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NUCLEAR TECHNOLOGIES AND MATERIALS ADVANCED REACTOR CONCEPTS-20

FAST MODULAR REACTOR SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

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GENERAL ATOMICS ELECTROMAGNETICS

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ACRONYMS

Acronym	Definition
AOO	Anticipated Operational Occurrence
ARC-20	Advanced Reactor Concepts-20
ARDP	Advanced Reactor Demonstration Program
ASME	American Society of Mechanical Engineers
BDBE	Beyond Design Basis Event
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Design Certification
DID	Defense-In-Depth
DOE	Department of Energy
EAB	Exclusion Area Boundary
EPA	Environmental Protection Agency
F-C	Frequency-Consequence
FDC	Functional Design Criteria
FMR	Fast Modular Reactor
FSAR	Final Safety Analysis Report
GA-EMS	General Atomics Electromagnetic Systems
GFR	Gas-cooled Fast Reactor
GT-MHR	Gas Turbine-Modular Helium Reactor
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
non-LWR	non-Light Water Reactor
NRC	Nuclear Regulatory Commission
NSRST	Non Safety Related with Special Treatment
NST	Non Safety Related with No Special Treatment
PAG	Protective Action Guide
PCS	Power Conversion System
PCU	Power Conversion Unit
PDC	Principal Design Criteria
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
QHO	Quantitative Health Objective
RFDC	Required Functional Design Criteria
RG	Regulatory Guide
RSF	Required Safety Function

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Acronym	Definition
RVCS	Reactor Vessel Cooling System
SDA	Standard Design Approval
SR	Safety Related
SRDC	Safety Related Design Criteria
SRP	Standard Review Plan
SSC	Structure, System, and Component
TCG	Turbine-Compressor-Generator
TI-RIPB	Technology Inclusive – Risk-Informed, Performance-Based
TLRC	Top-Level Regulatory Criteria

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1. INTRODUCTION

General Atomics Electromagnetic Systems (GA-EMS) is developing a helium-cooled Fast Modular Reactor (FMR) [Reference 1]. The project has been selected by the U.S. Department of Energy (DOE) for Advanced Reactor Concepts-20 (ARC-20) under Advanced Reactor Demonstration Program (ARDP). The long-term goal is to design, license, and commercialize the FMR plant by the mid-2030s. To achieve the goal of licensing the FMR, GA-EMS has been engaged with the Nuclear Regulatory Commission (NRC) from the initial stage of the project.

A fundamental aspect of the licensing process is the development of a comprehensive licensing basis. This entails creating a collection of documents and technical criteria that will serve as the foundation upon which the NRC will grant a license for a Standard Design Approval (SDA) or a Design Certification (DC) that will lead to the construction and operation of the nuclear facility. The NRC requires reactor designs to be evaluated based on several different kinds of events that are considered part of the licensing basis. Licensing basis events (LBEs) are certain event sequences that are chosen to be considered in the design of a nuclear power plant. These LBEs rely on plant structures, systems, and components (SSCs) to perform various safety functions. These SSCs are classified based on their safety and risk significance. As an effort to support the FMR pre-application regulatory engagement plan, GA-EMS has developed a process of safety classification of SSCs applicable to the FMR design.

Classification of SSCs for the FMR design will follow guidance developed by the Nuclear Energy Institute (NEI) and provided in NEI 18-04 [Reference 2]. This guidance provides an integrated and highly interdependent methodology for identifying and evaluating licensing basis events, classifying, and establishing performance criteria for SSCs, and evaluating defense-in-depth (DID) for advanced reactor designs. The process of SSC classification following this guidance will demonstrate the effectiveness of the selection process to facilitate the determination of safety and risk significant SSCs and the evaluation of DID adequacy. Also, some technology-inclusive, risk-informed, and performance-based (TI-RIPB) methods and process will be applied to the FMR SSC classification. It is very important that use of the TI-RIPB process for classification of SSCs follows guidance that is endorsed by the NRC and compatible with the latest regulatory framework. This is critical to ensuring that the FMR design meets the regulatory requirements and can be licensed and operated safely. The FMR design team will also engage with the NRC staff early and frequently in the design process to ensure that the classification of SSCs is consistent with the regulatory requirements.

The followings are included in this report:

- An overview of the regulations and guidance to be considered during development of the safety classification approach
- Description of the process of safety classification with a TI-RIPB approach

- Discussion of how probabilistic risk assessment (PRA) techniques will be used
- Summary and conclusions.

2. DESIGN FEATURES OF GA-EMS FMR

The FMR is a gas-cooled fast reactor (GFR), operating at system temperature range of 506 °C to 824 °C. It is a grid-capable power source with a gross electric output of ~44 MW. The reactor core uses helium coolant and uranium dioxide (UO_2) fuel pellets encapsulated in a silicon carbide (SiC) composite cladding, arranged in a triangular pitch, and forming a hexagonal fuel assembly.

The reactor core is an annular shape surrounded by solid reflector blocks of zirconium silicide (Zr_3Si_2) and graphite that preserve neutrons and enhance heat transfer. Zr_3Si_2 is a heavy reflector specifically developed for the GFR. This material is favorable in fast reactors to avoid power peaking around the core periphery from neutron thermalization.

Helium is chemically inert and will not aggravate an accident by contributing to any chemical or nuclear reaction. The use of helium as the coolant in combination with conventional fuel and effective neutron reflector offered enhanced neutronic and thermal efficiencies and several advanced safety characteristics such as efficient fuel utilization, high temperature operation, and inherently safe design that minimize the likelihood of accidents. For example, the helium coolant is intrinsically safe for it does not react with other materials or burn in air. The major systems and components are underground as illustrated in Figure 1.

The Power Conversion System (PCS) is a crucial component of the FMR power plant that converts the thermal energy generated by the reactor into electricity. The concept of the FMR PCS is similar to that for the gas turbine-modular helium reactor (GT-MHR). GA-EMS will develop the PCS of the FMR based on the previous experiences with the conceptual design of power conversion unit (PCU), leveraging the latest advancements in power conversion technology to optimize the efficiency and reliability of the PCS (i.e., PCU + generator system). The turbine-compressor-generator (TCG) are mounted on an inline vertical configuration. The generator is in a separate, connected vessel at the top of the PCU.

1. Reactor vessel
2. Reactor vessel cooling panel
3. Power conversion unit
4. Generator
5. Circulator
6. Maintenance cooling system
7. Steel containment
8. Concrete
9. Maintenance caps
10. Main control room
11. Water tank
12. Spent fuel storage
13. Fueling machine
14. Reactor building
15. Ground level

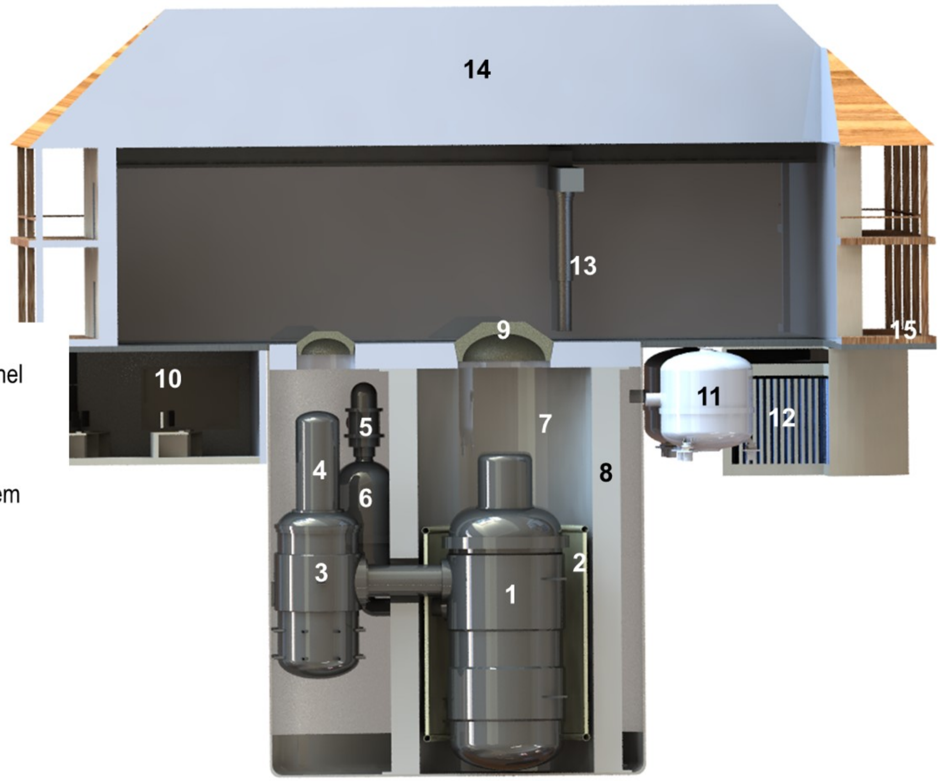

Figure 1. FMR Nuclear Island Components

One of the advanced design features of the FMR is its ability to passively remove decay heat from the core and vessel, regardless of whether helium is present. This is achieved through the implementation of a gravity-driven reactor vessel cooling system (RVCS). RVCS is always in operation [Reference 3] and continues passively removing the heat from the reactor vessel by natural circulation of water circulating in the RVCS loop. Unlike traditional gas-cooled reactors, which are typically packed with solid graphite, the FMR does not rely on conduction-cooldown. Instead, the passive safety of the core is primarily enhanced by the radiation heat transfer mechanism. For a rodded core like the FMR, the radiation heat transfer is the dominant heat transfer mechanism from the fuel rods to the surrounding solid structures, rather than conduction or convection.

Other design features, such as the large thermal margin, low power density, and annular core configuration, further enhance the passive safety of the core. Heat from the reactor vessel is transferred to the cooling panel of the RVCS through radiation. This system ensures that any decay heat generated by the core can be safely and efficiently removed, without the need for active cooling systems or other complex mechanisms. As a result, the FMR is able to offer exceptional levels of safety and reliability, making it an attractive option and a significant advancement in nuclear power generation technology.

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3. REGULATORY REQUIREMENTS AND GUIDANCES

3.1. Federal Regulations

The following are the federal regulations that are relevant to the licensing basis for nuclear power plants.

10 CFR Part 50:

This regulation establishes the licensing requirements and standards for the operation of commercial nuclear power plants. It covers various aspects such as safety, security, radiation protection, emergency preparedness, and decommissioning. Part 50 sets forth comprehensive safety and security requirements for nuclear power plants to ensure the protection of workers, the public, and the environment. It covers various aspects such as design, construction, equipment, operations, maintenance, emergency preparedness, and radiation protection.

The regulation establishes standards for radiation protection to limit and control occupational and public exposure to radiation. It includes requirements for monitoring, record-keeping, training, and the implementation of measures to minimize radiation risks. Part 50 mandates the development and maintenance of emergency preparedness plans by license holders. These plans outline procedures and measures to be taken in the event of an accident or other emergency, and they are coordinated with state and local authorities.

- 10 CFR 50.34 (Contents of applications; technical information) describes the minimum information required for (a) preliminary safety analysis reports (PSARs) supporting applications for a construction permit and (b) final safety analysis reports (FSARs) supporting applications for operating licenses.
- 10 CFR 50.69 describes requirements for risk-informing the categorization and treatment of SSCs using results and insights from a plant-specific PRA.
- 10 CFR Part 50 Appendix A provides the general design criteria for nuclear power plants, which include requirements related to reactor coolant systems, containment structures, emergency core cooling systems, and other important safety features.
- 10 CFR Part 50 Appendix B provides the quality assurance criteria for nuclear power plants as required by 10 CFR 50.34 for construction permit and operating license, and as required by 10 CFR 52.137 and 52.47 for an SDA or DC.

10 CFR Part 52:

This regulation governs the issuance of early site permits, standard DCs, combined licenses, SDAs, and manufacturing licenses for nuclear power facilities. For the FMR, the most relevant sections of 10 CFR Part 52 are as follows:

- 10 CFR 52.47 describes the information to be included in the FSAR supporting the application for a standard DC.

- 10 CFR 52.137 describes the information to be included in FSARs supporting SDAs.

An SDA or DC allows subsequent applicants to reference the certified design or design approval in their license applications, reducing duplicative reviews.

Related Guidance

- Reg. Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" provides guidance complying with the Commission's requirements in 10 CFR 50.69 with respect to the categorization of SSCs that are considered in risk-informing special treatment requirements.
- NUREG/BR-0303, "Guidance for Performance-Based Regulation" provides guidance on a process for developing performance-based alternatives in regulatory decision-making.

3.2. Risk-Informed Performance-Based Regulation

The NRC's policy statement titled "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" was issued in 1995 [Reference 4] and played a crucial role in expanding the use of risk-informed practices by both the NRC and the industry. A year later, the NRC commissioners issued a staff requirement memorandum (COMSECY-96-061) [Reference 5] that focused on prioritizing regulatory efforts on licensee activities that posed the highest risk to the public, in order to accomplish the agency's principal mission in a cost-effective and efficient manner. This memo was supported by a white paper (SECY-98-144) [Reference 6] issued by the agency in 1998 that defined terms such as "risk informed" and "performance based" and provided expectations for the implementation of risk-informed, performance-based approaches.

In 2012, the NRC published a strategic vision document, NUREG-2150 [Reference 7], that outlined options for a more comprehensive, holistic, risk-informed, and performance-based regulatory approach. This document built upon the prior policies of the NRC for the use of risk-informed practices and set expectations for all areas of the agency's activities in a comprehensive manner.

More recently, in SECY-18-0060 [Reference 8], the NRC staff proposed the development of a technology-inclusive, performance-based regulation as an alternative approach for licensing non-light-water reactors. The staff also proposed transforming the review process to use risk insights to guide the scope, focus, and depth of a review.

A risk-informed approach to regulation by the NRC incorporates an assessment of safety significance or relative risk, ensuring that regulations are appropriate to their importance in protecting public health and safety. This approach considers risk insights along with other factors to establish requirements that focus on design and operational issues relevant to safety. It enhances the deterministic approach by considering a broader range of safety challenges, prioritizing them based on risk significance, and allowing for the consideration of additional

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resources and uncertainties. It also improves decision-making by testing the sensitivity of results to key assumptions.

Performance-based regulation focuses on achieving measurable outcomes without direction from the NRC specifying how they should be achieved. It establishes performance criteria based on risk insights, deterministic analyses, and historical performance, providing flexibility to licensees to meet these criteria in ways that encourage improvement. It emphasizes monitoring system performance and offers incentives for safety enhancement without immediate safety concerns arising from failure to meet performance criteria. Measurable parameters can be included in regulations or license conditions, allowing for monitoring and assessment of performance.

The NRC has recommended the addition of 10 CFR Part 53 (Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants). The draft proposed rule in SECY-23-0021 offers a voluntary, performance-based alternative regulatory framework for licensing future commercial nuclear plants. In the context of this proposed rulemaking, future commercial nuclear plants, including non-light-water reactors (non-LWRs) and LWRs, would have the option to be licensed under Part 53. Applicants for these facilities would continue to have the option to be licensed under the existing requirements in 10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities) or 10 CFR Part 52 (Licenses, Certifications, and Approvals for Nuclear Power Plants).

3.3. Licensing Modernization Project and NEI 18-04

The Licensing Modernization Project (LMP), led by Southern Company and cost-shared by the U.S. DOE and other industry participants, proposed changes to specific elements of the current licensing framework and a process for implementation of the proposals. These proposals collectively led to modernization of the current licensing framework to support licensing of advanced non-LWRs. These proposals retained a high degree of nuclear safety, established stable performance-based acceptance criteria, and enabled near-term implementation of non-LWR design development, in support of national and industrial strategic objectives. The LMP objective was to support industry and NRC efforts to develop regulatory guidance for licensing advanced non-LWR plants.

The modernized framework is technology-inclusive to accommodate the variety of technologies expected to be developed. It is risk-informed because it employs an appropriate blend of deterministic and probabilistic inputs to each decision. It is performance based because it uses quantitative risk metrics to evaluate the risk significance of events and leads to formulation of performance requirements on the capability and reliability of structures, systems, and components to prevent and mitigate accidents. By utilizing a risk-informed, performance-based approach, the design and licensing efforts are more closely aligned with the safety objectives. The goal is efficient and effective development, licensing, and deployment of non-LWRs on aggressive timelines with even greater margins of safety than prior generations of technology. These goals fully support and reflect DOE and NRC visions for licensing and deploying advanced non-LWR plants.

The new framework consists of elements including:

- Establishment of TI-RIPB licensing-basis event selection,
- Classification of SSCs, and
- Establishment of predictable means to determine and preserve adequate DID.

These process steps are documented in NEI 18-04 Technical Report. This guidance provides an acceptable method for establishing the aforementioned topics as part of demonstrating that a specific design provides reasonable assurance of adequate radiological protection. The process is focused on establishing guidance for advanced designs so license applicants can develop inputs that can be used to demonstrate compliance with applicable regulatory requirements. The process does not exempt any reactor designer from existing regulations but describes an approach to inform the safety design approach, which can then be applied to demonstrate compliance with the regulations applicable to a reactor design. The process described in this guidance document is a systematic and reproducible process for selecting LBEs, classifying SSCs, and determining the adequacy of DID in the design of nuclear power plants.

The process is risk-informed, which means it uses insights from systematic risk assessment in combination with structured prescriptive rules to address the uncertainties which are not addressed in the risk assessment. The approach provides reasonable assurance that adequate protection is provided for public radiological protection.

The process is also performance-based. The performance-based approach evaluates the effectiveness relative to realizing desired outcomes that are achieved by using quantifiable performance metrics for LBE frequencies and consequences and performance requirements for SSC capabilities to prevent and mitigate events. This is an alternative to a prescriptive approach specifying particular features, actions, or programmatic elements to be included in the design or process as the means for achieving desired objectives.

The processes covered in this guidance document are integrated and highly interdependent, starting with the process for the selection of LBEs. The outcomes from executing the processes in this guidance support developing a risk-informed and performance-based safety basis for the design. The process is also helpful in developing a safety-focused application for NRC review by systematically demonstrating the following:

- The selected LBEs adequately cover the range of hazards specific to the design and reflect the appropriate SSC failure modes.

- The LBEs are defined in terms of successes and failures of SSCs that perform safety functions modeled in the PRA, which are responsible for preventing and mitigating unplanned radiological releases from any source within the plant.
- The SSCs that perform the safety functions are collectively capable, reliable, diverse, and redundant across the layers of defense in the design.
- The design and programmatic features included in the licensing application demonstrate the philosophy of DID, and the outcomes of DID adequacy evaluations ensure adequate layers of defense.
- Plant capabilities and programmatic capabilities are reconciled based on risk-informed insights to provide reasonable assurance of adequate protection.
- The safety and risk significance of plant SSCs and programmatic controls included in applications are commensurate with their scope and level of detail.

3.4. Reg. Guide 1.233

This regulatory guide (RG) [Reference 9] provides the NRC staff's guidance on using a TI-RIPB methodology to inform the licensing basis and content of applications for non-LWRs. This RG may be used by non-LWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants".

The NRC staff has determined that the methods described in NEI 18-04 constitute one acceptable means to identify LBEs, classify SSCs, establish special treatments, identify programmatic controls, and assess DID for non-LWRs. These activities also define a methodology for applicants to identify and provide the appropriate level of information needed to satisfy parts of the regulatory requirements in 10 CFR 50.34, 10 CFR 52.47, 10 CFR 52.79, 10 CFR 52.137, and 10 CFR 52.157.

4. LICENSING BASIS DEVELOPMENT PROCESS

The process for selecting and categorizing SSCs is part of a systematic and reproducible process for establishing the licensing basis. The other parts of licensing basis development are LBE selection and DID adequacy. The frequency-consequence evaluation criteria, hereafter referred to as the F-C Target, is shown in Figure 2 and is key to informing LBE selection, SSC classification, and DID adequacy. The F-C Target is divided into three regions corresponding to three LBE categories: anticipated operational occurrences (AOOs), design basis events (DBEs), and beyond design basis events (BDBEs). The fourth LBE category is design basis accidents (DBAs) which are postulated event sequences derived from DBEs. A separate white paper was prepared and submitted by GA-EMS describing the LBE selection process (Reference 10).

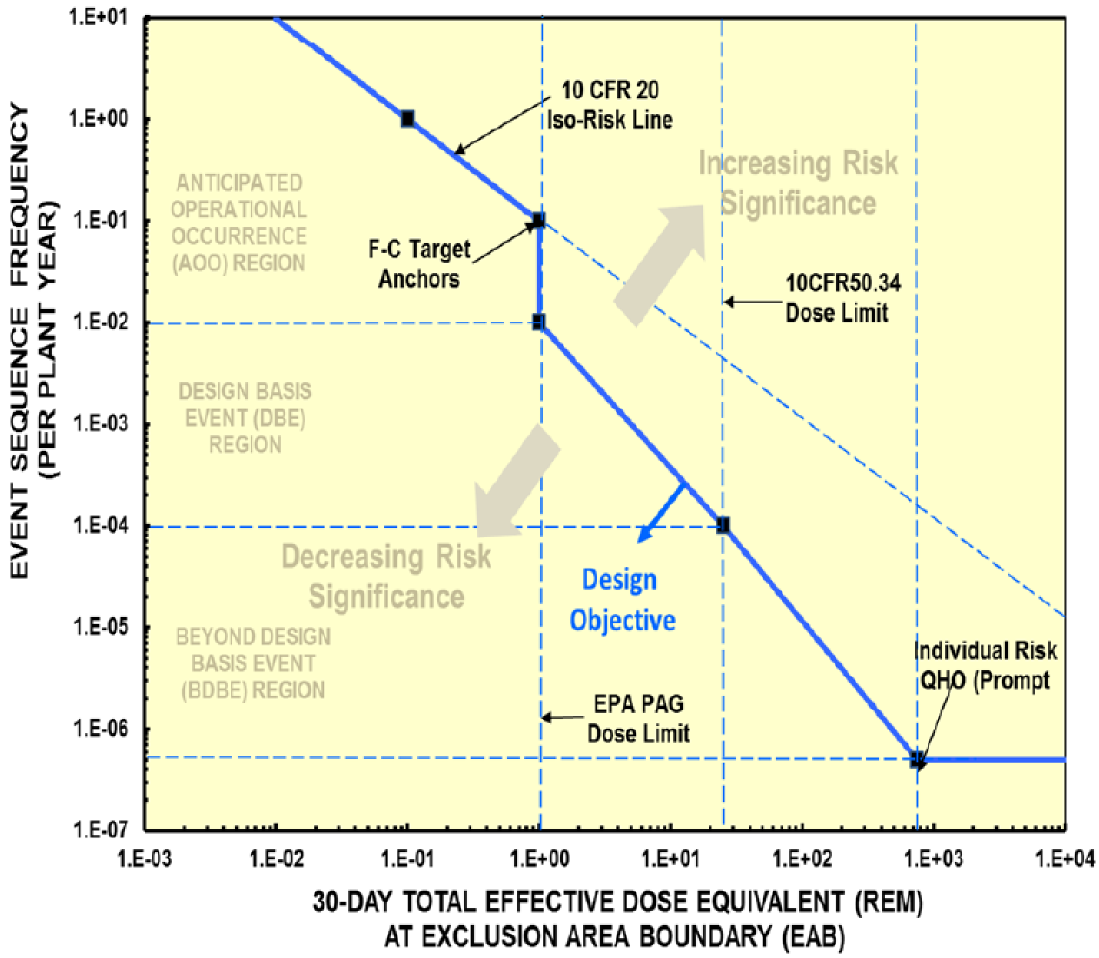

Figure 2 Frequency-Consequence Target

The frequency-consequence (F-C) evaluation criterion is a useful tool for assessing and managing risks associated with nuclear power plants. It provides a structured approach for evaluating hazards and identifying appropriate risk mitigation measures. The F-C evaluation criterion is a risk assessment approach used to evaluate the likelihood and potential impact of hazards that could result from the operation of nuclear systems. This approach is derived from the Top-Level Regulatory Criteria (TLRC), which outlines the regulatory requirements for ensuring the safety of certain operations. The F-C curve provides an acceptable limit in terms of the frequency of potential accidents and their associated consequences. The objective of the F-C curve is to establish the licensing basis, i.e., identify the event sequences that the design and operation of the plant need to be able to mitigate. The objective involves establishing criteria that define the acceptable frequencies for different levels of consequences.

Acceptable offsite dose evaluation criteria on the event sequence consequences for the LBE categories are defined by a frequency-consequence evaluation criteria derived from TLRC. Key elements of the TLRC used to develop the frequency-consequence criteria include:

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The F-C Target for lower frequency AOOs at frequencies of 10^{-1}/plant-year down to 10^{-2}/plant-year are set at a reference value of 1 rem corresponding with the EPA Protective Action Guide (PAG) limits [Reference 11] and consistent with the goal of avoiding the need for offsite emergency response for any AOO. It is expected that many LBEs will not result in the release any radioactive material, and the identification of plant capabilities to prevent such releases is a factor considered in the formulation of SSC safety classification and performance requirements.

The F-C Target for DBEs range from 1 rem at 10^{-2}/plant-year to 25 rem at 10^{-4}/plant-year with the dose calculated at the EAB for the 30-day period following the onset of the release. This aligns the lowest frequency DBEs to the limits in 10 CFR 50.34 and provides continuity to the lower end of the AOO criteria. A straight line on the log-log plot connects these criteria. The identification of plant capabilities to prevent releases is a factor considered in the formulation of SSC safety classification and performance requirements as discussed more fully in the section below on SSC safety classification. It is expected that many LBEs will not release any radioactive material.

The F-C Target for the BDBEs range from 25 rem at 10^{-4}/plant-year to 750 rem at 5×10^{-7}/plant-year to ensure that the quantitative health objective (QHO) for early health effects is not exceeded for individual BDBEs.

The licensing basis development process used by GA-EMS for the FMR project will follow the process described in NEI 18-04 and endorsed by the NRC in Reg. Guide 1.233. At this conceptual design stage, the process is at an early stage and awaits development of a plant-specific PRA during the preliminary design stage.

5. PROCESS FOR SELECTING AND CLASSIFYING SSC

5.1. Safety Classification Categories

Safety classification categories are defined as follows:

- Safety-Related (SR):
    - SSCs selected by the designer from the SSCs that are available to perform the required safety functions (RSFs) to mitigate the consequences of DBEs to within the LBE F-C Target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions
    - SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of DBBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target
- Non-Safety-Related with Special Treatment (NSRST):
    - Non-safety-related SSCs relied upon to perform risk-significant functions. Risk-significant SSCs are those that perform functions that prevent or mitigate and LBE

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from exceeding the F-C Target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs

  - Non-safety-related SSCs relied on to perform functions requiring special treatment for DID adequacy

- Non-Safety-Related with No Special Treatment (NST):
  - All other SSCs (with no special treatment required)

Safety-significant SSCs include all those SSCs classified as SR or NSRST. None of the NST SSCs are classified as safety significant.

5.2. SSC Safety Classification Process

The SSC safety classification process is depicted in Figure 3. The process is closely aligned with the process for selecting and evaluating LBEs. The SSC safety classification process uses six tasks that are described below. It is important to recognize that a given SSC may perform other functions that are not relevant to the safety functions that prevent or mitigate accidents represented in the LBEs.

Task 1: Identify SSC Functions in the Prevention and Mitigation of LBEs

The event sequences identified as LBEs rely on SSCs in the plant to perform a function to prevent or mitigate a release of radioactive material. For internal events caused by an equipment failure, the initiating event frequency is related to the unreliability of the SSC. For SSCs that mitigate the consequences of the initiating event, their performance capabilities are the focus of the safety classification process. For those SSCs that fail to respond to the initiating event, their failure probabilities reduce the frequency of the event sequence and establish reliability requirements for those SSCs. The output of this task is the identification of the SSC prevention and mitigation functions for all the LBEs.

Task 2: Identify and Evaluate SSC Capabilities and Programs to Support Defense-in-Depth

The purpose of this task is to provide a feedback loop from the evaluation of DID adequacy. A result of this evaluation is the identification of SSC functions and the associated SSC reliabilities and capabilities that are deemed necessary for DID adequacy. Such SSCs and their associated functions are regarded as safety-significant, and this information is used to inform the SSC safety classification is subsequent tasks.

Task 3: Determine the Required and Significant Safety Functions

RSFs are those necessary to meet the F-C Target for all DBEs and high-consequence BDBEs and to conservatively ensure that 10 CFR 50.34 dose requirements can be met. RSFs for high-consequence BDBEs are responsible for preventing them from increasing in frequency into the DBE region and outside the F-C Target by having sufficient reliability to keep the frequency low and in the BDBE range.

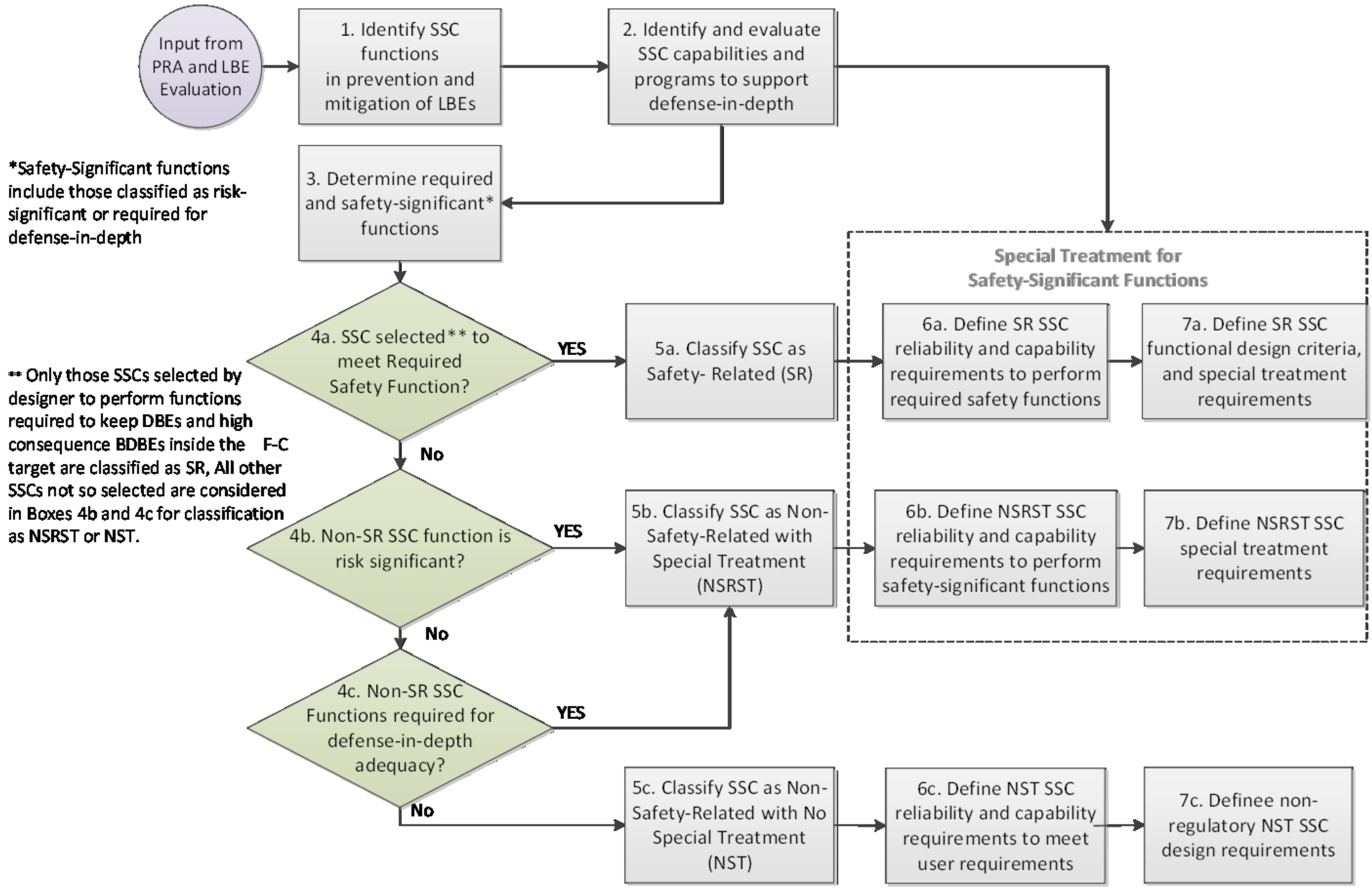

Figure 3 SSC Safety Classification Process

For each RSF, a decision is made on which SSC is selected to perform the RSF among those found to be available. Each DBE is protected by a set of SR SSCs. Other SSC functions may be classified as risk-significant or required for DID which would require reliability and capability requirements. Ideally, the SSC safety classification process uses input from a plant-specific PRA. At this early stage of conceptual design, historical PRAs and risk insights are used in place of a plant-specific PRA.

Safety significant SSCs include those that perform risk significant functions and those that perform functions that are necessary to meet defense-in-depth criteria. These safety significant SSCs

perform other functions that are not RSF. How individual SSC safety functions are classified relative to these function categories is resolved in Task 4.

Tasks 4 and 5: Evaluate and Classify SSC Functions

The purpose of Tasks 4 and 5 is to classify the SSC functions into one of three safety categories: Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST), and Non-Safety-Related with No Special Treatment (NST).

In Task 4A, each of the DBEs and any high consequence BDBEs are examined to determine which SSCs are available to perform the RSFs. The designer then selects one specific combination of available SSCs to perform each RSF that covers all the DBEs and high consequence BDBEs. These specific SSCs are classified as SR in Task 5A and are the only ones credited in the Chapter 15 safety analysis of the DBAs. All of the remaining SSCs are processed into one of the remaining two safety categories per Tasks 4B and 4C.

All SR SSCs are regarded as risk significant. However, it is also possible that some non-SR SSCs will meet the criteria for risk significance. In Task 4B, each non-safety related SSC is evaluated for its risk significance. A risk significant SSC function is one that is necessary to keep one or more LBEs within the F-C Target or is significant in relation to one of the cumulative risk metric limits. If the SSC is classified as risk significant and is not an SR SSC, it is classified as NSRST in Task 5B.

In Task 4C, a determination is made as to whether any of the remaining non-safety-related and non-risk significant SSC functions should be classified as requiring special treatment in order to meet criteria for defense-in-depth adequacy. Those that meet these criteria are classified as NSRST in Task 5B and the remaining SSCs are classified as NST in Task 5C.

All SSC functions classified as either SR or NSRST are regarded as safety significant. All non-safety significant SSC functions are classified as NST.

Task 6: SSC Reliability and Capability Requirements

The purpose of this task is to define the requirements for reliabilities and capabilities for SSC’s used in the evaluation of LBE frequence and consequences. For SSCs classified as SR or

NSRST, these requirements are used to develop regulatory design and special treatment requirements in Task 7. For those SSCs classified as NST, the requirements are part of the user design requirements.

For SSCs classified as SR, Functional Design Criteria (FDC) and lower-level design criteria are defined to capture design specific criteria which may be derived from the FMR Principal Design Criteria (PDC). These criteria are used to frame specific design requirements as well as special treatment requirements for SR classified SSCs. NSRST SSCs are not directly associated with FDC but are subject to special treatment as determined by the integrated decision-making process for evaluation of defense-in-depth.

Task 7: Determine SSC Specific Design and Special Treatment Requirements

The purpose of this task is to establish the specific design requirements for SSCs which include FDC for SR classified SSCs, regulatory design and special treatment requirements for each of the safety significant SSCs classified as SR or NSRST, and user design requirements for NST classified SSCs. For SSCs classified as SR, the design criteria are referred to as Safety-Related Design Criteria (SRDC). These are derived from the Required Functional Design Criteria (RFDC) that are in turn developed from the RSFs. RFDCs are defined to capture design-specific criteria that may be used in the formulation of PDC. After SR SSCs have been selected to perform the RSFs, the SRDCs are defined at the SSC level to assure meeting the RFDCs and the RSFs for the specified SSCs.

NSRST SSCs are not directly associated with RFDC but are subject to special treatment as determined by the integrated decision-making process for evaluation of DID adequacy. The term "special treatment" is used in a manner consistent with NRC regulations and NEI guidelines in the implementation of 10 CFR 50.69. Table 1 summarizes the types of special treatments considered in the formulation of requirements for each SSC category. Guidance on special treatment can be found in NRC documents like the Standard Review Plan (SRP) (Reference), NRC Reg. Guides, and industry documents from NEI and the American Society of Mechanical Engineers (ASME).

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Table 1 Summary of Special Treatments for SSCs

Special Treatment Category	Applicability[1] SR	NSRST	NST	Available Guidance[4]
Requirements Associated with SSC Safety Classification				
Document basis for SSC categorization by Integrated Decision-Making Process	√	√	√	Essentially the same as 10 CFR 50.69(c), Guidance in Reg. Guide 1.201, NEI-00-04 for all SSCs
Document evaluation of adequacy of special treatment to support SSC categorization	√			Essentially the same as 10 CFR 50.69(d), Guidance in Reg. Guide 1.201, NEI-00-04 for RISC-1 SSCs
		√		Essentially the same as 10 CFR 50.69(d), Guidance in Reg. Guide 1.201, NEI-00-04 for RISC-2 SSCs
Change control process to monitor performance and manage SSC categorization changes	√	√		Essentially the same as 10 CFR 50.69(e), Guidance in Reg. Guide 1.201, NEI-00-04 for RISC-1 and RISC-2 SSCs
Basic Requirements for all Safety-Significant SSCs				
Reliability Assurance Program including reliability and availability targets for SSCs in performance of Safety Functions	√	√		Essentially same as Reliability Assurance Program in SRP 17.4 for safety-significant SSCs, Guidance in SRP Chapter 19.3, ASME Section XI Reliability and Integrity Management Programs
Design Requirements for SSC capability to mitigate challenges reflected in LBEs	√	√		Guidance in NEI 18-04
Maintenance Program that assures target for SSC availability and effectiveness of maintenance to meet SSC reliability targets	√	√		Essentially same as 10 CFR 50.65 Maintenance Rule; link to Maintenance Rule consistent with 10 CFR 50.69 for RISC-1 and RISC-2 SSCs

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Special Treatment Category	Applicability[1] SR	NSRST	NST	Available Guidance[4]
Licensee Event Reports	√			Essentially the same as 10 CFR 50.69(f), Guidance in Reg. Guide 1.201, NEI-00-04 for RISC-1 and RISC-2 SSCs
10 CFR 60 Appendix B Quality Assurance (QA) Program	√			QA requirements consistent with 10 CFR 50 Appendix B should be risk-informed and performance-based and not compliance-based; guidance in SRF 17.5 QA for SR SSCs, 10 CFR 50.69, SRP 1.201
User provided QA Program for non-safety SSCs		√		QA requirements consistent with SRP 17.4 (Reliability Assurance Program) for non-safety-related, safety-significant SSCs should be risk-informed and performance-based and not compliance-based; guidance in SRF 17.5 QA for non-safety-related SSCs, 10 CFR 50.69, SRP 1.201
Additional Special Treatment Requirements				
Required Functional Design Criteria	√			Guidance in NEI 18-04, INL/EXT-14-31179
Technical Specifications	√	[2]		10 CFR 50.36, SRP
Seismic design basis	√	[3]	[3]	Essentially the same as for existing reactors for SR SSCs 10 CFR 100 Appendix A
Seismic qualification testing	√			Essentially the same as for existing reactors for SR SSCs 10 CFR 100 Appendix A, Reg. Guide 1.100
Protection against design basis external events	√			Essentially the same as for existing reactors for SR SSCs, Guidance in 10 CFR 100 Appendix A, SRP 3
Equipment qualification testing	√			Essentially the same as for existing reactors for SR SSCs 10 CFR 50.49
Materials surveillance testing	√			
Pre-service and risk-informed in-service inspections	√	[2]		See Reg. Guide 1.178

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Special Treatment Category	Applicability[1] SR	NSRST	NST	Available Guidance[4]
Pre-service and in-service testing	√	[2]		In-service testing needs to be integrated with Reliability Assurance Program

[1] The specific requirements for each applicable category should be evaluated on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. This is determined in the design and confirmed via an integrated decision-making process.

[2] The need for this special treatment for any NSRST is determined on a case-by-case basis, and when applicable, is applied to the specific functions to prevent and mitigate the applicable LBEs. This is determined via an integrated decision-making process.

[3] NSRST and NST SSCs are required to meet Seismic II/I requirements (required no to interfere with the performance of SR SSC RSFs following a safe shutdown earthquake.

[4] The references in this column are mostly applicable to LWRs, and hence they are offered as providing useful guidance.

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6. REQUIRED AND SAFETY SIGNIFICANT FUNCTIONS

Required safety functions (RSFs) are those necessary to meet the F-C Target for all DBEs and high-consequence BDBEs and to conservatively ensure that 10 CFR 50.34 dose requirements can be met. RSFs for high-consequence BDBEs are responsible for preventing them from increasing in frequency into the DBE region and outside the F-C Target by having sufficient reliability to keep the frequency low and in the BDBE range. Safety significant functions include those that perform risk significant functions and those that perform functions that are necessary to meet DID criteria.

The RSFs for the FMR are similar to other gas-cooled reactors but are accomplished with SSCs designed specifically for the FMR. The RSFs for the FMR are as follows:

- Control Radionuclides in Fuel System
- Control Radionuclides in Containment
- Control Heat Generation (Reactivity)
- Control Heat Removal
- Control Chemical Attack
- Maintain Core and Reactor Vessel Geometry

Other safety-significant functions that reduce risk and contribute to DID adequacy include providing forced convection cooling, reducing occupational exposure, reducing frequency of reactor scrams, and minimizing other unplanned outages.

7. SSC CLASSIFICATION

Active, passive, and inherent safety features are classified as SR SSCs in order to accomplish RSFs. Controlling radionuclides within the fuel and containment are key to meeting F-C Targets. Maintaining helium within the helium pressure boundary is not an RSF needed to remove core heat because the safety of the FMR during depressurization accidents relies primarily on radiation heat transfer. Sufficient forced or natural convective cooling can be accomplished regardless of whether the helium pressure boundary is intact or breached. The preliminary set of SR SSCs are listed in Table 2. In some cases, the list only identifies the system or subsystem while in other cases it is broken down into components. Often times, a system or subsystem has both SR and non-SR components depending on the specific RSF and LBE.

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Table 2 FMR Safety-Related SSCs and Functions

SSC	Function
Reactor Fuel	Minimize release of radionuclides from the fuel to the helium pressure boundary
Reactor Structural Support	Maintain reactor core in a coolable geometry and maintain geometry for reactivity control insertion
Reactivity Control - Control Rods - Shutdown Rods - Rod Release Mechanism	Insert negative reactivity and rapidly shutdown the nuclear reactor on command from the Reactor Protection System
Vessel System - Reactor Vessel and Support Structure - Pressure Relief System	Radiate heat passively from the reactor vessel to the RVCS Prevent Vessel System over pressurization
Containment System - Steel Containment - Reinforced Concrete around Containment - Containment Isolation Valves in interfacing subsystems	Minimize release of radionuclides from the containment to the environment through leak-tight barrier and containment isolation Protect the steel containment from external events
Reactor Vessel Cooling System - RVCS Panels - RVCS Piping - RVCS Water Tanks and Relief Valves	Transfer radiative heat from the reactor vessel to the atmosphere passively without electrical power
Reactor Protection System	Initiate rapid reactor shutdown (scram), containment isolation, and isolation of water sources

Non-SR SSCs may perform safety significant functions that perform risk significant functions or that perform functions that are necessary to meet DID criteria. Such SSCs have special treatments applied to meet reliability and capability requirements and are classified as NSRST. The preliminary set of NSRST SSCs are listed in Table 3.

All other SSCs are NST at this time. The recuperator is NST since it is not part of the helium pressure boundary and any leaks or failures between the high- and low-pressure sides of the recuperator has no safety significance but only effect plant efficiency. A containment heat and fission product removal system is currently not necessary. The RVCS can remove heat from the containment and natural processes within the containment are sufficient to eventually remove iodine and particulate fission products.

Table 3 FMR Non-Safety-Related with Special Treatment SSCs and Functions

SSC	Function
Reactivity Control - Rod Drive Mechanism	Regulate negative reactivity to adjust reactor power
Vessel System	Maintain integrity of helium pressure boundary
Precooler and Intercooler	Maintain integrity of helium pressure boundary and minimize frequency of water ingress
Reactor Vessel Cooling System - RVCS Heat Cooling System for water tank	Transfer heat from the RVCS water tank to the atmosphere using active forced convection cooling
Maintenance Cooling System	Minimize frequency of loss of forced cooling events
Power Conversion System	Minimize frequency of loss of forced cooling events
Helium Service System - Helium Storage - Helium Purification	Regulate helium inventory and minimize impurities during startup, shutdown, and normal operation
Plant Protection System	Initiate controlled reactor shutdown, forced cooling, and plant recovery
Fuel Handling - Fuel Handling Machine - Fuel Transfer Cask	Minimize radionuclide release and direct radiation exposure during refueling
Spent Fuel Storage	Minimize radionuclide release, direct radiation exposure, and criticality during spent fuel storage
Backup AC Power	Minimize frequency of loss of forced cooling events

8. SUMMARY AND CONCLUSIONS

The SSC safety classification approach is consistent with the LMP as described in NEI 18-04 and associated reference documents. The SSC safety classification approach has the following attributes:

- Systematic and Reproducible
- Reasonably Complete
- Provides Timely Input to Design Decisions
- Risk-Informed and Performance-Based

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- Consistent with Applicable Regulatory Requirements

The SSC safety classification presented in Section 4 is based on historical precedents from gas-cooled reactor design and licensing efforts. At this early stage of the conceptual design, a plant-specific PRA has not been started but insights from past PRA efforts have been applied. The approach is designed to ensure that an appropriate set of LBEs and RSFs are reflected in the FMR SSC safety classifications. This is essential to ensure that risk insights are appropriately reflected in the design and licensing decisions. Ultimately, SSC classification will be risk-informed consistent with LMP and NEI 18-04 once a full-scope PRA is completed.

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