH-OERC-BL (AOO) RFH-FDRC-1 (BDBE)
		EVST barrier (SR)		RFH-FDET-1 (BDBE)
		BLTC barrier (SR), degraded performance	Fuel handling building barrier and filtration (NSRST)	RFH-ESWR-2 (BDBE)

Material-at-risk (MAR)	Primary Barrier	Secondary Barrier	Tertiary Barrier	Licensing Basis Event (LBE)
				RFH-ESWR-3 (BDBE)
		BLTC barrier (SR)		RFH-FDBL-1 (BDBE)
		BLTC barrier (SR) PIC radionuclide boundary (NSRST)		RFH-ESWR-1 (DBE) RFH-FDPI-BL (BDBE) RFH-ESWR-CN (DBA)
		PIC radionuclide boundary (NSRST), degraded performance		RFH-FDPI-1 (BDBE)
		Failed Fuel Canister (SR)		No associated LBEs
		Spent Fuel Pool (NST, for radionuclide retention)		RFH-FDSP-1 (DBE) RFH-FDSP-2 (BDBE) RFH-FDSP-CN (DBA)
SPS Inventory	Primary SPS barrier (NSRST)	HAA barrier (NSRST), degraded performance		RRS-SPLX-CN (DBA)
		HAA barrier (NSRST)[iii]	RXB superstructure (NST), 2-hour exhaust time	RRS-SPLX-BL (DBE)
		SPS cell barrier (NSRST)	RAB substructure (NSRST, not for radionuclide retention), 2-hour exhaust time	RRS-SPLA-BL (DBE)
		SPS cell barrier (NSRST), degraded performance		RRS-SPLA-CN (DBA)
SPS Tritium Inventory	Intermediate cold trap SPS barrier (NSRST)			RRS-ISPL-BL (DBE) RRS-ISPL-CN (DBA)
SCG Inventory		SCG cell barrier (NSRST)		No associated LBEs

<table>
<tr><th>Material-at-risk (MAR)</th><th>Primary Barrier</th><th>Secondary Barrier</th><th>Tertiary Barrier</th><th>Licensing Basis Event (LBE)</th></tr>
<tr><td rowspan="5"></td><td rowspan="5">Primary SCG barrier (NSRST)</td><td>SCG cell barrier (NSRST) or vapor trap cell barrier (NSRST), degraded performance</td><td></td><td>RRS-CGR-CN (DBA)</td></tr>
<tr><td rowspan="2">Vapor trap cell barrier (NSRST)</td><td></td><td>SUD-CGR-2 (BDBE)</td></tr>
<tr><td rowspan="2">NHV isolation (NSRST) and filtration (NST)</td><td>RRS-CGR-1 (DBE)</td></tr>
<tr><td rowspan="2">HAA barrier (NSRST)</td><td>RRS-CGR-BL (DBE)</td></tr>
<tr><td>NHV filtration (NST)</td><td>SUD-CGR-1 (BDBE)</td></tr>
<tr><td rowspan="3">RWG Inventory</td><td rowspan="3">Gaseous radwaste barrier (NSRST)</td><td>Holdup tank vault (NST)</td><td></td><td>RRS-RWG-1 (DBE)</td></tr>
<tr><td>Vent Stack Filtration (NST)</td><td></td><td>RRS-RWG-2 (DBE)</td></tr>
<tr><td></td><td></td><td>RRS-RWG-CN (DBA)</td></tr>
</table>

[i] Fuel Cladding is credited as a SR radionuclide retention barrier and prevents larger radionuclide release but is assumed to fail for the pins involved in the LBEs.
[ii] The event description for LFF-SAO-2 is that the vessel head fails while the HAA succeeds. However, the MST used has the vessel head succeeding and the HAA failing. A source term with accurate representations of the radionuclide retention barriers will be developed for this LBE at the OL stage. While the consequence for this LBE is expected to increase, the increase is expected to be small due to the use of non-DBA analysis methods at the OL stage, versus the more conservative DBA methods that were used in the PSAR.
[iii] SPS piping has an additional barrier for fire protection surrounding the piping in the HAA that is not credited in the LBE analyses.