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Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Safety Classification and Performance Criteria for Structures, Systems, and Components

Wayne L Moe, Amir Afzali

March 2020

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Southern Company

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Abstract

This report, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Safety Classification and Performance Criteria for Structures, Systems and Components,” represents a key element in the development of a framework for the efficient licensing of advanced non-Light Water Reactors (non-LWRs). It is the result of a Licensing Modernization Project (LMP) led by Southern Company and cost-shared by the United States Department of Energy (DOE).

A companion report described the LMP approach to selecting and evaluating licensing basis events and for identifying structures, systems, and components (SSCs) that are classified as Safety-Related. The current document outlines the approach to performing the following additional tasks that impact the development SSC design performance targets:

- Complete the process of SSC safety classification by subdividing non-safety-related SSCs into subcategories to identify additional safety-significant SSCs that require special treatment
- Describe the LMP approach to the definition of risk significant SSCs
- Describe the LMP approach for defining safety-significant SSCs in terms of their risk significance and role in supporting defense-in-depth (DID) adequacy
- Provide guidance for the development of special treatment requirements, functional design criteria, and performance targets for the reliability and capability of safety-significant SSCs in the prevention and mitigation of Licensing Basis Events (LBEs)

In addition, a series of companion reports are under development as part of the LMP that are intended to support the LMP framework necessary to support the above tasks. These include reports on:

- LMP probabilistic risk assessment (PRA) approach
- LMP approach to selecting and evaluation of LBEs
- Risk-informed and performance-based evaluation of DID adequacy

The focus of this report is to identify potential technical issues related to the safety classification of SSCs and the derivation of design targets necessary to support SSC performance of PRA safety functions[*] in the prevention and mitigation of LBEs for advanced non-LWR technologies. Included in this report are design targets that are needed for the SSC to perform necessary mitigation and prevention functions. Targets for the SSC mitigation functions serve to support the capability to limit the consequences of LBEs to acceptable levels. Targets for the reliability of the SSCs serve to prevent other LBEs with unacceptable safety consequences. The LMP

[*] As used in this report, the term “PRA safety function” is defined as any function modeled in the PRA that protects one or more radionuclide barriers, or otherwise prevents or mitigates a release of radioactive material to the environment. Those safety functions that are relied on in the design to prevent or mitigate a design basis accident within 10 CFR 50.34 dose limits are referred to as required safety functions.

approach for classification and treatment of SSCs described in this report builds on the approach proposed for the Next Generation Nuclear Plant project, which in turn benefitted from earlier efforts for the Exelon Pebble Bed Modular Reactor and General Atomics' MHTGR high temperature gas reactor projects. The approach utilizes relevant aspects of risk-informed SSC classification methods that have been developed for existing and advanced LWRs and small modular reactors including those defined for implementation of 10 Code of Regulations (CFR) 50.69. In contrast to 10 CFR 50.59, which uses a risk-informed process to "back fit" risk insights into a deterministic safety classification process for operating LWRs, the LMP approach is a "forward fit" technology-inclusive, risk-informed, and performance-based (TI-RIPB) process to establish the initial SSC safety classification for advanced non-LWRs.

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List of Abbreviations

ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO*	Anticipated Operational Occurrence
ARDC	Advanced Reactor Design Criteria
ASME	American Society of Mechanical Engineers
BDBE*	Beyond Design Basis Event
CFR	Code of Federal Regulations
DBA*	Design Basis Accident
DBE*	Design Basis Event
DBEHL*	Design Basis External Hazard Level
DID	defense-in-depth
DOE	Department of Energy
EAB	Exclusion Area Boundary
ECA	Energy Conversion Area
ES*	Event Sequence
F-C*	Frequency-Consequence
F-C Target*	Frequency-Consequence Target
FSF	Fundamental Safety Function
GDC	General Design Criteria
GW(e)	Gigawatt electric
HPB	Helium Pressure Boundary
HPS	Helium Purification System
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IDP	Integrated Decision-Making Process
IE*	Initiating Event
ISI	In-Service Inspection
LBE*	Licensing Basis Event
LMP	Licensing Modernization Project
LWR	light water reactor
mHTGR	Modular High Temperature Gas-Cooled Reactor
MHTGR	a specific mHTGR designed by General Atomics
MST	Mechanistic Source Term
NDE	Non-Destructive Examination
NGNP	Next Generation Nuclear Plant
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 Guideline
PB	performance-based
PBMR	Pebble Bed Modular Reactor
POS	plant operating state
PRA	probabilistic risk assessment
PSF*	PRA Safety Function
PSID	Preliminary Safety Information Document
QHO	Quantitative Health Objective
RAP	Reliability Assurance Program
RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RCPB	reactor coolant pressure boundary
rem	Roentgen equivalent man
RFDC*	Required Functional Design Criteria
RI	risk-informed
RIM	Reliability and Integrity Management
RIPB-DM	risk-informed and performance-based integrated decision-making
RPV	Reactor Pressure Vessel
RS	Reactor System
RSF*	Required Safety Function
RV	Reactor Vessel
SCS	Shutdown Cooling System
SCWS	Shutdown Cooling Water System
SFC	Single Failure Criterion
SR*	Safety-Related
SRDC*	Safety-Related Design Criteria
SRP	Standard Review Plan
SSC	Structures, Systems, and Components
TI*	Technology-Inclusive
TI-RIPB*	Technology-Inclusive, Risk-Informed, and Performance-Based

*These terms have special meanings defined in this document. See Glossary of Terms.

1.0 INTRODUCTION

1.1 Purpose

Many of the current regulatory requirements for US nuclear power plants are based on light water reactor (LWR) technology used for generation of electricity, necessitating changes to the LWR framework to facilitate efficient, effective, and predictable licensing expectations for a spectrum of novel, advanced, non-light water reactors (non-LWRs). The Licensing Modernization Project (LMP), led by Southern Company and cost-shared by the U.S. Department of Energy (DOE) and other industry participants, has proposed changes to specific elements of the current licensing framework and a process for implementation of the proposals. These proposals were originally described in a series of draft white papers that were reviewed by industry stakeholders and the Nuclear Regulatory Commission (NRC) staff. Based on these reviews, industry guidance was prepared and documented in NEI 18-04[1] gathering key aspects of the draft white papers while addressing review comments in a form suitable for future NRC endorsement in a regulatory guide.

The purpose of this document is to define the LMP approach to Structures, Systems, and Components (SSC) safety classification and the derivation of design targets necessary to support SSC performance of safety functions in the prevention and mitigation of Licensing Basis Events (LBEs). Such targets include those to provide the necessary capabilities to perform their mitigation functions and those to meet their reliability targets to prevent LBEs with more severe consequences.

The LMP approach for classification and treatment of SSCs described in this document builds on the approach proposed for the Next Generation Nuclear Plant (NGNP) project,[2] which in turn benefitted from earlier efforts for the Exelon Pebble Bed Modular Reactor (PBMR)[3] and General Atomics MHTGR[4] projects. Use is made of relevant aspects of risk-informed SSC classification approaches that have been developed for existing and advanced LWRs and small modular reactors, including those defined for implementation of 10 Code of Federal Regulations (CFR) 50.69.[10]

A draft of this report in the form of a draft white paper was submitted for NRC review in 2017 and the NRC staff comments from this review are documented in Reference [6]. The guidance document for implementing the LMP methodology in NEI 18-04 includes a discussion on the LMP approach to SSC safety classification that addresses the NRC comments on the draft SSC report and other LMP reports on probabilistic risk assessment (PRA) development, LBE selection and evaluation, and evaluation of defense-in-depth (DID) adequacy. This SSC safety classification report reflects the clarifications on the LMP approach to SSC safety classification identified in NEI 18-04 and makes use of the LMP approach to PRA development which is the topic of a companion report.[7] As discussed in the companion report on LBE selection and evaluation,[8] the classification of SSC is highly integrated with the LBE selection and evaluation process and the approach to evaluating DID adequacy. The LMP approach to evaluating DID adequacy is covered in Reference [9]. As discussed in these reports, the processes for PRA development, LBE selection and evaluation, SSC safety classification and performance targets, and evaluation of DID adequacy are highly integrated processes.

1.2 Objective of This Document

This document describes a technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach for the safety classification of SSCs and the derivation of design targets necessary to support SSC performance of safety functions in the prevention and mitigation of LBEs. The objectives of this document are to:

- Describe the approach for the safety classification of SSCs within the LMP framework
- Present an approach and criteria for determining the risk significance and safety significance of SSCs
- Discuss the roles of SSC reliability and capability in the prevention and mitigation of accidents
- Present a top-down process for developing Required Functional Design Criteria (RFDC) and lower level, Safety-Related Design Criteria (SRDC) for implementation of SSC Required Safety Functions (RSFs)
- Prescribe the process for the development of SSC special treatment requirements in performance of their functions in the prevention and mitigation of LBEs
- Identify by reference relevant supporting regulatory guidance, precedents, and available references to assist in implementing the LMP approach to SSC safety classification and performance criteria
- Identify potential technical issues associated with implementing the LMP approach to SSC safety classification
- Provide the necessary links to the LMP approaches for PRA development, LBE selection and evaluation, and evaluation of DID adequacy

1.3 Scope

The SSC classification approach described in this document applies to a spectrum of advanced non-LWR designs including modular high temperature gas reactors (mHTGRs), molten salt reactors, and liquid metal-cooled reactors and is intended to be reactor technology-inclusive (TI).

For each of these general classes of reactors there are a number of variants using different materials, different neutron spectra, and both heterogeneous and homogeneous fuels. The scope of SSCs includes the barriers to radionuclide release and any SSCs that perform a safety function to protect one or more barriers.

The LMP approach to SSC safety classification is described in Section 2 of this document. Although certain aspects of safety classification have already been discussed as part of the report on LBE selection and evaluation, this section provides a more complete definition of the SSC classification process. Included in this section is a TI-RIPB approach for defining the risk significance and safety significance of SSCs, whose concepts are used to define the LMP SSC safety classes.

The derivation of performance criteria for SSCs in the prevention and mitigation of LBEs is the topic of Section 3 of this document. The section begins by illustrating the process for selecting the design criteria and design requirements for safety-related SSCs, including those SSCs that provide a barrier function.* The section concludes with a definition of the special treatments for safety-significant SSCs, which includes those SSCs that perform risk significant functions as well as those that may require special treatment for DID adequacy.

1.4 Summary of LMP Approach to SSC Safety Classification

The LMP methodology includes the following SSC safety classification categories:

- Safety-Related (SR):
    - SSCs selected by the designer from the SSCs that are available to perform the RSFs to mitigate the consequences of design basis events (DBEs) to within the LBE Frequency-Consequence evaluation target (F-C Target), and to mitigate design basis accidents (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 Beyond Design Basis Events (BDBEs) 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 on to perform risk-significant functions. Risk significant SSCs are those that perform functions that prevent or mitigate any LBE 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. Numerical risk significance criteria used for this purpose are presented in this report.
    - 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 in the LMP methodology perform functions that are either risk significant or necessary for adequate DID and include all those SSCs classified as SR or NSRST. None of the NST SSCs are classified as safety-significant.

The purpose of having SSC safety classes is to guide the development of SSC performance targets and special treatment requirements. The LMP approach to developing these targets and requirements is linked to the evaluation of DID adequacy and is described in detail in this report

* In this report, the term "barrier" is used to denote any plant feature that is responsible to either full or partial reduction of the quantity of radionuclide material that may be released during an accident. It includes features such as physical barriers or any feature that is responsible for mitigating the quantity of material including time delays that permit radionuclide decay.

and in the companion report on evaluation of DID adequacy. The purpose of these targets and requirements is to provide reasonable confidence in the SSC capabilities and reliabilities in performing functions identified in the LBEs consistent with the F-C Target and the regulatory dose limits for DBAs.

1.5 Relationship to Other LMP Topics/Documents

The SSC safety classification approach described in this report is intended to be used in conjunction with other aspects of the LMP framework described in the following supporting reports.

The LMP team prepared independent reports on each of the four major LMP elements. Additionally, the LMP team produced a narrative report describing the processes, events, and documents involved in producing the ultimate project deliverable product, NEI 18-04 "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development." Finally, the LMP team produced a report based on the experiences of early adopters of the LMP RIPB process which includes best practices, lessons learned, and frequently asked questions with responses. See Table 1-1 for the Southern Company document numbers of each of these reports.

Table 1-1. LMP Reports and Document Numbers

Report Title	*Southern Company Document Number*	*DOE OSTI Document Number*
Selection and Evaluation of Licensing Basis Events	SC-29980-100 Rev 1	TBD
Probabilistic Risk Assessment Approach	SC-29980-101 Rev 1	TBD
Safety Classification and Performance Criteria for Structures, Systems, and Components	SC-29980-102 Rev 1	TBD
Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy	SC-29980-103 Rev 1	TBD
Final Project Report	SC-29980-105 Rev. 1	TBD
LMP Lessons Learned, Best Practices, and Frequently Asked Questions	SC-29980-106 Rev 0	TBD

PRA Approach

The PRA approach report[7] describes a technology-inclusive approach for developing a PRA for an advanced non-LWR to support the design and provide risk insights for the selection of LBEs, safety classification of SSCs, and risk-informed evaluation of DID.

Licensing Basis Event Selection and Evaluation Approach

Key inputs to the selection of LBEs are derived from a PRA evaluation of the advanced non-LWR plant. These inputs together with deterministic inputs are used as part of a TI-RIPB approach for the selection and evaluation of LBEs. As part of the LBE selection and evaluation process described in the LBE report,[8] the advanced non-LWR designer will select a set of safety-related SSCs that are necessary and sufficient to perform the safety functions required to mitigate all the DBAs within the 10 CFR 50.34 dose limits using conservative assumptions, and to prevent any high consequence BDBE from migrating into the DBE region and exceeding the F-C Target. In the current report, additional safety classes of SSCs are defined and an approach to developing SSC performance targets and special treatment requirements from these classes is described.

Defense-in-Depth Adequacy

The DID report[9] presents a TI-RIPB approach for defining DID and evaluating the adequacy of DID in the design capabilities and in the selection of programs to assure DID adequacy. It also describes how DID is taken into account in the risk-informed decisions to select LBEs, safety classification of SSCs, and selection of SSC performance targets, which is described in this report.

LMP Final Report

The LMP team produced a narrative report describing the processes, events, and documents involved in producing the ultimate project deliverable product, NEI 18-04 "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development." This report contains a wealth of references to documents that future users of the LMP RIPB process may find useful. Tables within the report provide references to the NRC Agencywide Document Management System (ADAMS) Accession Numbers of many industry and NRC documents that future permit and license applicants may wish to reference in their own applications.

LMP Lessons Learned, Best Practices, and Frequently Asked Questions

The LMP team produced a report based on the experiences of early adopters of the LMP RIPB process that includes best practices, lessons learned, and frequently asked questions with responses. This report provides guidance to reactor designers on how to efficiently implement the LMP RIPB processes within their own organizations and answers to 32 frequently asked questions from reactor designers.

2.0 LMP SSC SAFETY CLASSIFICATION APPROACH FOR ADVANCED NON-LWRS

2.1 SSC Safety Classification Attributes

The desirable attributes of the SSC safety classification and performance targets process for advanced non-LWRs, based on the objectives of the LMP, are described below.

Systematic and Reproducible

In principle, the application of the safety classification process by different individuals who are given the same inputs would yield a reasonably consistent SSC safety classification and selection of associated performance targets and special treatment requirements. Any variations should only result from different states of knowledge that are fed into the process.

Sufficiently Complete

The SSC safety classification and performance targets process should be based on a sufficiently complete set of LBEs that are defined using the process defined in the companion report on LBE selection and evaluation. The LBEs defined using this process are capable of defining the challenges to the SSCs responsible for performing safety functions, providing radionuclide barriers, and supporting protective strategies for emergency planning and accident management. The SSCs defined in these LBEs are responsible for preventing and mitigating accidents involving single and multiple reactor modules, when applicable, and from multiple radionuclide sources that may be involved in an accident.

Available for Timely Input to Design Decisions

Importantly, the SSC safety classification and performance targets process should recognize that design decisions that are impacted by the process are made at an early stage of design and long before the licensing application is prepared. A key limitation in the progress of deploying advanced reactor technologies is the lack a predictable regulatory and licensing framework for a non-LWR. The SSC safety classification and performance targets process that is provided in this report should play an important role in supporting the optimization of the design with respect to safety and lead to greater transparency and predictability in the licensing process as it relates to SSC safety classification for advanced non-LWRs.

Risk-Informed and Performance-Based

The SSC safety classification and performance targets process should be RIPB, consistent with LMP objectives. Risk-informed, as contrasted with risk-based, means that the process will include an appropriate balance of deterministic and probabilistic elements, and will be consistent with the principles of DID. The terms performance-based are used to mean that the process will include measurable and quantifiable performance metrics and will be consistent with NRC policies on the use of performance-based alternatives. The interfaces with other RIPB decisions such as LBE selection and evaluation and implementation of DID strategies should be clearly defined.

Reactor Technology-Inclusive

When applying the process to different advanced non-LWRs having fundamentally different safety design methods, the approach offered in this report should yield appropriate SSC safety classification and performance targets that are consistent and clearly defined across the different reactor technologies. This process should be capable of addressing the unique safety issues for each non-LWR reactor technology. Specifically, the approach needs to support a consistent SSC safety classification and performance targets process for gas reactors, molten salt reactors, and liquid-metal-cooled reactors using both thermal and fast neutron spectra and employing a variety of safety design approaches.

Consistent with Applicable Regulatory Requirements

The SSC safety classification and performance targets process must account for current regulatory requirements with due regard to their applicability to advanced non-LWR technologies as well as associated safety design approaches. While future rule-making for advanced non-LWRs may be desirable, the goal is to define an acceptable approach for SSC safety classification that may be used within the intent of existing requirements.

2.2 Summary of SSC Safety Classification Approach

The LMP SSC safety classification[*] process is described in Figure 2-1. This process is designed to be used with the LMP process for selecting and evaluating LBEs as shown in Figure 2-2, which was introduced in the LMP LBE report. The tasks in the LBE process that are expected to receive the greatest regulatory involvement are identified in this figure. The information needed to support the SSC safety classification is available when Task 10 of the LBE selection and evaluation process in Figure 2-2 is completed in each phase of the design process. It is noted that SSCs are not classified in a vacuum but rather in the context of performing specific prevention and mitigation functions identified in the LBEs. Hence, this is an SSCs are classified in the context of the functions they perform in the prevention and mitigation of LBEs.

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[*] The SSC safety classification process classifies SSCs on the basis of the SSC safety functions reflected in the LBEs. Although the SSCs are classified, the resulting performance and special treatment requirements are for the specific functions identified in the LBEs.

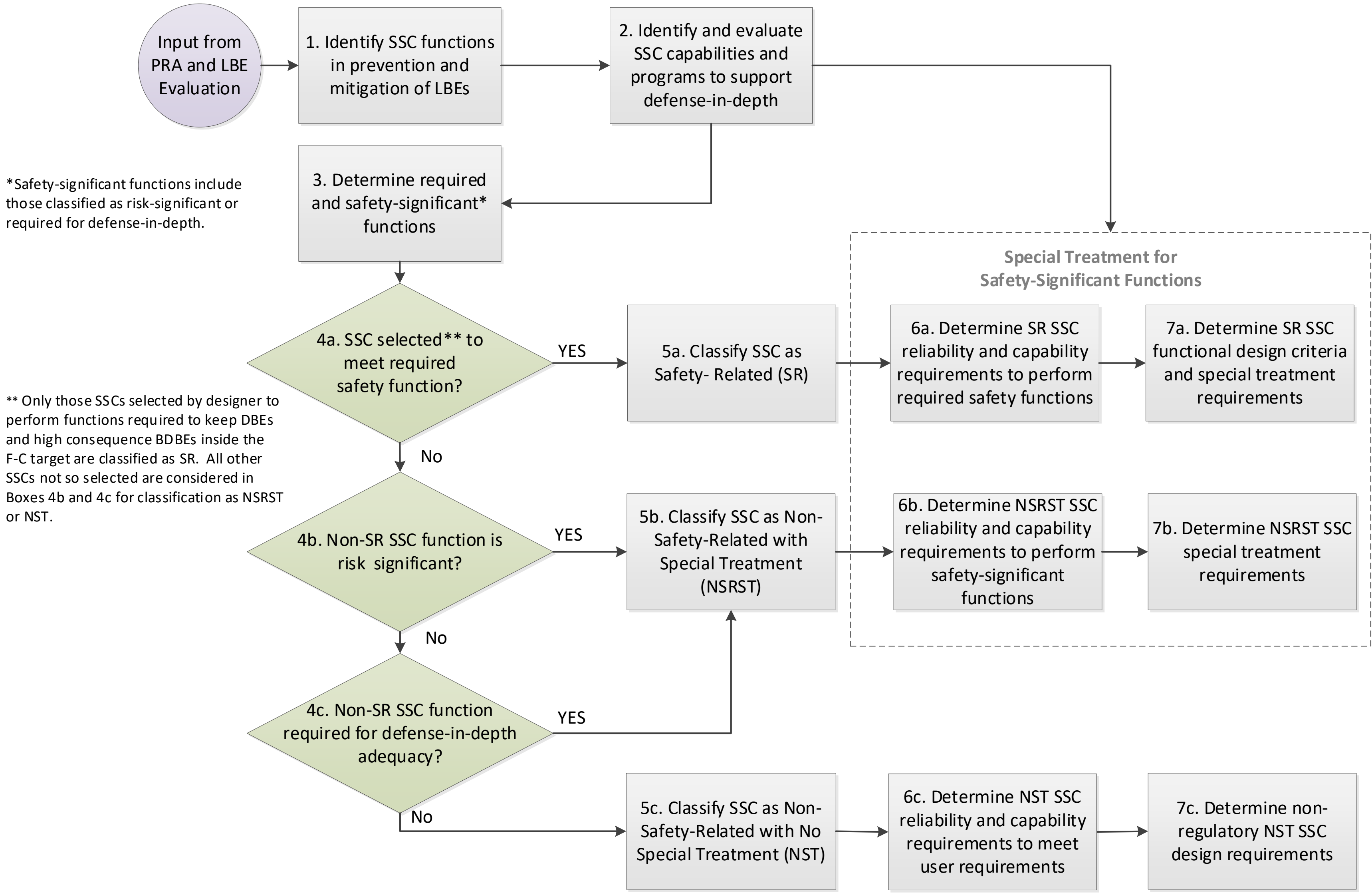

Figure 2-1. LMP SSC Function Safety Classification Process

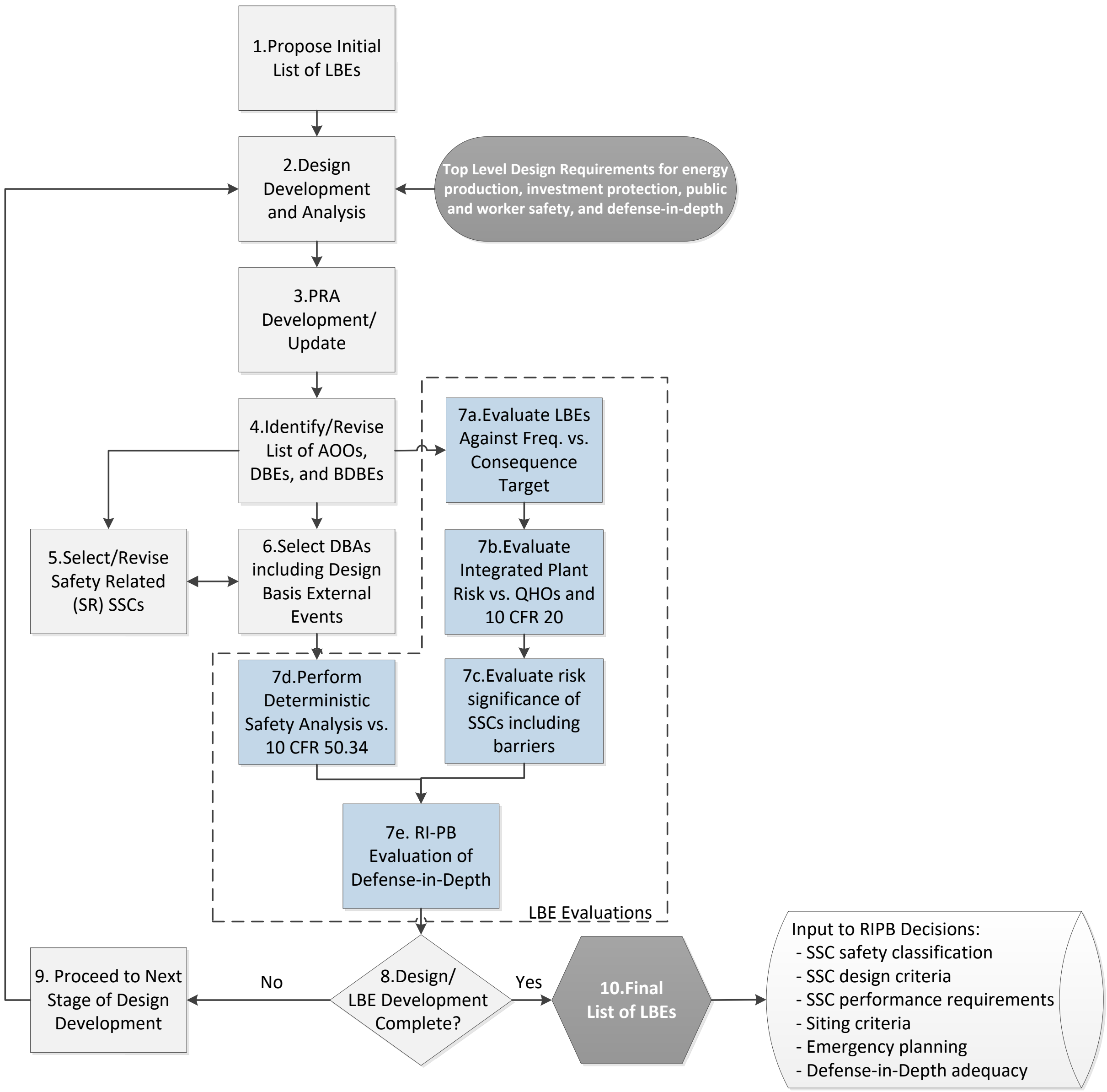

Figure 2-2. LMP Process for Selecting and Evaluating LBEs

The SSC safety classification process in Figure 2-1 is implemented in the seven tasks that are described below. This process is described as an SSC function classification process rather than an SSC classification process because only those SSC functions that prevent or mitigate accidents represented in the LBEs are of concern. A given SSC may perform other functions that are not relevant to LBE prevention or mitigation or functions with a different safety classification.

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

The purpose of this task is to review each of the LBEs, including, Anticipated Operational Occurrences (AOOs), DBEs, and BDBEs to determine the function of each SSC in the prevention and mitigation of the LBE. Each LBE is comprised of an Initiating Event (IE), a sequence of conditioning events, and end state. The IEs may be associated with an internal event such as an SSC failure or human error, an internal plant hazard such as a fire or flood, or an external hazard such as seismic event or external flood.

For those IEs caused by an equipment failure, the initiating event frequency is related to the unreliability of the SSCs, i.e., SSCs with higher reliability serve to prevent the IE. Thus, higher levels of reliability result in a lower frequency of IEs. For SSCs that successfully mitigate the consequences of the IE, their capabilities and safety margins to respond to the IE are the focus of the safety classification process and resulting special treatment. For those SSCs that fail to respond along the LBE event sequences, their reliabilities, which serve to prevent the LBE by reducing its frequency, are the focus of the reliability targets derived from classification and treatment process. 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, which is the topic of a separate LMP report. This evaluation includes an examination of the plant LBEs, identification of the SSCs responsible for the prevention and mitigation of accidents, and a set of criteria to evaluate the adequacy of DID. A result of this evaluation is the identification of SSC functions, and the associated SSC reliabilities and capabilities that are deemed to be 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 in subsequent tasks.

Task 3: Determine Required and Safety-Significant Functions

The purpose of this task is to define the safety functions that are required to meet the F-C Target for all the DBEs and the high consequence BDBEs, referred to as Required Safety Functions, as well as other safety functions of SSCs regarded as safety-significant. Safety-significant SSCs include those that perform risk significant functions and those that perform functions that are necessary to meet DID criteria. As explained more fully in the LMP PRA report, the scope of the PRA includes the plant SSCs that are responsible for preventing or mitigating the release of radioactive material. This broader category of functions is referred to as PRA Safety Functions (PSFs). Hence, the LBEs derived from the PRA include all the relevant SSC prevention and mitigation functions embodied within the PSFs.

As explained previously, there are some safety functions classified as RSFs that must be fulfilled to meet the F-C Target for the DBEs using realistic assumptions and dose requirements for the DBAs using conservative assumptions. In addition to these RSFs, there are additional functions that are classified as safety-significant when certain criteria regarding risk significance and DID adequacy are met, as explained below. In most cases, there are several combinations of SSCs that can perform these RSFs. How individual SSC safety functions are classified relative to these function categories is resolved in Task 4 and Task 5. Figure 2-3 illustrates the concepts used to classify SSC safety functions as risk-significant and safety-significant. As shown in this figure, the SSCs modeled in the PRA are limited to those that perform a PSF that prevents or mitigates a release from a radionuclide source within the scope of the PRA. A subset of the PRA modeled SSCs are classified as safety-significant when they are necessary for adequate DID or meet LMP risk significance criteria for SSCs. While SR SSCs may or may not meet the risk significance criteria, because they are the primary means of fulfilling the RSFs they are an element of DID adequacy. Further definition of these terms is provided in Section 2.4.

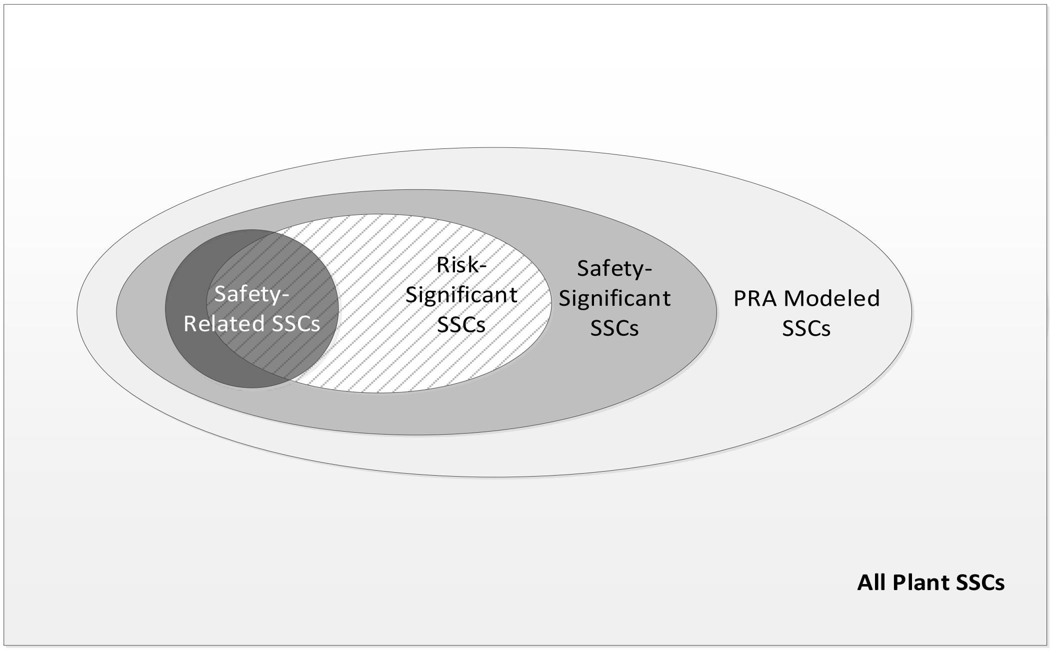

Figure 2-3. Definition of Risk-Significant and Safety-Significant SSCs

Tasks 4 and 5: Evaluate and Classify SSC Functions

The purpose of Tasks 4 and 5 is to classify the SSC functions modeled in the PRA into one of three safety categories: SR, NSRST, and NST.

Tasks 4A and 5A**:** In Task 4A, each of the DBEs and any high consequence BDBEs (i.e., those with doses above 10 CFR 50.34 limits) 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 included in the Chapter 15 safety analysis of the DBAs. All the remaining SSCs are processed further in Tasks 4b and 4c. All SR SSCs are also classified as safety-significant. SR SSCs may or may not be risk significant, depending on whether the risk significance criteria are met or not, but SR SSCs are always a necessary element of DID in the LMP methodology. An example of how SR SSCs were derived for the MHTGR is found in Appendix A. Additional examples of how SR SSCs are defined for the MHTGR and General Electric Power Reactor Innovative Small Module designs are found in the LMP LBE report.

Tasks 4B and 5B: Because each SR classified SSC identified in Task 4A is necessary to keep one or more LBEs inside the F-C Target, all SR SSCs are regarded in the LMP framework as risk significant. However, it is also possible that some non-SR SSCs will meet the LMP criteria for risk significance. In this task, 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 LBE cumulative evaluation risk metric limits that are defined in the LMP LBE report to evaluate the risk significance of LBEs. Examples of the former category are SSCs needed to keep the consequences below the AOO limits in the F-C Target, and DBEs where the reliability of the SSCs must be controlled to prevent an increase of frequency into the AOO region with consequences greater than the F-C Target. The SSC and LBE risk significance criteria are discussed in more detail in Section 2.4. If the SSC is classified as risk significant and is not an SR SSC, it is classified as NSRST in Task 5b. SSC functions that are neither safety-related nor risk significant are evaluated further in Task 4c.

Tasks 4C and 5C: In this task, 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 DID adequacy. The criteria for DID adequacy are discussed in Section 2.7 and in more detail in the companion LMP DID report. Those that meet these criteria are classified as NSRST in Task 5b and those remaining as NST in Task 5c.

At the end of this task, all SSC functions reflected in the LBEs will be placed in one of the three SSC function safety classes illustrated in Figure 2-4.

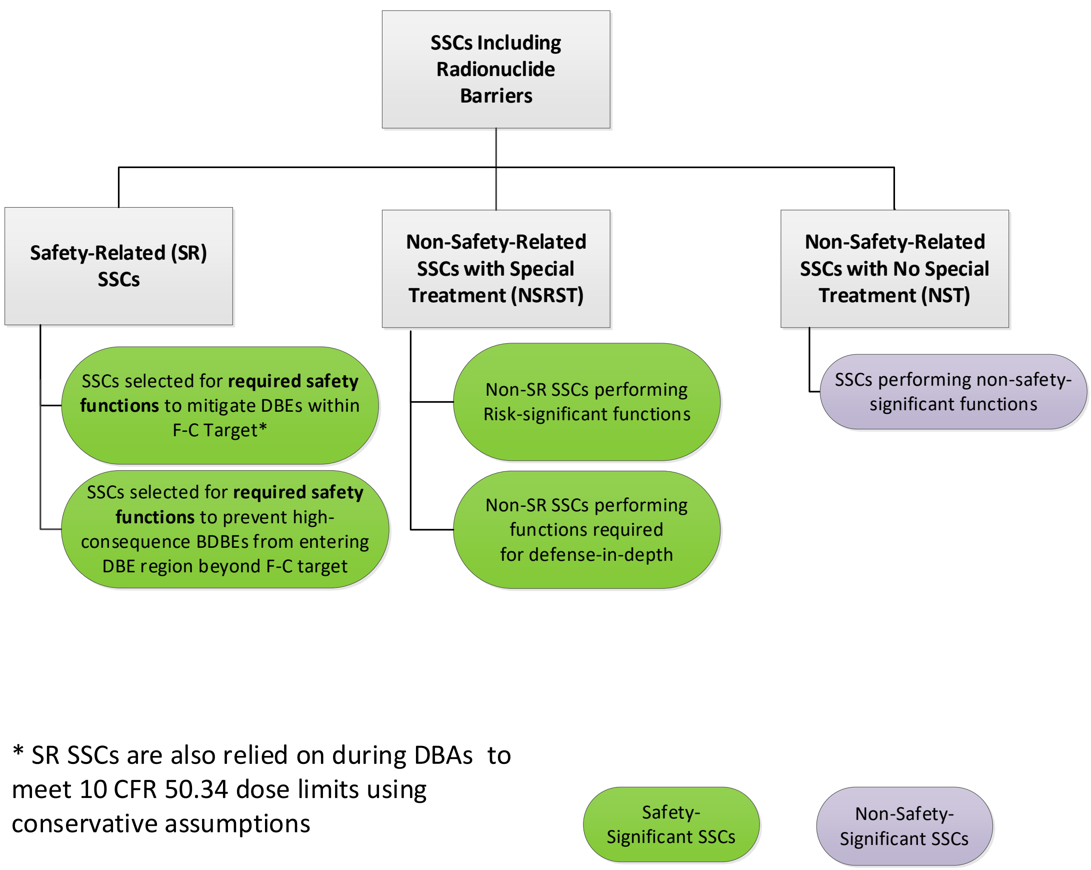

Figure 2-4. LMP SSC Safety Categories and Safety-Significant SSCs

Note that all SSC functions classified as either SR or NSRST are regarded as "safety-significant." All non-safety-significant SSC functions are classified in NST. Further discussion of how the LMP approach uses the concept of safety significance is found in Section 2.4.

The three SSC safety categories in Figure 2-4 have the same names as those developed in the NGNP and Exelon PBMR approaches, although the logic in deriving them is somewhat different. The LMP approach makes use of the concept of SSC safety significance that is associated with the 10 CFR 50.69[10] approach and also addresses the possibility that an SSC that is not safety-related nor risk-significant may be classified as safety-significant based on DID considerations. The LMP approach to assigning risk significance uses the concept of evaluating the impact of the SSC function on the ability to meet the F-C Target, as in the previous approaches, but also includes criteria based on risk significance metrics for the cumulative risk impacts of SSC functions across all the LBEs. Hence the LMP approach is in better alignment with the risk-informed safety classification process that is being implemented for 10 CFR 50.69.

Task 6: SSC Reliability and Capability Targets

For each of the SSC functions that have been classified in Task 5, the purpose of this task is to define the targets for reliabilities and capabilities for SSCs modeled in the PRA. For SSCs classified as SR or NSRST, which together represent the safety-significant SSCs, these targets are used to develop regulatory design and special treatment requirements in Task 7. For those SSCs classified as NST, the reliability and capability targets are part of the non-regulatory user design targets.

In order to meet the risk targets (F-C Target and cumulative risk targets), SSCs that are relied upon will need to meet defined reliability performance targets and will need to demonstrate DID adequacy. Strategies to achieve design reliability targets include use of passive design features and inherent capabilities, redundancy, diversity, and defenses against common-cause failures. Programmatic actions would be used to maintain performance within the design reliability targets.

Task 7: Determine SSC Specific Design and Special Treatment Requirements

The purpose of this task is to establish the specific design targets for SSCs which include RFDC 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 targets for NST classified SSCs. As explained more fully in Section 3 of this report, the specific SSC targets are tied to the SSC functions reflected in the LBEs and are determined utilizing the same integrated decision-making process used for evaluating the adequacy of DID. The links between the SSC functions and the LBEs are more clearly identified in the LBE selection and evaluation chart in Figure 2-2.

For SSCs classified as SR, RFDC and design criteria specific to SR SSCs, referred to as Safety-Related Design Criteria (SRDC), are identified in the LMP methodology as design specific requirements. These LMP derived requirements may be considered together with generic applicable Advanced Reactor Design Criteria (ARDC)[17] in formulating the principal design criteria for the license application. It is noted, however, that when considering the use of generic ARDC for this purpose, the LMP methodology does not include the application of the Single Failure Criterion (SFC) that is included in the ARDC language. In the LMP approach to formulating design requirements for SSCS, reliability and capability targets are used to inform the selection of special treatment requirements. This obviates the need to applying the SFC. Hence when ARDCs are included as part of the principal design criteria, the SFC language should be removed.

Examples of RFDC and SRDC that were developed for the MHTGR are discussed in Section 3 and in Appendix A. These criteria are used to frame specific performance targets as well as special treatment requirements for SR classified SSCs. NSRST SSCs are not directly associated with RFDC or SRDC but are subject to performance targets for reliability and capability as determined by the integrated decision-making process for evaluation of DID adequacy as set in Step 6. Guidance on the development of RFDC, SRDC, design targets, and special treatment requirements is found in Section 3 of this report using examples developed previously for the MHTGR as discussed in detail in the appendix.

The RFDC, SRDC, the reliability and capability targets for SR and NSRST SSCs, and special treatment requirements for SR and NSRST SSCs define safety-significant aspects of the descriptions of SSCs that are key aspects of the safety case.

The term "special treatment" is used in a manner consistent with NRC regulations and NEI guidelines in the implementation of 10 CFR 50.69. In Regulatory Guide 1.201,[11] the following definition of special treatment is provided:

> *"…special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions."*

In RIEP-NEI-16,[5] a distinction is made between special treatment as applied to safety-related SSCs and alternative special treatment afforded by 10 CFR 50.69. Alternative treatment requirements are differentiated from special treatment requirements in the use of "reasonable confidence" versus "reasonable assurance." More details on the development of specific SSC design and performance targets are provided in Section 3 of this report.

2.3 Comparison of LMP Approach to 10 CFR 50.69 Safety-Significance Categories

There are similarities between LMP SSC safety categories and the safety-significance categories associated with 10 CFR 50.69; however, there are several key differences as well. The four safety-significance categories associated with 10 CFR 50.69 are illustrated in Figure 2-5 in comparison with the three LMP SSC safety categories. The major difference stems from the fact that 10 CFR 50.69 is a back-fit approach to incorporating risk insights in safety classification that was originally based on a deterministic approach to defining safety-related SSCs. By contrast, the derivation of safety-related SSCs in the LMP framework is based on a forward-fit risk-informed approach that has the risk insights "baked in." In addition, all of the LMP safety-related SSCs are regarded as safety-significant in the LMP framework. SR SSCs may or may not be risk significant, depending on whether the risk significance criteria are met or not, but SR SSCs are always a necessary element of DID in the LMP methodology. As a result of this approach there is no LMP equivalent to RISC-3 in the 10 CFR 50.69 framework.

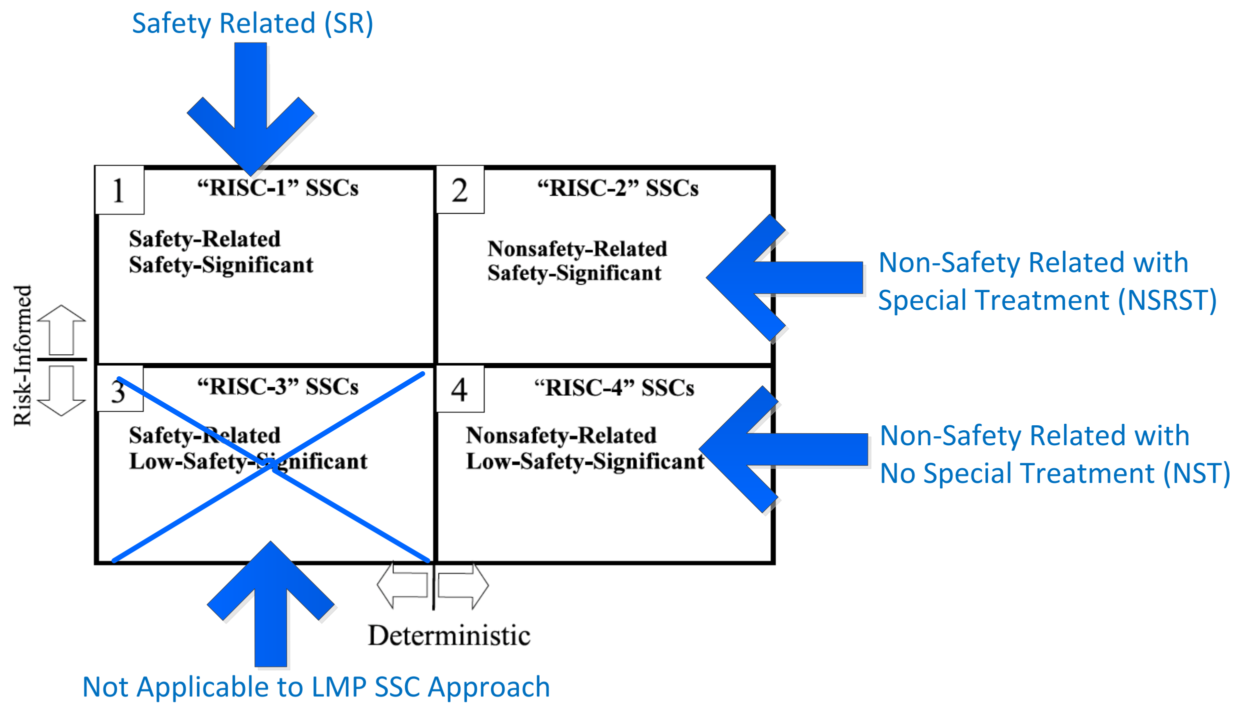

Figure 2-5. Comparison of LMP and 10 CFR 50.69 Risk-Informed Safety Categories[11]

A comparison of the respective safety categories within LMP and 10 CFR 50.69 is found in Table 2-1.

Table 2-1. Comparison of LMP and 10 CFR 50.69 SSC Safety Categories

10 CFR 50.69 SSC Category	Summary of Requirements Per RG 1.201	Applicability to LMP Safety Classification
RISC-1 Safety-Related Safety-Significant	SSCs are safety-related SSCs that the risk-informed categorization process determines to be significant contributors to plant safety. Licensees must continue to ensure that RISC-1 SSCs perform their safety-significant functions consistent with the categorization process, including those safety-significant functions that go beyond the functions defined as safety-related for which credit is taken in the categorization process.	Similar to the SR Category except that the derivation of safety-related SSCs is via a risk-informed process and the concept of safety significance is defined somewhat differently. In the LMP framework a safety-significant SSC is defined as one performing a risk significant function or an element of the plant capabilities for DID requiring special treatment. In both cases, safety significance is determined through an integrated decision-making process which incorporates elements of both probabilistic risk as well as traditional engineering insights. Risk significance is defined differently in the respective approaches because of the use of different risk metrics. In the LMP framework, all SR SSCs are risk significant having been developed through a risk-informed process that requires the identified SSCs to be safety-related to maintain LBE risk levels within the F-C Target.
RISC-2 Non-Safety-Related Safety-Significant	RISC-2 SSCs are those that are defined as non-safety-related, although the risk-informed categorization process determines that they are significant contributors to plant safety on an individual basis. The NRC staff recognizes that some RISC-2 SSCs may not have existing special treatment requirements. As a result, the focus for RISC-2 SSCs is on the safety-significant functions for which credit is taken in the categorization process.	Similar to NSRST Category using a somewhat different definition of safety-significant.

10 CFR 50.69 SSC Category	Summary of Requirements Per RG 1.201	Applicability to LMP Safety Classification
RISC-3 Safety-Related Low Safety-Significant	RISC-3 SSCs are those that are defined as safety-related, although the risk-informed categorization process determines that they are not significant contributors to plant safety. Special treatment requirements are removed for RISC-3 SSCs and replaced with high-level requirements. These high-level requirements are intended to provide sufficient regulatory treatment, such that these SSCs are still expected to perform their safety-related functions under design-basis conditions, albeit at a reduced level of assurance compared to the current special treatment requirements. However, Section 50.69 does not allow these RISC-3 SSCs to lose their functional capability or be removed from the facility.	This category is not applicable to the LMP SSC classification framework because all safety-related SSCs are derived from a risk-informed process and are by definition safety-significant. SR SSCs may or may not be risk significant, depending on whether the risk significance criteria are met or not, but SR SSCs are always a necessary element of DID in the LMP methodology. The LMP methodology for the selection of SR SSCs leads to the result that each SR SSC is the primary means of fulfilling the associated RSF. DID criteria preclude the reliance on one element of design to support the safety case but one of those elements is always provided by the SR SSCs.
RISC-4 Non-Safety-Related Low Safety-Significant	RISC-4 SSCs are those that are defined as non-safety-related, and that the risk-informed categorization process determines are not significant contributors to plant safety. Section 50.69 does not impose alternative treatment requirements for these RISC-4 SSCs. However, as with the RISC-3 SSCs, changes to the design bases of RISC-4 SSCs must be made in accordance with current applicable design change control requirements (if any), such as those set forth in 10 CFR 50.59.	Similar to NST Category.

2.4 Definition of Safety-Significant and Risk-Significant SSCs

2.4.1 Safety-Significant SSCs

The meaning of "safety-significant" SSC in the LMP framework is the same as that used in NRC regulations except that somewhat different significance criteria are used. The NRC glossary provides the following definition:

> *"When used to qualify an object, such as a system, structure, component, or accident sequence, this term identifies that object as having an impact on safety, whether determined through risk analysis or other means, that exceeds a predetermined significance criterion."*

Further insights into the meaning of this term in the context of safety classification under 10 CFR 50.69 can be gleaned from a definition found in Regulatory Guide 1.201:[11]

> *"The safety significance of SSCs is determined using an integrated decision-making process, which incorporates both risk and traditional engineering insights. The safety functions of SSCs include both the design-basis functions (derived from the safety-related definition) and functions credited for preventing and/or mitigating severe accidents. Treatment requirements are then commensurately applied for the categorized SSCs to maintain their functionality."*

In 10 CFR 50.69, the following definition is provided of a safety-significant function:

> *"Safety significant function means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk."*

The LMP approach to defining safety-significant functions is consistent with these definitions of safety significance and closely follows the version in 10 CFR 50.69. This definition is reflected in the approach to SSC safety classification in several respects. All the SSCs that perform safety-significant functions are classified as either SR or NSRST. NST classified SSCs perform functions that are neither risk significant nor required for DID. All the SR classified SSCs are regarded as safety-significant. SR SSCs often meet the risk significance criteria and even when they do not, they are the primary means of fulfilling the RSFs and hence are an essential element of DID adequacy. The NSRST SSCs include non-safety-related SSCs that perform risk significant functions, as well as other non-safety-related SSCs that perform functions necessary for DID adequacy.

The term "important to safety" that is used in the NRC regulatory framework including the Advanced Reactor Design Criteria and General Design Criteria is not used within the LMP methodology. All the SSCs that have risk significance or perform functions necessary for DID adequacy are contained within the LMP safety-significant SSCs and are either SR SSCs or NSRST SSCs. There are no non-safety-significant SSCs within the LMP methodology that are judged to be "important to safety." Hence it was deemed unnecessary to introduce an additional category called "important to safety" in order to formulate performance criteria for safety-significant SSCs.

2.4.2 Risk-Significant SSCs

In the LMP framework, an SSC is classified as risk significant if any of the following risk significance criteria are met for any SSC function included within the LBEs.

- A prevention or mitigation function of the SSC is necessary to meet the design objective of keeping all LBEs within the F-C Target. This F-C Target was introduced in the LMP LBE report and is presented here as Figure 2-6. 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 are within the F-C Target. Some non-SR SSCs perform functions that may be required to keep AOOs within the F-C Target. In such cases, these non-SR SSCs are also regarded as risk significant and classified as NSRST.

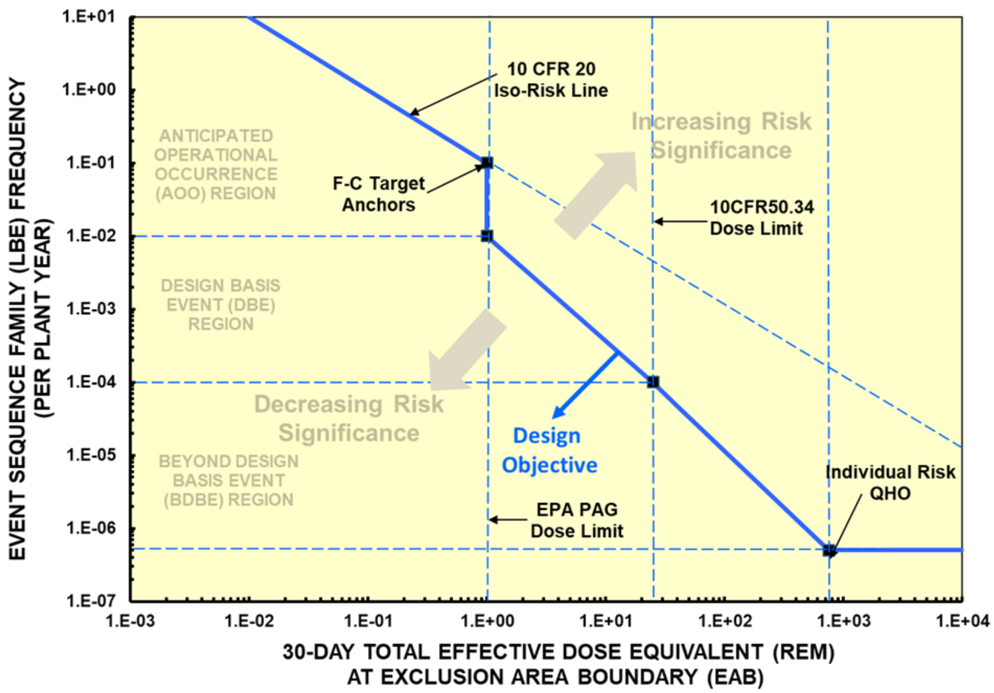

Figure 2-6. LMP Risk Significance Evaluation Target for Licensing Basis Events

- The SSC makes a significant contribution to one of the cumulative risk metrics used for evaluating the risk significance of LBEs. A significant contribution to each cumulative risk metric limit is satisfied when total frequency of all LBEs with failure of the SSC exceeds 1% of the cumulative risk metric limit. This SSC risk significance criterion may be satisfied by an SSC whether or not it performs functions necessary to keep one or more LBEs within the F-C Target. The cumulative risk metrics and limits include:

- The total mean frequency of all LBEs exceeding a site boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded based on mean estimates of frequencies and consequences. An SSC makes a significant contribution to this cumulative risk metric if the total mean frequency of exceeding a site boundary dose of 100 mrem associated with LBEs with the SSC failed is greater than 10^{-2}/plant-year.
- The average individual risk of early fatality within 1 mile of the Exclusion Area Boundary (EAB) based on mean estimates of frequencies and consequences shall not exceed 5×10^{-7}/plant-year to ensure that the NRC Safety Goal Quantitative Health Objective (QHO) for early fatality risk is met. An SSC makes a significant contribution to this cumulative metric if the individual risk of early fatalities associated with the LBEs with the SSC failed is greater than 5×10^{-9}/plant-year.
- The average individual risk of latent cancer fatalities within 10 miles of the EAB based on mean estimates of frequencies and consequences shall not exceed 2×10^{-6}/plant-year to ensure that the NRC Safety Goal QHO for latent cancer fatality risk is met. An SSC makes a significant contribution to this cumulative risk metric if the individual risk of latent cancer fatalities associated with the LBEs with the SSC failed is greater than 2×10^{-8}/plant-year.

As explained in the LBE report, the frequency units of per plant year are employed to enable the aggregation of risk contributions from accidents involving releases from one reactor module or radionuclide source as well as from multiple reactor modules and radionuclide sources.

The NGNP SSC white paper[2] recognized two situations in which the performance of an SSC was necessary to keep an LBE within the F-C Target. One situation is when a required safety function performed by a safety-related SSC is needed to keep the consequences of a DBE within the F-C Target or to prevent a high consequence BDBE from increasing in frequency into the DBE region and beyond the F-C Target. The other situation is when an SSC performs a function that is needed to keep the consequences of an AOO within the F-C Target or to prevent a high consequence DBE from increasing in frequency into the AOO region and beyond the F-C Target.

The cumulative risk limit criteria in the LMP SSC classification approach are provided to address the situation where an SSC may contribute to two or more LBEs which collectively may be risk significant even though the individual LBEs may not be significant. All LBEs within the scope of the supporting PRA should be included when evaluating these cumulative risk limits. In such cases, the reliability and availability of such SSCs may need to be controlled to manage the total integrated risks over all the LBEs. Cumulative risk metrics to support safety classification of SSCs were not used in the NGNP framework, although such metrics are used in the LMP approaches to evaluate LBEs and SSCs.

2.5 Definition of Risk-Significant LBEs

Each specific LBE can also be correlated to a level of risk significance by comparison of the LBE frequency and dose against the LBE risk significance goals reflected in the F-C Target.

Based on the convention of how risk significance is defined for accident sequences in the LWR PRA standards,* an LBE is considered risk-significant if any of the following criteria are met:

- The mean frequency of the LBE is at least 1% of the frequency at the LBE mean dose on the F-C Target as illustrated in Figure 2-7 and the LBE mean dose exposure is at least 2.5 mrem, which is 10% of the background exposure an average member of the U.S. population would receive during the 30 days of the LBE dose exposure calculation.†

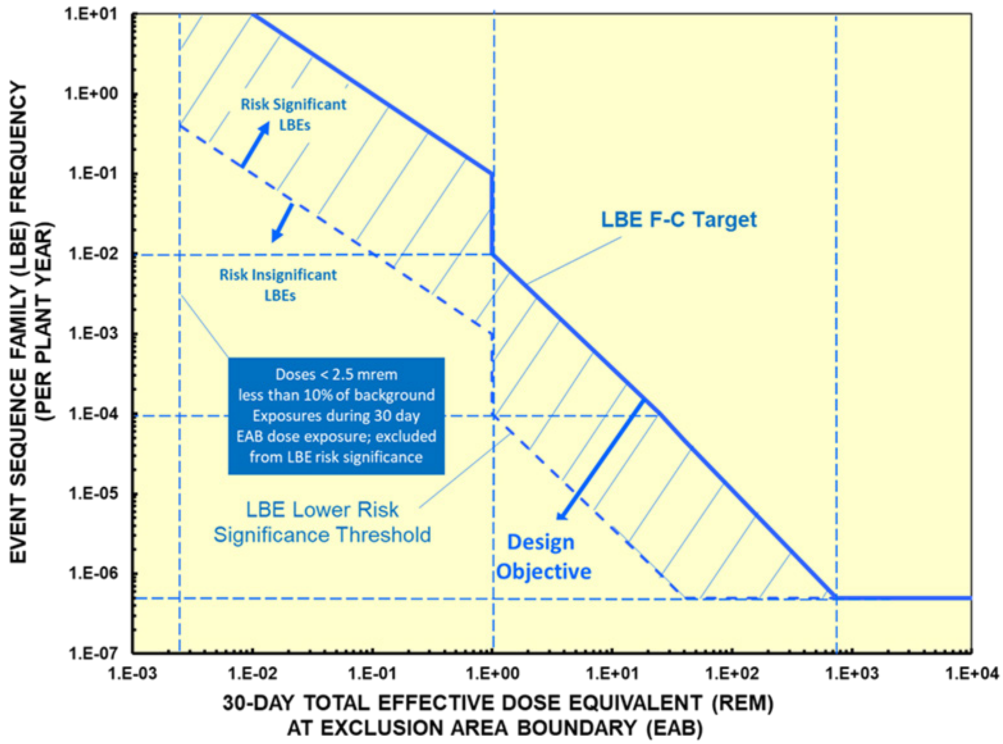

Figure 2-7. Use of the F-C Target to Define Risk Significant LBEs

- The LBE makes a significant contribution to one of the cumulative risk metrics used for evaluating the risk significance of LBEs. A significant contribution to each cumulative risk metric limit is satisfied when the contribution that the LBE makes to any of the cumulative risk metrics exceeds 1% of the metric limit. The cumulative risk metrics and limits include:
    - o The total frequency of exceeding of a site boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not

* In ASME/ANS RA-Sb-2013, risk-significant accident sequences are regarded as significant if they contribute at least 1% to core damage frequency or large early release frequency. These risk metrics are specific to LWRs. In the LMP framework, the same percentage is used but applied to each of the TI risk metrics defined in the LMP LBE report for evaluating the risk significance of individual LBEs via the F-C Target, and for evaluating the integrated risk via the cumulative risk metrics adopted for the LMP framework.

† According to the NRC Glossary, the average U.S. background radiation exposure is 300 mrem. Thus, during the 30 days of the dose exposure calculation, the background exposure would be 25 mrem. The lower limit for classifying LBEs as risk significant is judged to be 10% of this 30-day exposure.

exceeded. An LBE makes a significant contribution to this cumulative risk metric if the occurrence of the LBE would result in the plant exceeding a site boundary dose of 100 mrem at a frequency greater than 10^{-2}/plant-year.

- The average individual risk of early fatality within 1 mile of the EAB shall not exceed 5×10^{-7}/plant-year to ensure that the NRC Safety Goal QHO for early fatality risk is met. An LBE makes a significant contribution to this cumulative metric if the occurrence of the LBE would result in the plant risk of early fatalities exceeding 5×10^{-9}/plant-year.

- The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed 2×10^{-6}/plant-year to ensure that the NRC Safety Goal QHO for latent cancer fatality risk is met. An LBE makes a significant contribution to this cumulative risk metric if the occurrence of the LBE would result in the plant risk of latent cancer fatalities exceeding 2×10^{-8}/plant-year.

LBE risk significance is not used directly in the SSC safety classification process. Instead, SSC risk significance is determined using an aggregation of LBEs in which the SSC participates. However, risk significance of LBEs is considered as part of the evaluation of DID adequacy as explained more fully in the companion LMP report on DID adequacy.

2.6 Iterations Between Design Development and PRA

As discussed throughout NEI 18-04 and the supporting LMP reports, the application of the LMP methodology is an iterative process. In addition, there is flexibility in how and when the process tasks in the LMP methodology are applied. It is encouraged, though not required, that the PRA be introduced at an early stage of design when it is likely that the steps in the SSC safety classification have not been applied or perhaps only partially applied. In the case of the MHTGR[18] and Xe-100,[19] the PRAs were initiated at an early stage of design before the conceptual design had been completed. When the PRA was initiated in each of these cases, the PRA data was developed for SSCs assuming use of commercial grade equipment with no special treatment. Uncertainties in the PRA data parameters were initially assumed to be large to account for both the relative lack of operating experience and the lack of any special treatment controls.

When the SSC safety classification steps of the LMP are applied, reliability and capability targets are set for the safety-significant SSCs. These targets consider how reliable and capable the SSCs were assessed in the PRA and how much the performance may deviate without adversely impacting the risk significance of LBEs and SSCs relative to F-C Target and cumulative risk targets. Decisions regarding special treatment and application of design codes and standards for the SSCs are intended to be made via the Integrated Decision-Making Process (IDP), as discussed in the LMP report on defense-in-depth, which also is used for setting the reliability and capability targets. The management of uncertainty is a fundamental consideration in IDP evaluations and decisions. As a result of maturing design information, upgrades of the PRA following the implementation of the SSC classification process would be expected to reduce the level of uncertainty in the estimation of the PRA parameters for those SSCs having the greatest risk significance and may reduce the special treatment.

2.7 SSCs Required for Defense-in-Depth Adequacy

In the LMP framework, an IDP is used to evaluate the design and risk-informed decision to ensure adequacy of design and DID. For the purpose of safety classification, the LMP framework assumes that any SSCs that do not meet the risk-significance criteria will be classified as safety-significant only if the IDP determines that some form of special treatment is necessary to establish the adequacy of DID. This makes sense because the DID evaluation, which will incorporate traditional engineering judgments made via an integrated decision-making panel, will consider additional sources of uncertainty that are fully resolved in the PRA, including measures to enforce assumptions made in the PRA, and measures necessary to address uncertainties beyond the PRA. If a non-risk significant SSC is classified as safety-significant, it simply means that some type of special treatment is needed to address the adequacy of DID.

As a result, the universe of safety-significant SSCs in the LMP framework includes both risk significant SSCs as well as SSCs that perform functions where some form of special treatment is determined to be needed to meet DID adequacy criteria. All safety-significant SSCs are classified as SR or NSRST. All NST SSCs are not safety-significant. This provides a nexus between the LMP SSC safety classification approach and the special treatment targets for SR and NSRST SSCs as discussed in Section 3.

2.8 Conformity with Risk-Informed and Performance-Based Principles

The SSC classification process described in this report demonstrates that each SSC function reflected in the LBEs is:

- Assigned to a safety category selected from three choices: (1) safety-related; (2) non-safety-related with special treatment; and (3) non-safety-related with no special treatment.
- Specified with design targets that are commensurate with the associated safety and risk significance as described above.

The design targets specified for each SSC enables fulfillment of one or more of the following purposes consistent with a performance-based approach:

1. Specifying and procuring SSCs that satisfy performance targets
2. Verifying over time that performance is maintained
3. Controlling activities that could impact performance
4. Providing assessment and feedback of operational results to adjust programmatic activities or design as needed to meet desired outcomes

Application of performance-based principles enables accomplishing any or all of the above purposes in a manner that conforms with the NRC's definition as provided in "White Paper on Risk-Informed and Performance-Based Regulation."[12] If the performance-based principles are rigorously applied, the specified design targets will efficiently provide appropriate levels of assurance or confidence (as addressed in 10 CFR 50.69) that the SSCs will perform their design basis functions.

The design basis functions that are fulfilled are reflected in the specific LBEs associated with each SSC as described in the LBE report. The function performed is applicable to LBEs in which the SSC mitigates the consequences of the challenge, as well as those in which reliability of the SSC helps to reduce the frequency of LBEs with higher consequences. As part of the LBE selection and evaluation process, the designer selects a set of safety-related SSCs that is necessary and sufficient to perform the safety functions required to mitigate all the identified DBEs within the F-C risk significance criteria as well as prevent any high consequence BDBE from migrating into the DBE region and exceeding those risk significant criteria. Each SSC that is modeled in the PRA must perform a function in response to LBEs in one or more of the regions of the F-C chart. The selection of the SR SSCs is then used to define the DBAs where only the SR SSCs are credited to demonstrate that the dose requirements of 10 CFR 50.34 are met using conservative assumptions. The conservative safety analysis of the DBAs is considered bounding for all the realistically evaluated DBEs.

The performance-based principles that apply toward the categorization process described in Section 2 of this report are those that would enable fulfillment of the NRC's definitions of risk-informed and performance-based safety as provided in the Commission's "White Paper on Risk-Informed and Performance-Based Regulation." The objective of applying RIPB principles is, as indicated in the Commission's white paper, to focus on design requirements that enable accomplishment of performance objectives as well as avoid specifying requirements that do not contribute to risk reduction. One of the benefits of invoking 10 CFR 50.69 is that alternative design targets can replace prescriptive requirements. The alternative design targets are more targeted toward accomplishing performance objectives associated with each SSC.

The MHTGR example described in Appendix A offers a specific instance where the benefits of a performance-based approach could become apparent. As mentioned in Section 2.1, the objective of the process described is to be consistent with NRC policies on the use of performance-based alternatives. The interfaces with other RIPB decisions such as LBE selection and evaluation, and implementation of DID strategies, should be clearly defined.

The example application of this process in Appendix A describes the core heat removal functions that are provided by the Heat Transport System (HTS) and the Shutdown Cooling System (SCS) for the MHTGR. The systematic process of how the MHTGR design team selected the combination reactor, reactor vessel, and Reactor Cavity Cooling System (RCCS) as safety-related SSCs is described in some detail. The MHTGR design team rejected the option to rely on the passive heat sinks in the reactor building as the ultimate heat sink because that approach involved the need to address uncertainties regarding concrete degradation. These uncertainties were removed through the use of a robust and reliable RCCS, which, as a normally operating system, could be continuously monitored to provide assurance of its effectiveness to remove heat. This is an example of how deterministic DID considerations had a tangible impact on the selection of safety-related SSCs and selection of DBAs for the MHTGR.

In the LMP framework, an IDP is used to evaluate the design and risk-informed decision to ensure adequacy of DID. SSCs that do not meet the risk-significance criteria are classified as safety-significant only if some form of special treatment is needed to demonstrate the adequacy of DID. This makes sense because the DID evaluation will consider: 1) sources of uncertainty that may not be fully resolved in the PRA frequency dose estimates; 2) measures to enforce assumptions made in the PRA; and 3) measures to reinforce the bases for screening events out of the PRA. If a non-risk significant SSC is classified as safety-significant, it must be necessary to address the adequacy of DID. Otherwise, there would be no justification to classify a non-risk significant SSC as safety-significant.

In the context of risk-informed and performance-based principles, the SSC categorization process that has been described above offers an approach in which risk insights, engineering analysis and judgment including the principle of DID and the incorporation of safety margins, and performance history are used, to: (1) focus attention on the most important activities; (2) establish objective criteria for evaluating performance; (3) develop measurable or calculable parameters for monitoring system and licensee performance; (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes; and (5) focus on the results as the primary basis for safety decision-making.

3.0 DEVELOPMENT OF SSC DESIGN AND PERFORMANCE CRITERIA

This section describes the LMP approach for defining the design targets for each of the three SSC safety categories: SR, NSRST, and NST. These design targets begin with the identification of the SSC functions that are required to meet user targets for energy production, investment protection, worker and public safety, and licensing. SSC functions associated with the prevention and mitigation of release of radioactive material from the plant are modeled in the PRA and represented in the LBEs. The first priority in establishing the design targets for all the SSCs associated with the prevention and mitigation of release of radioactive material is to ensure that the capability and reliability of each SSC is sufficient for all the SSC functions represented in the LBEs, including the AOOs, DBEs, BDBEs, and DBAs. A related priority is to provide reasonable confidence that the reliability and capability of the SSCs are achieved and maintained throughout the lifetime of the plant.

Those SSCs that are classified as safety-related are expected to meet applicable regulatory requirements as well as reactor-specific RFDC.

3.1 Required Functional Design Criteria for Safety-Related SSCs

As noted in the previous section, SSCs classified as SR perform one or more safety functions that are required to perform either of the following:

1. Mitigate DBEs within the F-C Target and DBAs within 10 CFR 50.34 dose limits
2. Prevent any high consequence BDBEs (those with doses exceeding 10 CFR 50.34 dose limits) from exceeding 1×10^{-4}/plant-year in frequency and thereby migrate into the DBE region of the F-C evaluation

These RSFs are used within the LMP framework to define a set of reactor-specific RFDCs from which SRDC may be derived. Because the RFDCs are derived from a specific reactor technology and design, supported by a design specific PRA, and related to a set of design specific RSF, each non-LWR design would need to develop its own RFDCs. The purpose of the RFDCs and SRDCs is to form a bridge between the safety classification of SSCs and the derivation of performance and special treatment requirements for those SSCs that perform an RSF.

Guidance for the development of RFDCs for advanced non-LWRs under the LMP framework is available by reviewing the approach that was used for the MHTGR[13] which is described in Appendix A. Although these examples were developed in the 1980s, they are relevant to the implementation of the LMP framework because the MHTGR used the same approach to select the LBEs and to select the SR SSCs.

The detailed elements of this process were developed for the MHTGR and were intended to be followed by the PBMR and NGNP projects when sufficient design information and supporting PRA evaluations became available. Again, the examples from the MHTGR are used because the MHTGR is the only non-LWR that has used an RIPB process for SSC safety classification and the development of SSC design criteria and performance targets that is similar to that defined within the LMP framework. For the MHTGR, these RFDCs were defined to be applied in conjunction with GDC in Appendix A to 10 CFR 50 that were screened for applicability to the MHTGR.

From the MHTGR example, the top-down development of required safety functions and sub-functions includes a total of 15 required safety functions and associated RFDCs, which are listed in Table 3-1. Note that while the supported GDCs may be applicable to a range of advanced reactor designs, the RFDC are reactor-, technology-, and design-specific. They also focus on the inherent and passive design features of the reactor without specifying the specific SSCs that are needed to perform each function. Important principles of DID have been incorporated into these criteria that utilize multiple fission product barriers and multiple, independent, and diverse means of fulling the required safety functions. These criteria, however, are specific to the MHTGR design.

Table 3-1. MHTGR Required Safety Functions and Associated Required Functional Design Criteria*

Required Safety Function	Required Functional Design Criteria
Retain Radionuclides in Fuel Particles	I: The reactor fuel shall be designed, fabricated, and operated in such a manner that minor radionuclide releases from the fuel to the primary coolant will not exceed acceptable values.
Control Chemical Attack	II: The vessel and other components that limit or prevent the ingress of air or water shall be designed, fabricated, and operated in such a manner that the amount of air or water reacting with the core will not exceed acceptable values.
Control Heat Generation	III: The reactor shall be designed, fabricated, and operated in such a manner that the inherent nuclear feedback characteristics will ensure that the reactor thermal power will not exceed acceptable values. Additionally, the reactivity control system(s) shall be designed, fabricated, and operated in such a manner that during insertion of reactivity, the reactor thermal power will not exceed acceptable values.
Control Heat Removal	IV: The intrinsic dimensions and power densities of the reactor core, internals, and vessel, and the passive cooling pathways from the core to the environment, shall be designed, fabricated, and operated in such a manner that the fuel temperatures will not exceed acceptable values.
Control with Movable Poisons	V: Two independent and diverse sets of movable poison equipment shall be provided in the design. Either set shall be capable of limiting the heat generation of the reactor to acceptable levels during off-normal conditions.
Shutdown Reactor	VI: The equipment needed to sense, command, and execute a trip of the control rods, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions.
Shutdown Reactor Diversely	VII: The equipment needed to sense, command, and execute a trip of the reserve shutdown control equipment, along with any necessary electrical power, shall be designed, fabricated, operated, and maintained in such a manner that the shutdown of the reactor core is assured during off-normal conditions.

* The Preliminary Safety Information Document for the Standard MHTGR[13] refers to these RFDCs as "Principal Design Criteria."

Required Safety Function	Required Functional Design Criteria
Maintain Geometry for Insertion of Movable Poisons	VIII: The design, fabrication, operation, and maintenance of the control rod guide tubes, the graphite core and reflectors, the core support structure, the core lateral restraint assemblies, the reactor vessel, and reactor vessel support shall be conducted in such a manner that their integrity is maintained during off normal conditions as well as provide the appropriate geometry that permits the insertion of the control rods into the outer reflector to effect reactor shutdown.
	IX: The design, fabrication, and operation of the reserve shutdown control equipment guide tubes, the graphite core and reflectors, the core support structure, the core lateral restraint assemblies, the reactor vessel, and reactor vessel support shall be conducted in such a manner that their integrity is maintained during off-normal conditions, as well as provide the appropriate geometry that permits the insertion of reserve shutdown control material to effect reactor shutdown.
Transfer Heat to Ultimate Heat Sink	X: A highly reliable, passive means of removing the heat generated in the reactor core and radiated from the reactor vessel wall shall be provided. The system shall remove heat at a rate which limits core and vessel temperatures to acceptable levels during a loss of forced circulation.
Limit Fuel Hydrolysis	XI: The steam, feedwater and other cooling systems shall include a reliable means to limit the amount of steam and water that can enter the reactor vessel to an acceptable level.
Limit Fuel Oxidation	XII: The primary system/boundary shall be designed and fabricated to a level of quality that is sufficient to ensure high reliability of the primary system/boundary integrity needed to prevent air ingress during normal and off-normal conditions. The plant shall be designed, fabricated, operated, and maintained in a manner that ensures that the primary system boundary design limits are not exceeded.
Conduct Heat from Core to Vessel Wall	XIII: The reactor core shall be designed and configured in a manner that will ensure sufficient heat transfer by conduction, radiation, and convection to the reactor vessel wall to maintain fuel temperatures within acceptable limits following a loss of forced cooling. The materials which transfer the heat shall be chosen to withstand the elevated temperatures experienced during this passive mode of heat removal. This criterion shall be met with the primary coolant system both pressurized and depressurized.
Radiate Heat from Vessel Wall	XIV: The vessel shall be designed in a manner that will ensure that sufficient heat is radiated to the surroundings to maintain fuel and vessel temperatures within acceptable limits. This criterion shall be met with the primary coolant system in both a pressurized and depressurized condition.
Maintain Geometry for Conduction and Radiation	XV: The design, fabrication, operation, and maintenance of the core support structure, graphite core and reflectors, core lateral restraint assembly, reactor vessel, reactor vessel support, and reactor building shall be in such a manner that their integrity is maintained during off-normal conditions so as to provide a geometry conducive to removal of heat from the reactor core to the ultimate heat sink and maintain fuel temperatures within acceptable limits.

The process for identifying the required safety functions for a given reactor starts with a review of the PSFs, the safety functions modeled in the PRA, for the prevention and mitigation of LBEs and identifying which of those safety functions, if not fulfilled, would likely increase the consequences of any of the DBEs beyond the F-C Target. This normally involves the performance of sensitivity analyses* in which the performance of each safety function that mitigates the consequences of each DBE is removed or assumed to be less effective and consequences reevaluated. From the RSFs, a top-down logical development is used to define the functional requirements that must be fulfilled for the reactor design to meet each RSF. The RFDCs may be viewed as criteria that are defined in the context of the specific reactor design features that are necessary and sufficient to meet the required safety function. These MHTGR examples are provided as guidance for how this is to be done.

3.2 Design Requirements for Safety-Related SSCs

For each of the RFDCs, each advanced non-LWR under the LMP framework will need to identify a set of design requirements that will be assigned to the safety-related systems assigned to perform the required safety functions. These SSC level requirements are provided only for SR SSCs and are referred to as SRDC. Again, to provide guidance on this task of the process, some examples from the MHTGR are used.

For the MHTGR, RFDC I is assigned to fulfill the required safety function "Retain Radionuclides" listed in Table 3-1 above, as well as the retention of radionuclides in the "Reactor System (RS)," which is a safety-related SSC whose SRDCs are listed in Chapter 4 of the Preliminary Safety Information Document (PSID).[13] The associated SRDCs derived from RFDC I include:

1. The RS shall limit releases of the following key radionuclides from the plant during short-term (0 to 2-hr) and long-term (0 to 30-day) accidents to:

Nuclide	PAG (User) Limit (Ci)		10 CFR 100 Limit (Ci)	
	Short Term	Long Term	Short Term	Long Term
Kr-88	≤ 170	≤ TBD	≤ 3,400	≤ TBD
Xe-133	≤ TBD	≤ 2,300	≤ TBD	≤ 46,200
I-131	≤ 2.6	≤ 29	≤ 78	≤ 870
Sr-90	≤ 0.1	≤ 1.2	≤ 3.0	≤ 36
Ag-ll0m	≤ TBD	≤ TBD	≤ TBD	≤ TBD
Cs-137	≤ TBD	≤ TBD	≤ TBD	≤ TBD

The above Curie release limits from the plant were derived from and meet the EPA Protective Action Guideline (PAG) and l0 CFR l00 dose limits, respectively, using the meteorology and breathing rates from NRG Regulatory Guide 1.4, as well as the effectivities from Regulatory Guide 1.109 which were the appropriate references when the MHTGR examples were developed.

* This is just one example of the use of sensitivity analyses in the LMP framework. Sensitivity analyses are also performed in the development of the PRA and in the risk-informed and performance-based evaluation of defense-in-depth as part of the approach to addressing uncertainties in the estimation of LBE frequencies and consequences. Requirements for performing these analyses are covered in ASME/ANS-RA-S-1.4. Guidance for performing uncertainty analysis in the PRA is available in NUREG-1855. Insights from the uncertainty analysis are also an important input to the risk-informed and performance-based evaluation of defense-in-depth.

2. The RS shall limit radionuclide releases from the core in a manner such that the exposure to personnel shall be <10 percent of limits specified in 10 CFR 20 (applies to normal operation and AOOs only).
3. The RS shall include features to control radiation exposure to plant personnel from all core-derived radiation sources (including direct shine radiation). It is noted that the LMP framework considers all the radionuclide sources within the plant.
4. The RS shall control radiation in a manner sufficient to facilitate total, collective occupational exposure to <100 man-rem/Gigawatt electric [GW(e)]-yr. (This criterion applies to normal operation and AOOs only.)
5. The RS along with the Reactor Vessel System and the Building and Structures System "shall assure that the Reactor Building access shall be ≥ 40 hr/wk."
6. The RS shall retain radionuclides sufficiently so that the radiation due to fission product plate-out shall be less than 10 mR/h for planned maintenance or100 mR/h for unplanned maintenance.
7. The RS shall be designed to meet the Top-Level Regulatory Criteria for the Standard MHTGR given in Section 3.2 of the PSID.

It should be noted that even though the RS was classified as safety-related because of its role in preventing DBE consequences from exceeding 10 CFR 100 dose limits, its capability to retain radionuclides is required for all of the LBEs in the AOO, DBE, and BDBE regions. If fuel performance was below the fuel performance specification during normal operation, the capability to meet 10 CFR 20 would be in question. Thus, even though the safety-related SSCs are derived from the DBEs and high consequence BDBEs, their capabilities may also be needed to keep the AOOs with the F-C Target.

Another example of SRDCs derived from the MHTGR RFDCs are those that were developed for the RCCS, a safety-related SSC whose functions support RFDC IV for the required safety function "Remove Core Heat."

1. The RCCS shall have the capability to remove sufficient decay heat from the reactor core to prevent overheating of the outer control rods, the reactor, vessel, and vessel internals.
2. The RCCS shall have the capability of removing sufficient decay heat from the reactor core to maintain peak fuel temperatures below 1600°C (2900°F).
3. The RCCS shall provide the required decay heat removal capability for the "duration of the HTS and SCS shutdown whether the vessel is pressurized (with full primary coolant inventory) or depressurized."

4. Offsite radionuclide releases are to be limited as necessary to meet the numerical dose guidelines of the Top-Level Regulatory Criteria.*
5. In the event of a loss of primary coolant pressure boundary integrity, the RCCS shall be capable of withstanding a 69 kPa (10 psi) differential pressure.

As seen in the above examples for the MHTGR, the SRDCs are performance-based and keyed to required safety functions, derived from the LBEs, and used to systematically select the safety-related SSCs. This feature of the MHTGR example has been incorporated into the LMP framework.

3.3 Evaluation of SSC Performance Against Design Requirements

Although the safety-related SSCs are derived from an evaluation of the RSFs to mitigate the DBEs and DBAs, the safety-related and non-safety-related SSCs are evaluated against the full set of LBEs including the AOOs, and BDBEs as well as normal plant operation at the plant level to ensure that the F-C Target is met. This leads to design targets for both the SR and non-safety-related SSCs across the full set of LBEs, including the DBAs.

For example, in the MHTGR the Helium Purification System (HPS), which is classified as non-safety-related, has requirements to monitor the circulating activity to confirm that fuel performance is acceptable during normal operation. In addition, there is a non-safety-related plate-out probe that is periodically inspected to monitor circulating activity. It was determined for the MHTGR that plant technical specifications impacting these systems will be needed to ensure that the fuel performance requirements are being maintained. The lesson from this MHTGR example for the LMP framework is that LBEs in the AOO region need to be considered in the formulation of technical specifications.

3.4 Barrier Design Requirements

SSCs that provide functions that support the retention of radioactive material within barriers have associated regulatory design requirements that are derived from the evaluation of the LBE against the F-C Target and the RFDCs. Barriers are key part of the layers of defense considered in the evaluation of DID adequacy as discussed more fully in the companion report on evaluation of DID adequacy. These functions include “barrier functions” in which the SSC serves as a physical or functional barrier to the transport of radionuclides and indirect functions in which performance of an SSC function serves to protect one or more other SSCs that may be classified as barriers. However, a more complete perspective on the role of barriers and the SSCs that protect each barrier needs to consider the barrier response to each of the LBEs derived from the PRA. The LBEs delineate the barrier failure modes, the challenges to barrier integrity, and the interactions between SSCs that influence the effectiveness of each barrier, and the extent of barrier independence. The evaluation of mechanistic source terms that help determine the offsite doses provides another performance metric for evaluating the effectiveness of each barrier.

* This is a reference to the F-C Target derived from Top Level Regulatory Criteria that was developed for the MHTGR.

In the case of the MHTGR, there are three physical barriers to radionuclide transport:

1. The fuel/core barrier
2. The Helium Pressure Boundary (HPB) barrier
3. The reactor building barrier

When viewed across all the LBEs, each barrier within a layer of defense plays a specific role in the retention of radionuclides; however, those roles are different on different LBEs. A full picture of the synergistic roles that each of the SSCs that comprise these barriers needs to consider the ways in which the SSCs mutually support the fundamental function of radionuclide retention. For example, the safety-related vessel system provides most of the HPB; however, its RSF is to control chemical attack and to maintain core geometry to facilitate control of core heat removal and core heat generation. As a result, the most important function of the vessel system is to support the capability of the fuel particles to retain the radionuclides rather than to prevent leakage of helium coolant. There are also some small pipes on the HPB that are not safety-related but that have design requirements related to performance of a leak tight barrier during AOOs and many DBEs. Even during sequences when the HPB is breached, the design requirements for the HPB provide the capability to retain a large fraction of any fission products that are released from the fuel. However, the most important radionuclide retention functions of the HPB are to protect the fuel barrier integrity. A secondary function is to retain radionuclides that may be released from the fuel. Even though the MHTGR employed a vented confinement concept for the reactor building, the reactor building still has significant retention capabilities for releases from the fuel following depressurization of the primary system.

The reactor building in the MHTGR is classified as safety-related, not for its radionuclide retention capabilities as a separate barrier, but for its capabilities to provide the required safety functions to control chemical attack and maintain core geometry as required for the control of heat generation and control of core heat removal. Thus, like the HPB barrier, the most important radionuclide retention function of the reactor building is to protect the fuel barrier integrity. A secondary function is to retain radionuclides that may be released from the HPB.

This characterization of the safety functions of barriers is not unlike the situation with the current generation of LWRs that have three physical barriers including the fuel barrier, the reactor coolant system pressure boundary, and the containment. Each of these barriers provides a capability to retain radionuclides for different categories of event sequences. However, a full assessment of the safety significance of these barriers needs to consider the dependencies and interactions among these barriers as part of an assessment of individual layers of defense, over-dependence on a single feature or other DID considerations. A breach of the reactor coolant system pressure boundary results in a loss of coolant accident, which in turn challenges the fuel barrier by requiring operation of a coolant injection and residual heat removal system to prevent core damage. If the core heat removal systems are lost during a transient with the coolant pressure boundary initially intact, the coolant system pressure relief valves are challenged and result in temporarily breaching the second barrier depending on the response of the relief valves. If severe core damage occurs, a number of severe accident phenomena may subsequently occur that could challenge the containment integrity. Core damage could occur in certain situations and could result in a containment bypass condition in which all three barriers are degraded concurrently. Our current understanding of the safety significance of barriers in LWRs has greatly benefitted from the performance of PRAs. In the MHTGR and LWR examples, the PRA provides an effective way to assess the performance and capabilities of the barriers and this risk insight has been incorporated into the LMP framework.

It is noted that some non-LWRs employ functional layers of defense that are different from the physical barriers employed in this MHTGR example. As noted previously, in the LMP framework, the term "barrier" is used to denote any plant feature that is responsible for either full or partial reduction of the quantity of radionuclide material that may be released during a layer of defense response. It includes features such as physical barriers or any feature that is responsible for mitigating the quantity of material including time delays that permit radionuclide decay.

In summary, the definition of requirements for layers of defense cannot be fully developed simply by examining the capability of barriers to retain radionuclides. The fact that barriers are not independent for any reactor concept precludes such a simplistic approach. A systematic development of SSC design targets needs to consider a full spectrum of layer of defense challenges, barrier interactions, and dependencies. A full examination of the challenges, interactions and dependencies requires the performance of a technically sound PRA. Hence, it is critical that the approach to formulating requirements for barriers and other SSCs be linked to a systematic identification and evaluation of LBEs supported by a PRA such as that described in the companion LBE and DID reports.

3.4.1 Purpose of Special Treatment

The purpose of special treatment is reflected in the Regulatory Guide 1.201 definition of this term:

> *"…special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions."*

In the context of the LMP framework, this definition of special treatment is realized by those measures taken to provide "reasonable confidence" that SSCs will perform their functions reflected in the LBEs making use of the definition of alternative special treatment in RIEP-NEI-16. The applicable functions include those that are necessary to prevent initiating events and accidents and other functions needed to mitigate the impacts of initiating events on the performance of plant safety functions. Assurance is first accomplished by achieving and monitoring the levels of reliability and availability that are assessed in the PRA and that are determined to be necessary to meet the LBE risk evaluation criteria. These measures are focused on the prevention functions of the SSCs. Assurance is further accomplished by achieving and monitoring the capabilities of the SSCs in the performance of their mitigation functions with adequate margins to address uncertainties. The relationships between SSC reliability and capability in the performance of functions to that are needed to prevent and mitigate accidents are defined further in the next section.

The activities above are a subpart of the overall set of programmatic activities included design, manufacturing, construction, and operations of the plant that provide greater assurance that the plant capabilities and performance outcomes remain within the design basis. The broader list of possible programmatic actions presented in Section 3.5 is evaluated as part of the DID adequacy evaluation described in the companion DID report. The actual special treatments applied to a given SSC are influenced by RIPB considerations as described in the following discussion.

3.4.2 Relationship Between SSC Capability, Reliability, Mitigation, and Prevention

The safety classification of SSCs is made in the context of how the SSCs perform specific safety functions for each LBE in which they appear. If the SSC function is successful within an event sequence, the SSC helps to mitigate the consequences of the LBE. The reliability of the SSC serves to prevent the occurrence of the LBE by lowering its frequency of occurrence.

The safety classification process and the corresponding special treatments serve to control the frequencies and consequences of the LBEs within the F-C Target and to ensure that the cumulative risk targets are not exceeded. The LBE frequencies are a function of the frequencies of initiating events from internal events, internal and external hazards, and the reliabilities and capabilities of the SSCs (including the operator) to prevent and mitigate LBEs. The SSC capabilities include the ability to prevent an initiating event from progressing to an accident, to mitigate the consequences of an accident, or both. In some cases, the initiating events are failures of SSCs themselves, in which case the reliability of the SSC in question serves to limit the initiating event frequency. In other cases, the initiating events represent challenges to the SSC in question, in which case the reliability of the SSC to perform a safety function in response to the initiating event needs to be considered. Finally, there are other cases in which the challenge to the SSC in question is defined by the combination of an initiating event and combinations of successes and failures of other SSCs in response to the initiating event. All of these cases are included in the PRA and represent the set of challenges presented to a specific SSC.

A simple model of three SSCs (hereinafter referred to as SSC-1, SSC-2 and SSC-3) involved in three related LBEs for a hypothetical reactor is illustrated in Figure 3-1. The simplified event tree in this figure identifies a function of SSC-1 to prevent fuel damage from some initiating event caused by failure of SSC-3. If that function is successfully fulfilled, it leads to LBE-1 in which there is successful termination without fuel damage and no release. If SSC-1 fails in this function, fuel damage occurs, and the function of SSC-2 is to mitigate or limit the release resulting in LBE-2 and a small offsite dose denoted as d_{low}. If SSC-2 fails to perform this function, there is an unmitigated release resulting in LBE-3 with a higher offsite dose denoted as d_{high}.

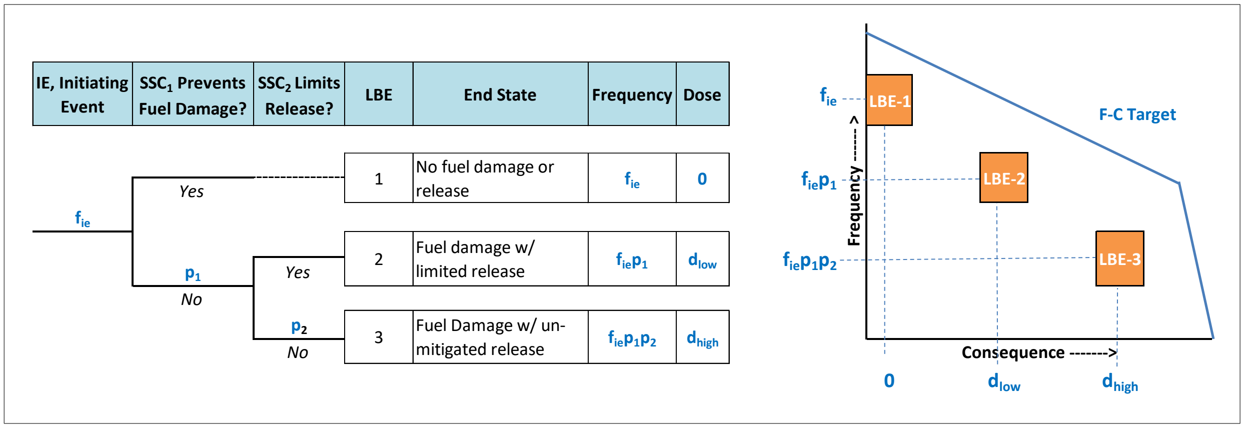

Figure 3-1. Capability and Reliability of an SSC to Mitigate and Prevent LBEs

Depending on the LBEs, the SSCs in this hypothetical problem perform both prevention and mitigation functions as shown in Table 3-2. Depending on the function, there are different performance attributes that would be the focus of any special treatment. The reliability of SSC-3 is an important attribute that would help reduce the frequency of the initiating event. The reliability of SSC-1 serves to prevent LBE-2 and LBE-3 by reducing their frequencies. The reliability of SSC-3 serves to reduce the frequency of LBE-3 and an unmitigated release. The mitigation functions of SC-2 and SC-3, however point to different attributes. For SC-1 the capability of the SSC to mitigate the challenge caused by the initiating event in preventing fuel damage expands the definition of its design targets beyond those to have a high reliability for the prevention function. Similarly, the capability of SSC-2 to mitigate the challenges associated with fuel damage expands on the definition of its design targets beyond those to perform at a high reliability.

Table 3-2. Performance Attributes for SSC Prevention and Mitigation Functions

SSC	LBEs	Function	SSC Performance Attribute for Special Treatment
Initiating Event (caused by SSC3 failure)	1,2,3	Prevent initiating event	Reliability of SSC-3 to prevent initiating event
SSC1	1	Mitigate initiating event	Capability of SSC-1 to mitigate initiating event challenge
	2	Prevent fuel damage	Reliability of SSC-1 to prevent fuel damage
	3		
SSC2	2	Mitigate fuel damage	Capability of SSC-2 to mitigate fuel damage
	3	Prevent unmitigated release	Reliability of SSC-2 in preventing unmitigated release

This example is presented to show that in the formulation of special treatment requirements, it is important to understand how the treatments may influence the reliability of the SSCs in their prevention functions, as well as the capability of the SSCs in their mitigation functions. Some special treatments may influence the capability or reliability of the SSC, others may influence both capability and reliability.

3.4.3 Role of SSC Safety Margins

SSC safety margins play an important role in the development of SSC design targets for reliability and performance capability. Acceptance limits on SSC performance are set with safety margins between the level of performance that is deemed acceptable in the safety analysis and the level of performance that would lead to damage or adverse consequences for all the LBEs in which the SSC performs a prevention or mitigation function. The magnitude of the safety margins in performance are set considering the uncertainties in performance, the nature of the associated LBEs, and criteria for adequate DID, as explained more fully in the LMP DID report. The ability to achieve the acceptance criteria in turn reflects the design margins that are part of the SSC capability to mitigate the challenges reflected in the LBEs.

A second example of the use of margins is in the selection of reliability performance targets. The reliability targets are set to ensure that the underlying LBE frequencies and consequences meet the LBE evaluation criteria with sufficient margins. These safety margins are also evaluated in the DID evaluation.

A third example of safety margins is the evaluation of margins between the frequencies and consequences of the LBEs and the F-C Target and the margins between the cumulative risk metrics and the cumulative risk targets used for LBE evaluation. These risk margins are evaluated as part of the evaluation of DID.

3.5 Specific Special Treatment Requirements for SR and NSRST SSCs

Special treatment requirements specified under the LMP framework for SSCs are derived from the following sources:

- PBMR[3] and NGNP[2] white papers on SSC safety classification
- 10 CFR 50 special treatment requirements applicable to SR SSCs
- 10 CFR 50.69 special treatment requirements for RISC-1 applicable to SR SSCs and RISC-2 SSCs applicable to NSRST SSCs

As explained in Table 2-1, all SR classified SSCs are both safety-significant and safety-related and hence correspond to 10 CFR 50.69 RISC-1 SSCs. Additionally, all NSRST SSCs are safety-significant and non-safety-related and, therefore, they correspond to 10 CFR 50.69 RISC-2 SSCs.

Table 3-3 summarizes the types of special treatments considered in the formulation of special treatment requirements. This table list categories of special treatment, applicability to each of the three LMP safety categories, and available guidance for developing the specifics of the special treatment.

Table 3-3. Summary of Special Treatments Considered for SR and NSRST SSCs

Special Treatment Category	Applicability[1] SR SSC	NSRST SSC	NST SSC	Available Guidance[4]
Requirements Associated with SSC Safety Classification				
Document basis for SSC categorization by Integrated Decision-Making Process[5]	√	√	√	Essentially the same as 10 CFR 50.69(c),[10] Guidance in RG 1.201,[11] NEI-00-04[14] 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 RG 1.201, NEI-00-04 for RISC-1 SSCs
		√		Essentially the same as 10 CFR 50.69(d), Guidance in RG 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 RG 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 LBE safety functions	√	√		Essentially same as Reliability Assurance Program in Standard Review Plan (SRP) 17.4 for safety-significant SSCs, Guidance in SRP Chapter 19.1, ASME Section XI Reliability and Integrity Management Programs[21]
Design Requirements for SSC capability to mitigate challenges reflected in LBEs	√	√		Guidance in Section 3 of this report, MHTGR PSID[13]
Maintenance Program that assures targets for SSC availability and effectiveness of maintenance to meet SSC reliability targets	√	√		Essentially same as 10 CFR 50.65 Maintenance Rule; link to MR consistent with 10 CFR 50.69 for RISC-1 (SR) and RISC-2 (NSRST) SSCs
Licensee Event Reports	√	√		Essentially same as 10 CFR 50.69(f), Guidance in RG 1.201, NEI-00-04 for RISC-1 and RISC-2 SSCs
10 CFR 50 Appendix B Quality Assurance Program	√			QA requirements consistent with 10 CFR 50 Appendix B should be risk-informed and performance-based and not compliance-based; guidance in SRP 17.5 Quality Assurance for safety-related 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 SRP 17.5 Quality Assurance for non-safety-related SSCs, 10 CFR 50.69, SRP 1.201

Special Treatment Category	Applicability[1] SR SSC	NSRST SSC	NST SSC	Available Guidance[4]
Additional Special Treatments				
Functional design criteria	√			Guidance in Section 3 of this report, INL/EXT-14-31179[17]
Technical Specifications	√	2		10 CFR 50.36, SRP, MHTGR PSID
Seismic design basis	√	3	3	Essentially the same as for existing reactors for safety-related SSCs 10 CFR 100 Appendix A
Seismic qualification testing	√			Essentially the same as for existing reactors for safety-related SSCs, 10 CFR 100 Appendix A, RG 1.100
Protection against design basis external events	√			Essentially the same as for existing reactors for safety-related SSCs, Guidance in 10 CFR 100 Appendix A, SRP 3
Equipment qualification testing	√			Essentially the same as for existing reactors for safety-related SSCs, 10 CFR 50.49
Materials surveillance testing	√			
Pre-service and In-service inspection via Reliability Integrity Management	√	2		ASME Section XI Reliability and Integrity Management Programs
Pre-service and in-service testing	√	2		In-service testing needs to be integrated with Reliability Assurance Program

[1] The applicability of any category of special treatment to any SSC must 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 by 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] SR classified SSCs are required to perform their safety functions following a Safe Shutdown Earthquake; NSRST SSCs are required to perform their safety functions following an Operational Basis Earthquake; NSRST and NST SSCs required to meet Seismic II/I requirements (required not to interfere with the performance of SR SSC safety functions 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. In this column, the term “essentially” is used to mean that non-LWR guidance under the LMP framework will need to be developed because the referenced documents were developed specifically for LWRs in which risk insights have been “back-fit.” Not all references in this column have been formally endorsed by the NRC.

[5] Integrated Decision-Making Process is discussed more fully in the LMP report on DID and is similar to that described in NEI-00-04.

It should be noted that the applicability of any category of special treatment to any SSC must 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 by design and confirmed via an IDP that is part of the LMP methodology for evaluating DID adequacy. The referenced guidance is for LWRs and would need to be modified as appropriate for the specific design and advanced non-LWR technology.

The first category of treatments listed for SSCs listed in Table 3-3 is associated with documenting and maintaining the technical bases of the SSC safety classification. The special treatments listed under this category are in alignment with requirements in 10 CFR 50.69 and associated guidance for SSCs in RISC-1 and RISC-2.[10][14][5]

The second category includes basic design targets for all safety-significant SSCs and hence are applicable to both SR and NSRST SSCs. These include the need to provide a reliability assurance program that includes reliability and availability targets as well as capability targets that are keyed on the safety functions defined in the LBEs. These reliability and availability targets are used to inform all the additional special treatment requirements that may be needed. The specific special treatment requirements are determined by an IDP that is responsible for ensuring the adequacy of DID.

The applicability of special treatment to the SSC safety categories identified in Table 3-3 is provided for general guidance only, and it is not intended to be prescriptive. The applicability of any special treatment to any SSC must be evaluated by design and confirmed via an IDP on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. For example, the designer may determine that the targets for reliability, availability, and capability of an NSRST SSC derived from the LBE risk evaluation criteria can be met with commercial grade equipment. If that is the case, then no special treatment is necessary for that specific NSRST other than monitoring performance within design ranges, whereas adequate assurance for other SSCs may require some additional special treatment. The IDP is part of the LMP process for evaluating DID adequacy as discussed in detail in the companion DID report.

Although the LMP process for selecting special treatment requirements is designed to be technology-inclusive, the actual requirements for any advanced non-LWR would necessarily be reactor technology- and design-specific. This is true because the definition of the LBEs and their frequencies, doses, and uncertainties would be design-specific.

The purpose of any special treatment requirement is to provide adequate assurance that the SSC will perform its functions in the prevention and mitigation of LBEs. Each treatment is intended to assure that the SSC has adequate reliability and capability to perform these functions.

3.5.1 Reliability Assurance for SSCs

All safety-significant SSCs, including those in the SR and NSRST categories, should be included in a Reliability Assurance Program (RAP) similar to that described in SRP 17.4.[15] The reliability and availability targets established in RAP are used to focus the selection of special treatments that are necessary and sufficient to achieve these targets and to assure they will be maintained for the life of the plant.

An example of this approach was developed by a special working group in ASME Section XI which was tasked to develop Reliability and Integrity Management (RIM) requirements for SSCs comprising the HPB for modular HTGRs. The term "Reliability and Integrity Management" was used in lieu of "in-service inspection" (ISI) because the scope of activities contained within the RIM program were broader than what has traditionally been included with the framework of ISI programs. RIM is based on the risk-informed ISI programs that were developed for operating reactors but is intended to be used to establish the initial inspection program for passive components rather than change an existing program. The objective of the RIM program was to select and implement a combination of design, fabrication, inspection, surveillance, operation, and maintenance requirements that are necessary and sufficient to meet plant level risk and reliability goals in an efficient manner. Hence, the RIM program includes the scope of activities normally associated with ISI as well as a broader scope of activities. The RIM program is currently contained in ASME Boiler and Pressure Vessel Code Section XI Division 2.[21]

The following example of how this RIM concept was first introduced in an application to the PBMR[16] is provided to demonstrate how special treatment requirements can be developed from SSC reliability and performance targets as described in this report for safety-significant SSCs. The PBMR example is provided to more clearly articulate the concepts recommended in this report for developing reliability and performance targets for safety-significant systems. Figure 3-2 shows the basic steps in developing a RIM program as applied in the PBMR pilot study.

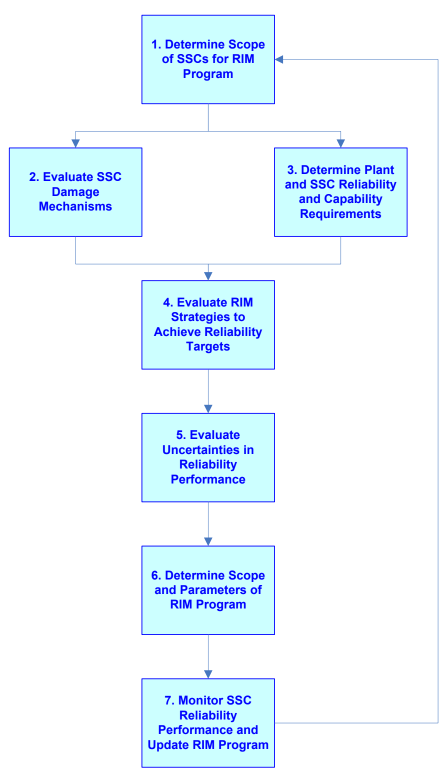

Figure 3-2. Development of Special Treatments via Reliability Integrity Management[16]

The scope of the RIM program is defined by the user in Step 1. A deterministic evaluation of damage mechanisms is performed in Step 2, which is used to help formulate strategies to achieve the reliability targets and performance objectives in Step 3. In Step 4, strategies to reduce or eliminate damage mechanisms and to achieve reliability targets are evaluated. In Step 5, uncertainties in the ability to predict the reliability performance are assessed using PRA methods for this purpose. As part of this step, deterministic evaluations are performed that include the incorporation of DID measures to address uncertainties that cannot be fully quantified. This results in the selection of additional inspections beyond those needed to meet the reliability targets. A set of RIM strategies is then selected in Step 6. The RIM approach is performance-based, as indicated in Step 7, where SSC reliability and health performance are monitored with periodic feedback loops to effect risk-informed and performance-based changes to the program.

In the PBMR RIM pilot study, the scope of SSCs that were selected included the piping and pressure vessels that comprised the HPB and the RCCS piping in the PBMR design. The SSC reliability targets were established in two stages. First, the plant level targets were identified according to the F-C Target used to select and evaluate LBEs. Second, the F-C criteria used in this example were based on requirements set by the South African regulatory authority for the PBMR pilot study.

The plant level requirements for the SSCs in this pilot study included the following:

Plant Level Reliability Goal No. 1: The frequency of events and event sequences involving loss of Reactor Pressure Vessel (RPV) structural integrity in the control of core heat removal and core heat generation shall be less than 10^{-8} per reactor year.

Plant Level Reliability Goal No. 2: The frequency of events and event sequences involving primary system depressurization due to failures of HPB SSCs with equivalent break size greater than 10mm shall be less than 10^{-2} per reactor year.

Plant Level Reliability Goal No. 3: The frequency of events and event sequences involving MPS depressurization due to failures of HPB SSCs in the Power Conversion Unit with equivalent break size greater than 230 mm shall be less than 10^{-6} per reactor year.

Plant Level Reliability Goal No. 4: The frequency of events and event sequences involving HPB depressurization due to failures of HPB SSCs connected to the top of the RPV and communicating with the core cavity with equivalent break size greater than 50mm shall be less than 10^{-6} per reactor year.

In addition to the above goals that are derived from South African regulatory requirements, the following additional SSC reliability goal was set to establish the parameters of the RIM program for the RCCS in order to meet non-mandatory plant reliability and investment risk management goals.

Plant Level Reliability Goal No. 5: The frequency of events and event sequences involving significant flooding in the reactor cavity from the RCCS involving either long outages or loss of plant investment shall be less than 10^{-5} per reactor year. This goal was used to address non-regulatory user requirements for advanced non-LWRs.

From the above plant level reliability targets, reliability targets for individual pipe welds on the HPB were established as part of the effort to complete Step 3 of the RIM procedure. These weld reliability targets are shown in Table 3-4. It is noted that these goals and associated SSC reliability allocations discussed below are for a particular design and different regulatory environment but are included in this report as an example to illustrate the approach.

Table 3-4. Allocation of Plant Goal 3 to Individual Pipe Welds

Pipe Size Range, mm	Weld Count	Weld Allowance	Target Frequency >230mm Breaks	
			Per Reactor-yr	Per Weld-yr
1000-2000	355	400	<1.00E-07	<2.50E-10
500-1000	259	300	<1.00E-07	<3.33E-10
230-500	503	750	<3.00E-07	<4.00E-10
100-230	490	750	<3.00E-07	<4.00E-10
50-100	152	250	N/A	N/A
10-50	528	750	N/A	N/A

The results of the PBMR RIM Pilot Study identified the RIM strategies and associated special treatment requirements that were derived from the plant and SSC level reliability targets, including the following examples:

- Design requirements were identified to ensure that HPB degradation mechanisms such as thermal creep, graphite dust erosion, and thermal fatigue were maintained at acceptable levels.
- On-line leak detection was found to be the most effective means of achieving and maintaining the SSC reliability targets. This led to requirements for an on-line leak detection system that included the capability to detect a small leak in the HPB within 24 hours of initiation of the leak with a probability of detection of 90%.
- Even though the in-service Non-Destructive Examination (NDE) of piping systems was found not be required to meet the SSC reliability targets, given the on-line leak detection capability described in the previous bullet, 10% of the pipe welds were selected for periodic NDE as a DID measure. The practice of setting a 10% minimum for NDEs for piping welds as a DID measure was selected to be consistent with NRC guidance for implementing RI-ISI programs for operating reactors in RG 1.178. Criteria to select the specific welds for NDE were established based on available inspection volume, degradation mechanism assessment results, and evaluation of pipe failure consequences.
- NDE requirements for the reactor vessel and other vessels in the HPB were established to ensure high confidence that safety functions associated with controlling core and reactor building geometry were met as well as Plant Level Goal 1.

3.5.2 Capability Targets for SSCs

All safety-significant SSCs in the SR and NSRST categories should have the capability to perform the safety functions to mitigate the challenges reflected in the LBEs responsible for the safety classification. SR SSCs must be capable of mitigating the DBAs within the 10 CFR 50.34 dose limits. These SR SSCs shall include appropriate functional design criteria for such functions as discussed previously in Section 3.1. Additional special treatment requirements for SR SSCs should be developed to provide assurance that the capability to perform their designated safety functions is maintained during the operating lifetime of the plant. Examples of SSC capability targets that were developed for the MHTGR SR classified SSCs are presented in the appendix. The guiding principle is that the targets should be performance-based and yield high confidence that the SSC functions will be performed during the identified LBEs. Specific capability targets for other non-LWR concepts and design will necessarily be reactor technology and design specific.

3.5.3 Quality Assurance Requirements for Safety-Significant SSCs

Table 3-3 contained the LMP guidance that QA requirements for SR SSCs be consistent with the requirements in 10 CFR 50 Appendix B. However, this is not intended to imply that advanced non-LWRs should be expected to adopt the compliance-based approach that is often used by operating reactors to address QA requirements. An important statement in Appendix B is listed as follows:

> *"The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety."*

In the LMP framework, the importance to safety of an SSC is embodied in the definitions of safety-significant and risk significant safety functions which in turn are based on the SSC safety functions reflected in the LBEs. It is expected that all the special treatment requirements for safety-significant SSCs classified in SR and NSRST, will be performance-based and directly linked to the SSC reliability and capability targets for preventing and mitigating LBEs. Thus, the industry codes and standards for meeting QA requirements for SR classified SSCs, while consistent with 10 CFR 50 Appendix B, should be applied in a risk-informed and performance-based, and not compliance based as with current operating reactors. Any QA requirements that do not contribute to providing assurance that SSC reliability and capability targets will be met are not considered to be:

> *"…consistent with their importance to safety."*

An RIPB application of Appendix B should focus on the specific SSC functions and associated LBEs that were used to establish the SSC safety classification and safety-significance determination. Only those QA requirements that add confidence that the SSC reliability and capability targets are met and contribute to reducing uncertainty in meeting the performance targets would be consistent with their importance to safety in an RIPB context.

SSCs classified as NSRST are also expected to meet appropriate risk-informed and performance-based QA requirements; however, these are not based on full 10 CFR 50 Appendix B compliance, but rather on requirements similar to those applied to non-safety-related, safety-significant SSCs in the RAP. This is consistent with SRP 17.4 RAP, which refers to SRP Section 17.5, Part V, "Non-Safety-Related SSC Quality Controls."

3.5.4 Quality Assurance Requirements for PRA

Consistent with PRAs for LWRs as discussed in RG 1.200,[20] the concept of "quality assurance" as covered in 10 CFR 50 Appendix B is not applicable to performance of non-LWR PRA in support of the LMP methodology. The idea of "quality" in PRA is referred to in the ASME/ANS PRA standards as "Technical Adequacy" as referred to in the title of Reference [20] and more recently referred to by the NRC as "Technical Acceptability." The technical adequacy of a PRA is determined by satisfying the technical requirements in the PRA standard which includes requirements for performing PRA Peer Reviews.

3.6 Conformity with Risk-Informed and Performance-Based Principles

The capability and reliability of an SSC is determined to a significant extent by performance enhancements described in 10 CFR 50.69 as "special treatment." Special treatment for an SSC, as a term used in 10 CFR 50.69, refers to activities, processes, and/or controls that are performed or used in the design, installation, maintenance, and operation of SSCs as a means of:

- Specifying and procuring SSCs that satisfy performance targets
- Verifying over time that performance is maintained
- Controlling activities that could impact performance
- Providing assessment and feedback of results to adjust activities as needed to meet desired outcomes

Accomplishments of these performance objectives is needed to provide required assurance or confidence that these reliability and capability targets are achieved and maintained throughout the lifetime of the plant.

In the context of risk-informed and performance-based principles, the SSC categorization process that has been described above offers an approach in which risk insights, engineering analysis and judgment including the structured evaluation of DID adequacy and the incorporation of safety margins, and performance history are used, to: (1) focus attention on the most important activities; (2) establish objective criteria for evaluating performance; (3) develop measurable or calculable parameters for monitoring system and licensee performance; (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes; and (5) focus on the results as the primary basis for safety decision-making.

These desired characteristics are addressed as follows:

Focus Attention on Important Activities

The provisions of 10 CFR 50.69 enable safety categorization methodologies that move away from the deterministic and prescriptive aspects of 10 CFR 50.34 where all treatment is required to meet the "reasonable assurance" standard. In contrast, a risk-informed process as described in Section 3 of this report enables withdrawing special treatments that are unimportant to safety when appropriately justified by risk information. For the LMP, this results in three, rather than two, categories of safety classification.

Objective Criteria for Performance

A monitoring program similar to the provisions of 10 CFR 50.65 can serve to provide the basic reliability and availability data to establish objective criteria for performance. SR SSCs should fall under a 10 CFR 50.65 program. The criteria may not be entirely quantitative because qualitative criteria are also acceptable under a performance-based approach.

Measurable Parameters

It is not possible to generalize the measurable (calculable) performance-based parameters in a technology-inclusive way to cover all possible non-LWR technologies. Under 10 CFR 52, any application shall include detailed information about risk-informed categorization of SSCs as well as performance targets that will be consistent with an appropriate response to LBEs. Similarly, 10 CFR 52.47 establishes that appropriate measurable parameters for LBEs will be developed for each SSC depending on the technology and design.

Flexibility that Rewards Improved Outcomes

The flexibility provided by 10 CFR 50.69 relate to SSC performance capability and reliability that needs to be achieved based on the RIPB methodology in the LMP. It can provide major performance-based benefits in terms of greater use of non-safety grade equipment for NSRST SSC, longer equipment outage times, longer surveillance and inspection intervals, lower operational dose and reduced equipment wear from excessive testing. The performance monitoring activities focus improvements on safety-significant deficiencies through the application of objective criteria for measurable parameters.

Focus on Results

A key feature of the implementation of the RIPB principles is that results of any specific activity are not considered in isolation but as a part of an integrated safety decision-making process that prioritizes meaningful safety outcomes over compliance with prescriptive requirements. This type of decision-making avoids sub-optimized outcomes and produces results more in line with the safety goals and objectives for advanced reactor design that were identified for these activities.

4.0 SUMMARY OF LMP APPROACH TO SSC SAFETY CLASSIFICATION

The LMP methodology has adopted the use of the safety classification categories developed from the NGNP white paper on SSC safety classification for use on advanced non-LWRs while drawing on insights from 10 CFR 50.69, and the bases for SSC classification in each category described below. As described in Section 3 of this document, the SSC classification process is comprised of three safety classes of SSCs:

- Safety-Related:
    - SSCs selected by the designer from the SSCs that are available to perform the 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 BDBE 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:
    - Non-safety-related SSCs relied on to perform risk significant functions. Risk significant SSCs are those that perform functions that prevent or mitigate any LBE 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:
    - All other SSCs (with no special treatment required)

Safety-significant SSCs are defined as those that are risk-significant or perform a function necessary to meet DID adequacy criteria. Safety-significant SSCs include all those classified as SR or NSRST. All NST SSCs are not safety-significant.

The RIPB SSC performance and special treatment requirements identified in Section 3 of this document for SR and NSRST SSCs to assure the SSC capability and reliability in performing functions consistent with achieving LBE performance, including uncertainties, within the F-C Target and regulatory dose limits selected for use for advanced non-LWRs designs and licensing.

5.0 GLOSSARY OF TERMS

LMP Term	Acronym	Definition	Source
Terms Associated with Functions			
Fundamental Safety Function	FSF	Safety functions common to all reactor technologies and designs; includes control heat generation, control heat removal and confinement of radioactive material	IAEA-TECDOC-1570
PRA Safety Function	PSF	Reactor design specific SSC functions modeled in a PRA that serve to prevent and/or mitigate a release of radioactive material or to protect one or more barriers to release. In ASME/ANS-Ra-S-1.4-2013 these are referred to as "safety functions." The modifier PRA is used in the LMP GD to avoid confusion with safety functions performed by Safety-Related SSCs.	LMP, ASME/ANS-Ra-S-1.4-2013
Prevention Function	--	An SSC function that, if fulfilled, will preclude the occurrence of an adverse state. The reliability of the SSC in the performance of such functions serves to reduce the probability of the adverse state.	LMP
Mitigation Function	--	An SSC function that, if fulfilled, will eliminate or reduce the consequences of an event in which the SSC function is challenged. The capability of the SSC in the performance of such functions serves to eliminate or reduce any adverse consequences that would occur if the function were not fulfilled.	LMP
Required Safety Function	RSF	A PRA Safety Function that is required to be fulfilled to maintain the consequence of one or more DBEs or the frequency of one or more high-consequence BDBEs inside the F-C Target	LMP
Required Functional Design Criteria	RFDC	Reactor design-specific functional criteria that are necessary and sufficient to meet the RSFs	LMP
Safety-Related Design Criteria	SRDC	Design criteria for SR SSCs that are necessary and sufficient to fulfill the RFDCs for those SSCs selected to perform the RSFs	LMP
Terms Associated with Licensing Basis Events			
Anticipated Operational Occurrence	AOO	Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as	LMP

LMP Term	Acronym	Definition	Source
		AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.	
Design Basis Event	DBE	Infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective and scope of DBEs form the safety design basis of the plant.	LMP
Beyond Design Basis Event	BDBE	Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant -year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification.	LMP
Design Basis Accident	DBA	Postulated accidents that are used to set design criteria and performance objectives for the design of Safety-Related SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SR SSCs classified are available to mitigate postulated accident consequences to within the 10 CFR 50.34 dose limits.	LMP
Licensing Basis Event	LBE	The entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include normal operation, AOOs, DBEs, BDBEs, and DBAs.	LMP
Frequency-Consequence Target	F-C Target	A target line on a frequency-consequence chart that is used to evaluate the risk significance of LBEs and to evaluate risk margins that contribute to evidence of adequate defense-in-depth	LMP
Risk-Significant LBE	--	An LBE whose frequency and consequence meet a specified risk significance criterion. In the LMP framework, an AOO, DBE, or BDBE is regarded as risk-significant if the combination of the upper bound (95th percentile) estimates of the frequency and consequence of the LBE are within 1% of the F-C Target AND the upper bound 30-day TEDE dose at the EAB exceeds 25 mrem.	LMP
Terms Associated with Plant Design and Structures, Systems, and Components			

LMP Term	Acronym	Definition	Source
Design Basis External Hazard Level	DBEHL	A design specification of the level of severity or intensity of an external hazard for which the Safety-Related SSCs are designed to withstand with no adverse impact on their capability to perform their RSFs	LMP
Plant		The collection of site, buildings, radionuclide sources, and SSCs seeking a single design certification or one or more operating licenses under the LMP framework. The plant may include a single reactor unit or multiple reactor modules as well as non-reactor radionuclide sources.	LMP
Multi-Reactor Module Plant	--	A plant comprising multiple reactor modules that are designed and constructed using a modular design approach. Modular design means a nuclear power plant that consists of two or more essentially identical nuclear reactors (modules) and each reactor module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other reactor module co-located on the same site, even though the nuclear power plant may have some shared or common systems.	Multi-module plant adapted from ASME/ANS-Ra-S-1.4-2013, modular design from 10CFR52.1
Safety-Related SSCs	SR SSCs	SSCs that are credited in the fulfillment of RSFs and are capable to perform their RSFs in response to any Design Basis External Hazard Level	LMP
Non-Safety-Related with Special Treatment SSCs	NSRST SSCs	Non-safety-related SSCs that perform risk-significant functions or perform functions that are necessary for defense-in-depth adequacy	LMP
Non-Safety-Related with No Special Treatment SSCs	NST SSCs	All SSCs within a plant that are neither Safety-Related SSCs nor Non-Safety-Related SSCs with Special Treatment SSCs	LMP
Risk-Significant SSC	--	An SSC that meets defined risk significance criteria. In the LMP framework, an SSC is regarded as risk-significant if its PRA Safety Function is: a) required to keep one or more LBEs inside the F-C Target based on mean frequencies and consequences; or b) if the total frequency LBEs that involve failure of the SSC PRA Safety Function contributes at least 1% to any of the LMP cumulative risk targets. The LMP cumulative risk targets include: (i) maintaining the frequency of exceeding 100 mrem to less than 1/plant-year; (ii) meeting the NRC safety goal QHO for individual risk of early fatality; and (iii) meeting the NRC safety goal QHO for individual risk of latent cancer fatality.	LMP

LMP Term	Acronym	Definition	Source
Safety-Significant SSC	--	An SSC that performs a function whose performance is necessary to achieve adequate defense-in-depth or is classified as risk-significant (see Risk-Significant SSC.)	LMP
Safety Design Approach	--	The strategies that are implemented in the design of a nuclear power plant that are intended to support safe operation of the plant and control the risks associated with accidental releases of radioactive material and protection of the public and plant workers. These strategies normally include the use of robust barriers, multiple layers of defense, redundancy, and diversity, and the use of inherent and passive design features to perform safety functions.	LMP
Terms Associated with Risk-Informed and Performance-Based Regulation and Decision-Making			
Defense-in-Depth	DID	"An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures."	NRC Glossary
Layers of Defense	--	Layers of defense are those plant capabilities and programmatic elements that provide, collectively, independent means for the prevention and mitigation of adverse events. The actual layers and number are dependent on the actual source and hazard posing the threat. See Defense-in-Depth.	LMP
Performance-Based	PB	An approach to decision-making that focuses on desired objective, calculable or measurable, observable outcomes, rather than prescriptive processes, techniques, or procedures. Performance-based decisions lead to defined results without specific direction regarding how those results are to be obtained. At the NRC, performance-based regulatory actions focus on identifying performance measures that ensure an adequate safety margin and offer incentives and flexibility for licensees to improve safety without formal regulatory intervention by the agency.	Adapted from NRC Glossary definition of performance-based regulation in order to apply to both design decisions and regulatory decision-making
Risk-Informed	RI	An approach to decision-making in which insights from probabilistic risk assessments are considered with other sources of insights	Adapted from NRC Glossary definition of performance-based regulation in order to apply to both design decisions and regulatory decision-making

LMP Term	Acronym	Definition	Source
Risk-Informed and Performance-Based Integrated Decision-Making	RIPB-DM	The union of risk information and performance information to achieve performance-based objectives	
Terms Associated with Probabilistic Risk Assessment			
Initiating Event	IE	A perturbation to the plant during a plant operating state (POS) that challenges plant control and safety systems whose failure could potentially lead to an undesirable end state and/or radioactive material release. An Initiating Event could degrade the reliability of a normally operating system, cause a standby mitigating system to be challenged, or require that the plant operators respond in order to mitigate the event or to limit the extent of plant damage caused by the Initiating Event. These events include human-caused perturbations and failure of equipment from either internal plant causes (such as hardware faults, floods, or fires) or external plant causes (such as earthquakes or high winds). An Initiating Event is defined in terms of the change in plant status that results in a condition requiring shutdown or a reactor trip (e.g., loss of main feedwater system, small reactor coolant pressure boundary [RCPB] breach) when the plant is at power, or the loss of a key safety function (e.g., decay heat removal system) for non-power modes of operation. A specific type of Initiating Event may be identified as originating from a specific cause as defined in terms such as “flood-induced transient” or “seismically-induced RCPB breach.”	ASME/ANS-Ra-S-1.4-2013
Event Sequence	ES	A representation of a scenario in terms of an Initiating Event defined for a set of initial plant conditions (characterized by a specified POS) followed by a sequence of system, safety function, and operator failures or successes, with sequence termination with a specified end state (e.g., prevention of release of radioactive material or release in one of the reactor-specific release categories. An event sequence may contain many unique variations of events (minimal cut sets) that are similar in terms of how they impact the performance of safety functions along the event sequence.	ASME/ANS-Ra-S-1.4-2013
Event Sequence Family	-	A grouping of event sequences with a common or similar POS, Initiating Event, hazard group, challenges to the plant safety functions, response of the plant in the performance of each safety function, response of each radionuclide transport barrier, and end state. An event sequence family may involve a single event sequence or several event sequences grouped together. Each release category may include one or more event sequence families. Event sequence families are not required to be explicitly	

LMP Term	Acronym	Definition	Source
		modeled in a PRA. Each event sequence family involving a release is associated with one and only one release category.	
End State		The set of conditions at the end of an Event Sequence that characterizes the impact of the sequence on the plant or the environment. In most PRAs, end states typically include success states (i.e., those states with negligible impact) and Release Categories.	ASME/ANS-Ra-S-1.4-2013
PRA Technical Adequacy	--	A set of attributes that define the technical suitability of a PRA capability to provide fit-for-purpose insights to risk-informed decision-making. It includes consideration of realism, completeness, transparency, PRA model-to-plant as-designed and as-built fidelity state, and identification and evaluation of uncertainties relative to risk levels. Strategies to achieve technical adequacy include conformance to consensus PRA standards, performance of PRA peer reviews, and structured processes for PRA model configuration control, maintenance and updates, and incorporation of new evidence that comprises the state of knowledge reflected in the PRA model development and its quantification.	LMP
Plant Operating State	POS	A standard arrangement of the plant during which the plant conditions are relatively constant, are modeled as constant, and are distinct from other configurations in ways that impact risk. POS is a basic modeling device used for a phased-mission risk assessment that discretizes the plant conditions for specific phases of an LPSD evolution. Examples of such plant conditions include core decay heat level, primary coolant level, primary temperature, primary vent status, reactor building status, and decay heat removal mechanisms. Examples of risk impacts that are dependent on POS definition include the selection of Initiating Events, Initiating Event frequencies, definition of accident sequences, success criteria, and accident sequence quantification.	ASME/ANS-Ra-S-1.4-2013
Mechanistic Source Term	MST	A source term that is calculated using models and supporting scientific data that simulate the physical and chemical processes that describe the radionuclide inventories and the time-dependent radionuclide transport mechanisms that are necessary and sufficient to predict the source term.	ASME/ANS-Ra-S-1.4-2013

6.0 REFERENCES

[1] Nuclear Energy Institute, “Risk-Informed Performance-Based Technology Inclusive Guidance for Advanced Reactor Licensing Basis Development,” NEI 18-04, Revision 0B, May 25, 2019.

[2] Idaho National Laboratory, “Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper,” INL/EXT-10-19505, [ADAMS ML102660144], September 2010.

[3] PBMR Pty. Ltd, “Safety Classification of Structures, Systems, And Components for The Pebble Bed Modular Reactor,” PBMR Document Number 043553, August 24, 2006.

[4] U.S. Department of Energy, “Safety Related Structures, Systems, and Components for the Standard MHTGR,” DOE-HTGR-87-003, January 1987.

[5] Nuclear Energy Institute, RIEP-NEI-16, “Risk Informed Engineering Programs” (10 CFR 50.69), Revision 0, November 2016.

[6] U.S. Nuclear Regulatory Commission, “NRC Staff High Level Comments on Draft White Paper on Safety Classification and Performance Criteria,” [ADAMS ML17319A210, Enclosure 3], November 21, 2017.

[7] Idaho National Laboratory, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Probabilistic Risk Assessment Approach,” Revision 0, August 2019.

[8] Idaho National Laboratory, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Selection and Evaluation of Licensing Basis Events,” Revision 0, August 2019.

[9] Idaho National Laboratory, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy,” Revision 0, August 2019.

[10] 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” December 2015.

[11] Regulatory Guide 1.201 (For Trial Use), “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” Revision 1, May 2006.

[12] SECY 1998-0144, “White Paper on Risk-Informed and Performance-Based Regulation (Revised),” June 22, 1998, and Staff Requirements Memorandum dated March 1, 1999.

[13] U.S. Department of Energy, “Preliminary Safety Information Document for the Standard MHTGR,” DOE-HTGR-86-024, September 1988.

[14] Nuclear Energy Institute, NEI-00-04, 10 CFR 50.69 “SSC Categorization Guideline,” July 2005.

[15] U.S. Nuclear Regulatory Commission, Standard Review Plan, NUREG-0800, Chapter 17.4 “Reliability Assurance Program,” Revision 1, May 2014.

[16] Fleming, K. N., et al, “Reliability and Integrity Management Program for PBMR Helium Pressure Boundary Components,” American Society of Mechanical Engineers; HTR2008: 4. International Topical Meeting on High Temperature Reactor Technology, Washington, DC, September 28 – October 1, 2008.

[17] U.S. Department of Energy “Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors,” INL/EXT-14-31179 Revision 1, December 2014.

[18] U.S. Department of Energy, “Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor,” DOE-HTGR-86-011, Revision 5, April 1988.

[19] Idaho National Laboratory, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, High Temperature, Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration,” Revision 0, August 2018.

[20] Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities,” Federal Register, U.S. Nuclear Regulatory Commission, Revision 2, March 2009.

[21] ASME BPVC-XI-2 - 2019, “BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,” ISBN: 9780791872949, 2019.

APPENDIX A EXAMPLE OF SSC SAFETY CLASSIFICATION AND DERIVATION OF DESIGN CRITERIA AND REQUIREMENTS FOR MHTGR

A.1 Example Selection of Safety-Related SSCs for MHTGR

The purpose of this section is to prevent an example of how SR classified SSCs were developed for the MHTGR.[13] This example is selected because it is the only non-LWR example available that has applied a RIPB approach to effect SSC safety classification and the approach that was followed is essentially the same process adopted within the LMP framework. Other non-LWR designs will necessarily have a different selection of SSCs, however this example is expected to provide useful guidance for the implementing the process.

DBAs correspond to the traditional off-normal events analyzed in Chapter 15 of the Safety Analysis Report. The approach to selection of LBEs that is more fully explained in the LMP LBE paper allows the transition to be made from the traditional deterministic plant response with only safety-related SSCs responding to DBAs to all SSCs responding to DBEs in the way SSCs are modeled in the PRA, so that both the conservative and expected plant behavior are understood.

As noted in Figure 2-2, to begin the design in Task 1, an initial set of prospective LBEs is identified from which to make some of the initial design decisions. The LBEs are then refined in subsequent tasks based on information provided by the initial PRA.

For consistency with current regulatory requirements, DBAs are identified by assuming that only SSCs classified as safety-related are available to perform the safety functions required to meet 10 CFR 50.34 dose criteria. The DBAs are defined by examining each of the DBEs and noting which SSCs are available and not available to support each function according to the way they are modeled in the PRA which includes all relevant combinations of SSC successes and failures. The designer then selects (Task 5 in Figure 2-2) which SSCs are to be classified as safety-related among those available to support each required safety function for each DBE. A required safety function is one that must be fulfilled to meet the 10 CFR 50.34 dose limits using conservative assumptions. After the safety-related SSCs are selected, all of the DBEs are reanalyzed with only the safety-related SSCs responding in a mechanistically conservative manner. Following this process leads to the definition of DBAs for each of the DBEs in Task 6 in Figure 2-2.

DBAs generally do not have the same sequence of events as corresponding DBEs, since the latter consider the expected plant response with all SSCs responding, whether they are safety-related or not. This means that some of the DBAs would have frequencies that are much lower than the DBE frequency cutoff of 10^{-4}/plant-year.

As noted previously, each DBE is evaluated to identify which SSCs are available and not available to support each required safety function (i.e., those safety functions that must be met to maintain the consequences of the DBE within 10 CFR 50.34 dose limits using conservative assumptions). The safety functions defined for the MHTGR, with the required safety functions are shown in Figure A-1. The development of this figure is based on an exhaustive set of consequence analyses for a wide spectrum of LBEs. One of the required safety functions is

"Remove Core Heat." To determine which SSCs are classified as safety-related requires an examination of each of the DBEs and an analysis of which SSCs are available to support that function for each DBE.

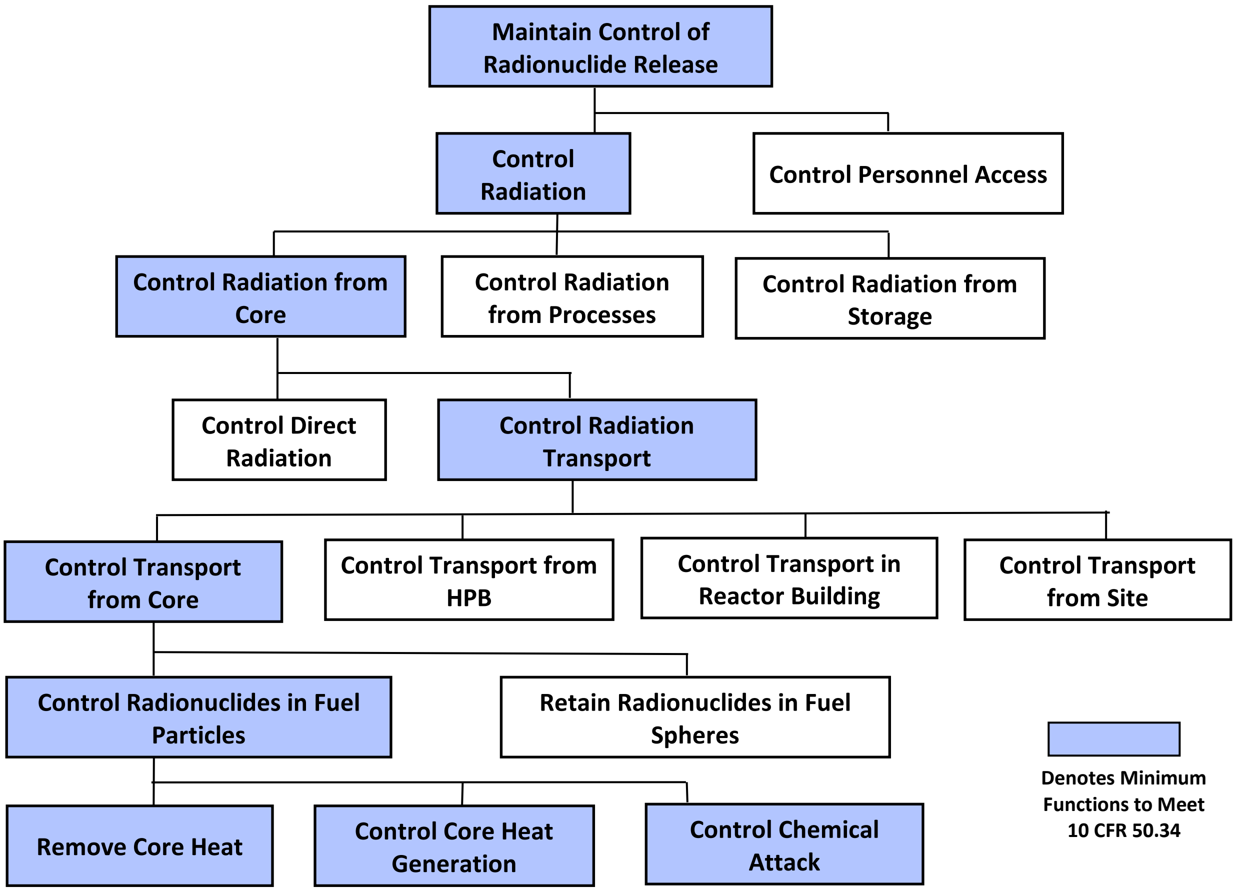

Figure A-1. Required Safety Functions for MHTGR

Consider DBE-11 that is defined in the small HPB leak event tree in Figure A-2. The evaluation of the core heat removal SSCs for that DBE is shown in Table A-1. For this DBE, there are two sets of SSCs that are capable of providing this safety function, both involving the reactor and the reactor vessel with one transferring heat into the RCCS and the other transferring heat into the passive heat sinks in the reactor cavity of the reactor building. There are other DBEs such as that defined by Sequences 2 and 4 in the event tree where the required core heat removal function is provided by the Heat Transport System (HTS), and the Shutdown Cooling System (SCS), respectively.

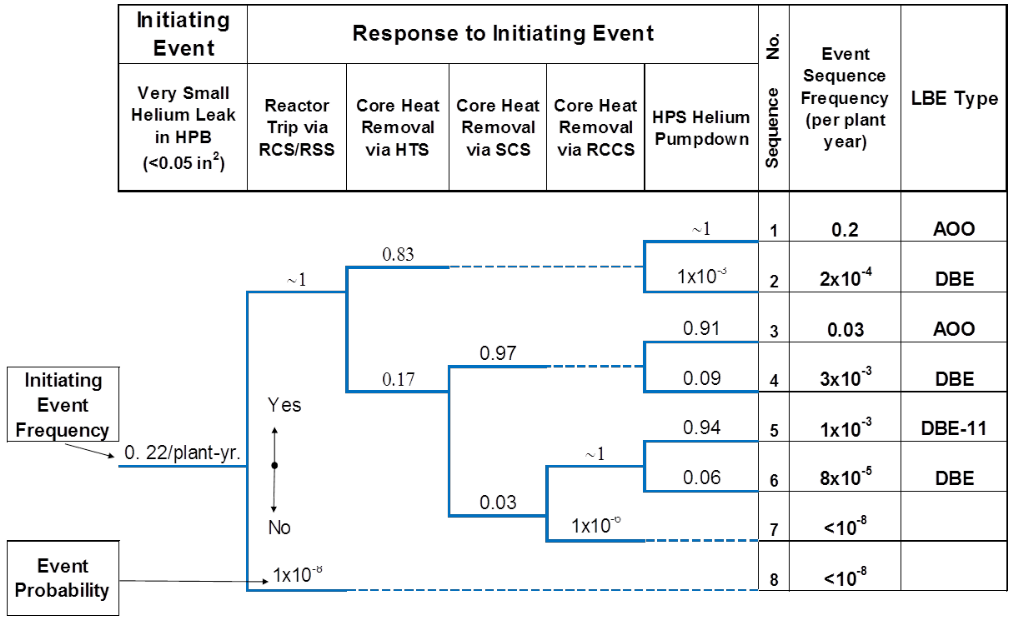

Figure A-2. MHTGR Event Tree for Very Small Helium Leak

Table A-1. Evaluation of Core Heat Removal SSCs for DBE-11

SSC Combinations Capable of Providing Core Heat Removal	Available for DBE-11?
• Reactor • Heat Transport System • Energy Conversion Area (ECA)	No
• Reactor • Shutdown Cooling System • Shutdown Cooling Water System (SCWS)	No
• Reactor • Reactor Vessel (RV) • Reactor Cavity Cooling System (RCCS)	Yes
• Reactor • Reactor Vessel • Reactor Building (RB) passive heat sinks	Yes

This evaluation is applied to each of the DBEs to determine which combinations of SSCs are available to support each required safety function. As shown in Table A-2, there are two options for selecting a set of safety-related SSCs that are capable of operation for all of the DBEs. The MHTGR design team selected the combination reactor, reactor vessel, and RCCS as safety-related SSCs. The option that relied on the passive heat sinks in the reactor building as the ultimate heat sink was rejected, as that approach involved the need to address uncertainties regarding concrete degradation, which are removed with a robust and reliable RCCS. This is an example of how deterministic defense-in-depth considerations had a tangible impact on the selection of safety-related SSCs and selection of DBAs for the MHTGR.

Table A-2. Evaluation of MHTGR SSCs for Core Heat Removal Safety Function

Alternate Sets of SSCs	Design Basis Events									SSCs Classified as SR?
	DBE 1	DBE 2	DBE 3	DBE 4	DBE 5	DBE 6/7	DBE 8/9	DBE 10	DBE 11	
• Reactor • HTS • ECA	No	No	No	No	No	No	No	No	No	No
• Reactor • SCS • SCWS	No	Yes	Yes	No	Yes	Yes	Yes	Yes	No	No
• Reactor • RV • RCCS	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
• Reactor • RV • RB	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No

When this process is completed for each required safety function, it is possible to define DBAs for each of the DBEs where only safety-related SSCs are assumed to be operable and all of the non-safety-related SSCs are assumed to be failed.* The very small helium leak event tree is presented again in Figure A-3 showing the DBA derived from DBE-11 and SR SSC selections.

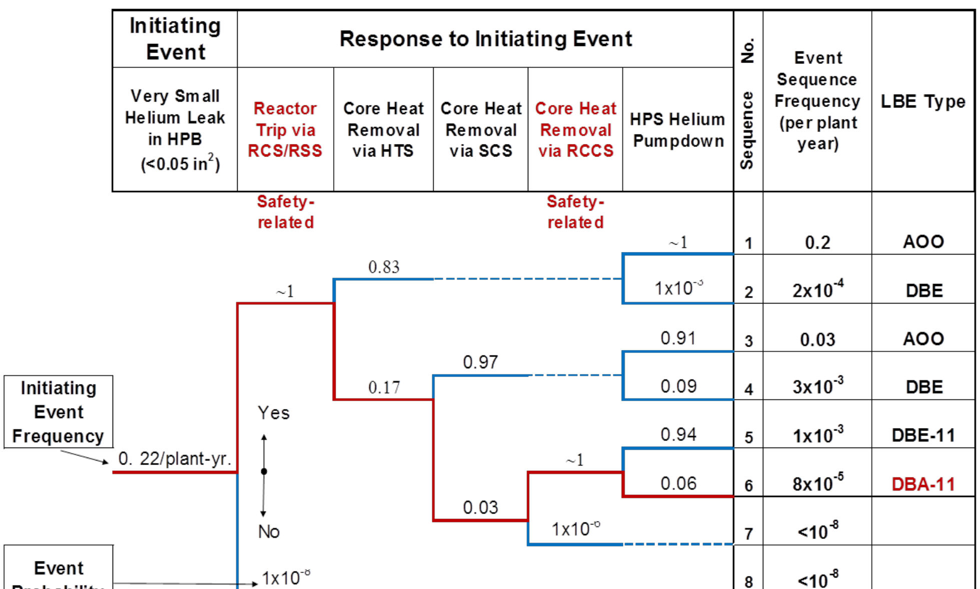

Figure A-3. MHTGR Very Small Helium Leak Event Tree with Selected DBA and Safety-Related SSCs

*If successful operation of non-safety classified SSCs produces plant conditions that are more adverse than that associated with assumed failure, the more conservative plant conditions shall be assumed in the DBA evaluation.

A.2 Safety-Related Design Criteria for Safety-Related SSCs

As noted in the previous section, SSCs classified as SR perform one or more safety functions that are required to:

1) Mitigate DBEs within the F-C target and DBAs within 10 CFR 50.34 dose limits, or

2) Prevent any high consequence BDBEs (those with doses exceeding 10 CFR 50.34 dose limits) from exceeding $1x10^{-4}$/plant-year in frequency and thereby migrate into the DBE region of the F-C evaluation.

These required safety functions are used to define a set of reactor specific SRDCs from which SSC regulatory design requirements may be derived. Guidance for the development of SRDCs for advanced non-LWRs under the LMP framework is available by reviewing the approach that was used for the MHTGR. The detailed elements of this process were developed for the MHTGR[4] and were intended to be followed by the PBMR and NGNP projects when sufficient design information and supporting PRA evaluation became available. Again, the examples from the MHTGR are used because it is the only non-LWR that has used a RIPB process for SSC safety classification and the development of SSC design criteria and performance requirements that is similar to that proposed within the LMP framework. For the MHTGR, these SRDCs were proposed to be applied in conjunction with General Design Criteria in Appendix A to 10 CFR 50 that were screened for applicability to the MHTGR.

The safety functions defined for the MHTGR were shown previously in Figure A-1. The functions that are shaded are those determined to be necessary and sufficient to keep the DBEs within the F-C target and to keep the DBAs within 10 CFR 50.34 dose limits using conservative assumptions. The four required functions shown at the bottom of this figure were used to define the first four safety-related design criteria for the MHTGR. These required safety functions include:

- Control Radionuclides in Fuel Particles
- Remove Core Heat
- Control Core Heat Removal
- Control Chemical Attack

Note that other advanced non-LWR designs would have their own reactor specific safety functions. This MHTGR example is presented only for guidance. For each of these functions, a set of SRDCs was developed that serve as the high-level success criteria for the safety functions.

From these four functions and associated SR design criteria, a set of sub-functions was systematically developed in a top-down fashion so that requirements for specific SSCs could be defined. This top-down development of sub-functions is illustrated in Figure A-4. Below each of the sub-functions in this figure, design selections are identified which are responsible for performing the associated functions. Those identified with open circles are intrinsic properties of the MHTGR design that work together with the equipment selections, identified with shaded circles. These equipment items are the SR SSC for which specific design requirements are developed from the associated SR design criteria.

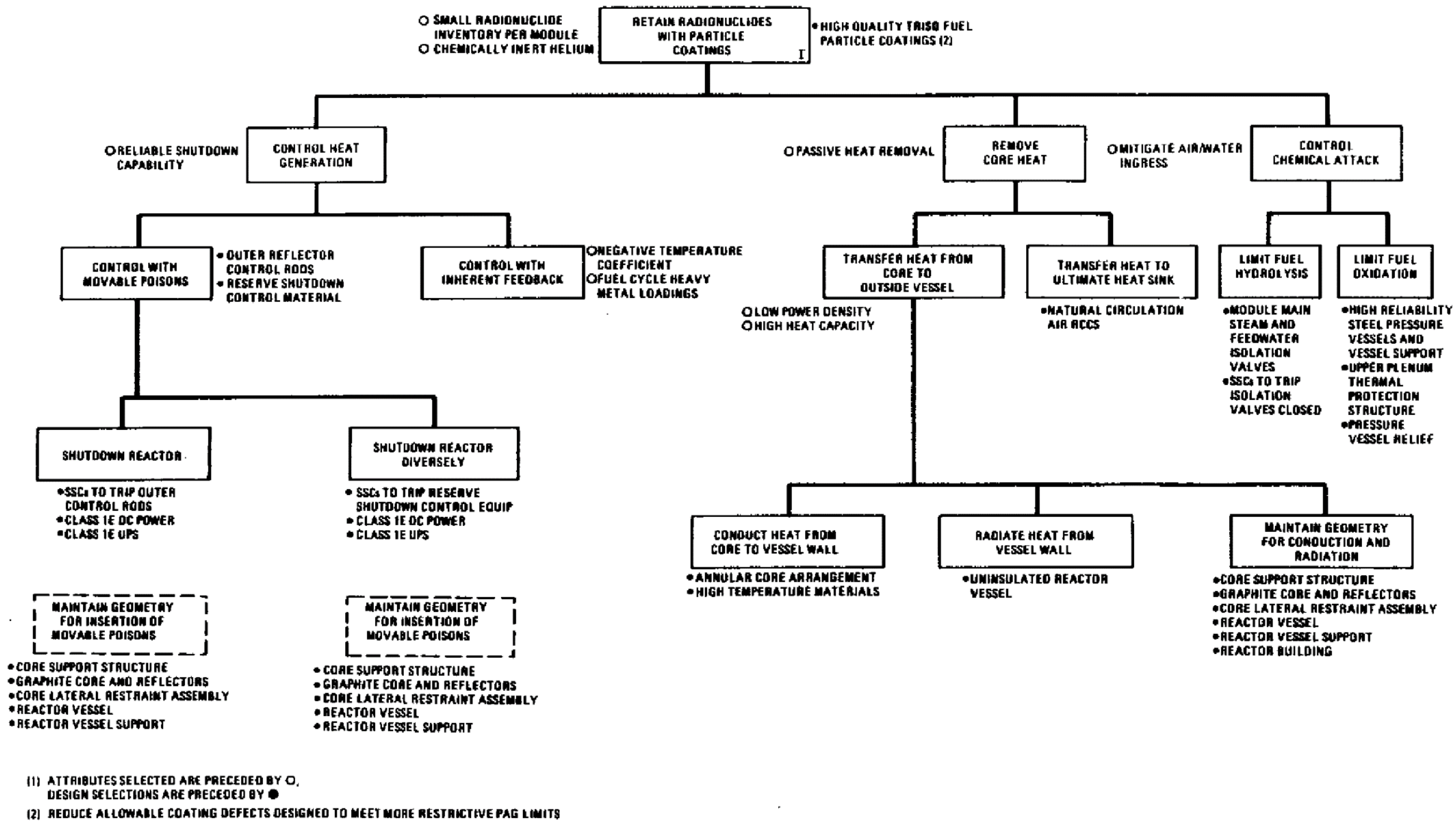

Figure A-4. Safety-Related Functions, Sub-Functions, and Design Selections for MHTGR[4]

The top-down development of required safety functions and sub-functions includes a total of 15 required safety functions and associated SRDCs which are listed in Table A-3. Note that while the supported GDCs may be applicable to a range of advanced reactor designs, the SRDC are reactor technology- and design-specific. They also focus on the inherent and passive design features of the reactor without specifying the specific SSCs that are selected to perform each function. Important principles of defense-in-depth have been incorporated into these criteria such as use of multiple fission product barriers and multiple, independent, and diverse means of fulling the required safety functions. However, these criteria are specific to the MHTGR design.

Table A-3. MHTGR Required Safety Functions and Associated Required Functional Design Criteria*

Required Safety Function	Required Functional Design Criteria
Retain Radionuclides in Fuel Particles	I: The reactor fuel shall be designed, fabricated, and operated in such a manner that minor radionuclide releases from the fuel to the primary coolant will not exceed acceptable values.
Control Chemical Attack	II: The vessel and other components that limit or prevent the ingress of air or water shall be designed, fabricated, and operated in such a manner that the amount of air or water reacting with the core will not exceed acceptable values.
Control Heat Generation	III: The reactor shall be designed, fabricated, and operated in such a manner that the inherent nuclear feedback characteristics will ensure that the reactor thermal power will not exceed acceptable values. Additionally, the reactivity control system(s) shall be designed, fabricated, and operated in such a manner that during insertion of reactivity, the reactor thermal power will not exceed acceptable values.
Control Heat Removal	IV: The intrinsic dimensions and power densities of the reactor core, internals, and vessel, and the passive cooling pathways from the core to the environment, shall be designed, fabricated, and operated in such a manner that the fuel temperatures will not exceed acceptable values.
Control with Movable Poisons	V: Two independent and diverse sets of movable poison equipment shall be provided in the design. Either set shall be capable of limiting the heat generation of the reactor to acceptable levels during off-normal conditions.
Shutdown Reactor	VI: The equipment needed to sense, command, and execute a trip of the control rods, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions.
Shutdown Reactor Diversely	VII: The equipment needed to sense, command, and execute a trip of the reserve shutdown control equipment, along with any necessary electrical power, shall be designed, fabricated, operated, and maintained in such a manner that the shutdown of the reactor core is assured during off-normal conditions.
Maintain Geometry for Insertion of Movable Poisons	VIII: The design, fabrication, operation, and maintenance of the control rod guide tubes, the graphite core and reflectors, the core support structure, the core lateral restraint assemblies, the reactor vessel, and reactor vessel support shall be conducted in such a manner that their integrity is maintained during off normal conditions as well as provide the appropriate geometry that permits the insertion of the control rods into the outer reflector to effect reactor shutdown.

*The MHTGR PSID[13] referred to these RFDCs and "Principal Design Criteria."

Required Safety Function	Required Functional Design Criteria
	IX: The design, fabrication, and operation of the reserve shutdown control equipment guide tubes, the graphite core and reflectors, the core support structure, the core lateral restraint assemblies, the reactor vessel, and reactor vessel support shall be conducted in such a manner that their integrity is maintained during off-normal conditions, as well as provide the appropriate geometry that permits the insertion of reserve shutdown control material to effect reactor shutdown.
Transfer Heat to Ultimate Heat Sink	X: A highly reliable, passive means of removing the heat generated in the reactor core and radiated from the reactor vessel wall shall be provided. The system shall remove heat at a rate which limits core and vessel temperatures to acceptable levels during a loss of forced circulation.
Limit Fuel Hydrolysis	XI: The steam, feedwater and other cooling systems shall include a reliable means to limit the amount of steam and water that can enter the reactor vessel to an acceptable level.
Limit Fuel Oxidation	XII: The primary system/boundary shall be designed and fabricated to a level of quality that is sufficient to ensure high reliability of the primary system/boundary integrity needed to prevent air ingress during normal and off-normal conditions. The plant shall be designed, fabricated, operated, and maintained in a manner that ensures that the primary system boundary design limits are not exceeded.
Conduct Heat from Core to Vessel Wall	XIII: The reactor core shall be designed and configured in a manner that will ensure sufficient heat transfer by conduction, radiation, and convection to the reactor vessel wall to maintain fuel temperatures within acceptable limits following a loss of forced cooling. The materials which transfer the heat shall be chosen to withstand the elevated temperatures experienced during this passive mode of heat removal. This criterion shall be met with the primary coolant system both pressurized and depressurized.
Radiate Heat from Vessel Wall	XIV: The vessel shall be designed in a manner that will ensure that sufficient heat is radiated to the surroundings to maintain fuel and vessel temperatures within acceptable limits. This criterion shall be met with the primary coolant system in both a pressurized and depressurized condition.
Maintain Geometry for Conduction and Radiation	XV: The design, fabrication, operation, and maintenance of the core support structure, graphite core and reflectors, core lateral restraint assembly, reactor vessel, reactor vessel support, and reactor building shall be in such a manner their integrity is maintained during off-normal conditions so as to provide a geometry conducive to removal of heat from the reactor core to the ultimate heat sink and maintain fuel temperatures within acceptable limits.

A.3 Regulatory Design Requirements for Safety-Related SSCs

For each of the RFDCs, the LMP framework will identify a set of regulatory design requirements which will be assigned to the safety-related systems assigned to perform the required safety function. These SSC level requirements are referred to in the LMP methodology as Safety-Related Design Criteria (SRDC). For the MHTGR, RFDC I for the required safety function "Retain Radionuclides" listed in Table A-3 above, the retention of radionuclides is assigned to the "Reactor System (RS)," which is a safety-related SSC whose design criteria are listed in Chapter 4 of the PSID.[13] The associated SRDCs derived from RFDC 1 include:

1. The RS shall limit releases of the following key radionuclides from the plant during short-term (0 to 2-hr) and long-term (0 to 30-day) accidents to:

Nuclide	PAG (User) Limit (Ci)		10 CFR 100 Limit (Ci)	
	Short Term	Long Term	Short Term	Long Term
Kr-88	≤ 170	≤ TBD	≤ 3,400	≤ TBD
Xe-133	≤ TBD	≤ 2,300	≤ TBD	≤ 46,200
I-131	≤ 2.6	≤ 29	≤ 78	≤ 870
Sr-90	≤ 0.1	≤ 1.2	≤ 3.0	≤ 36
Ag-ll0m	≤ TBD	≤ TBD	≤ TBD	≤ TBD
Cs-137	≤ TBD	≤ TBD	≤ TBD	≤ TBD

The above Curie release limits from the plant were derived from and meet the PAG and l0 CFR l00 dose limits, respectively, using the meteorology and breathing rates from NRG Regulatory Guide 1.4 and the effectivities from Regulatory Guide 1.109.

2. The RS shall limit radionuclide release from the core so that exposure to personnel shall be <10 percent of limits specified in 10 CFR 20 (applies to normal operation and AOOs only).

3. The RS shall include features to control radiation exposure to plant personnel from all core-derived radiation sources (including direct shine radiation).

4. The RS shall control radiation sufficiently to facilitate total, collective occupational exposure to <100 man-rem/GW(e)-yr. This criterion applies to normal operation and AOOs only.

5. The RS along with the RVS and the Building and Structures System "shall assure that the Reactor Building access shall be ≥ 40 hr/wk."

6. The RS shall retain radionuclides sufficiently so that the radiation due to fission product plate-out shall be less than 10 mR/h for planned maintenance or100 mR/h for unplanned maintenance.

7. The RS shall be designed to meet the Top-Level Regulatory Criteria for the Standard MHTGR given in Section 3.2 of the PSID.

It should be noted that even though the RS was classified as safety-related because of this role in preventing DBE consequences from exceeding 10 CFR 100 dose limits, its capability to retain radionuclides is required in all the LBEs in the AOO, DBE, and BDBE regions. If fuel performance was poor during normal operation, the capability to meet 10 CFR 20 would be in question. Hence, even though the safety-related SSCs are derived from the DBEs and high consequence BDBEs, their capabilities are also needed to keep the AOOs with the F-C target.

Another example of SRDCs derived from the MHTGR RFDCs are those that were developed for the Reactor Cavity Cooling System (RCCS), a safety-related SSC whose functions support RFDC IV for the required safety function "Remove Core Heat."

1. The RCCS shall have the capability to remove sufficient decay heat from the reactor core to prevent overheating of the outer control rods, the reactor, vessel, and vessel internals.

2. The RCCS shall have the capability of removing sufficient decay heat from the reactor core to maintain peak fuel temperatures below 1600°C (2900°F).

3. The RCCS shall provide the required decay heat removal capability for the "duration of the HTS and SCS shutdown whether the vessel is pressurized (with full primary coolant inventory) or depressurized."

4. Offsite radionuclide releases are to be limited as necessary to meet the numerical dose guidelines of the Top-Level Regulatory Criteria.*

5. In the event of a loss of primary coolant pressure boundary integrity, the RCCS shall be capable of withstanding a 69 kPa (10 psi) differential pressure.

A.4 Evaluation of SSC Performance Against Design Requirements

Although the safety-related SSCs are derived from an evaluation of the required safety functions to mitigate the DBEs and DBAs, the safety-related and non-safety-related SSCs are evaluated against the full set of LBEs including the AOOs, and BDBEs as well as normal plant operation at the plant level to ensure that the F-C target is met. This leads to design requirements for both the safety-related and non-safety-related SSCs across the full set of LBEs, including the DBAs.

For example, in the MHTGR the Helium Purification System, which is not classified as safety-related, has requirements to monitor the circulating activity to confirm that fuel performance is acceptable during normal operation. In addition, there is a non-safety-related plate-out probe that is periodically inspected to monitor circulating activity. Plant technical specifications impacting these systems will be needed to ensure that the fuel performance requirements are being maintained.

*This is a reference to the F-C Target derived from Top Level Regulatory Criteria that was developed for the MHTGR.

APPENDIX B LMP DOCUMENTATION AND FREQUENTLY ASKED QUESTIONS

B.1 LMP Documentation

The LMP team prepared independent reports on each of the four major LMP elements. Additionally, the LMP team produced a narrative report describing the processes, events, and documents involved in producing the ultimate project deliverable product, NEI 18-04 "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development." Finally, the LMP team produced a report based on the experiences of early adopters of the LMP RIPB process which includes best practices, lessons learned, and frequently asked questions and responses. Table B-1 lists the Southern Company document numbers of each of these reports. The documents are available via the DOE's Office of Scientific and Technical Information (OSTI) public document repository (https://www.osti.gov).

Table B-1. LMP Reports and Document Numbers

Report Title	Southern Company Document Number	DOE OSTI Document Number
Selection and Evaluation of Licensing Basis Events	SC-29980-100 Rev 1	TBD
Probabilistic Risk Assessment Approach	SC-29980-101 Rev 1	TBD
Safety Classification and Performance Criteria for Structures, Systems, and Components	SC-29980-102 Rev 1	TBD
Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy	SC-29980-103 Rev 1	TBD
Final Project Report	SC-29980-105 Rev. 1	TBD
LMP Lessons Learned, Best Practices, and Frequently Asked Questions	SC-29980-106 Rev 0	TBD

Licensing Basis Event Selection Approach

Inputs to the selection of LBEs are derived from a PRA of an advanced non-LWR plant. These inputs together with deterministic inputs, such as design selections and selection of Safety-Related (SR) SSCs, are used as part of the selection and evaluation of LBEs. As part of the LBE selection and evaluation process described in the LBE report, the engineering and safety analysis effort will result in a selection of a set of SR SSCs that are necessary and sufficient to perform the PRA Safety Functions (PSFs) required to keep the Design Basis Events (DBEs) within the Frequency-Consequence (F-C) target, and to prevent any high-consequence Beyond Design Basis Event (BDBE) from migrating into the DBE region and exceeding the F-C Target. The SR SSCs are then relied upon to mitigate all the Design Basis Accidents (DBAs) within the dose limits of 10 CFR 50.34 using conservative assumptions.

Probabilistic Risk Assessment Approach

This report outlines the approach to develop a PRA for advanced non-LWR plants in support of risk-informed and performance-based (RIPB) applications. Future advanced non-LWR license applications will include a design-specific PRA that is capable of supporting the applications for NRC permit(s) or license(s). When introduced at an early stage of the design, the PRA is expected to result in a more efficient risk management process. This report outlines the relevant regulatory policy and guidance for this type of PRA, describes the approach to be followed for the development of the PRA, and sets forth PRA topics that need to be addressed in order to facilitate successful design and more safety focused preparation and review of the license application.

SSC Safety Classification and Performance Requirements Approach

Information developed from and used in the development of the PRA to define event sequences and evaluate their frequencies and consequences is an input to the SSC safety classification and development of SSC performance targets. Information from the PRA is used to establish the necessary and sufficient conditions of SSC capability and reliability in order for LBE frequencies, consequences, and uncertainties to stay within the frequency-consequence evaluation criteria derived from the TLRC and to implement risk management strategies to control the total integrated risk of the plant. Reliability targets for SSCs are determined based on the need to maintain each LBE within its LBE category (Anticipated Operational Occurrence, Design Basis Event, or Beyond Design Basis Event). RIPB SSC capability targets are defined in part by the selected design margins between the LBE frequencies and dose limits for that LBE category. Special treatment requirements for SSCs are derived to achieve the necessary and sufficient degree of reliability and capability of the SSCs. This is discussed in a companion report on the LMP SSC safety classification approach.

Defense-in-Depth Adequacy

The PRA models and supporting assumptions are based in part on the plant capabilities for DID reflected in the design, as well as assumptions about the limits placed on design and operation of the plant by assumed programmatic DID measures. Information developed in the PRA is used to help evaluate the SSCs responsible for preventing and mitigating accidents. The PRA also plays an important role in the identification of key sources of uncertainty, and this supports a feedback loop to identify possible enhancements to plant capability and programmatic aspects of DID. Hence, the PRA provides important input to the risk-informed evaluation of DID, complements the NRC's deterministic approach and traditional DID philosophy, and provides a more objective, RIPB means to systematically demonstrate DID adequacy and preservation. This is discussed in a companion report on the LMP approach to evaluating DID adequacy.

LMP Final Report

The LMP team produced a narrative report describing the processes, events, and documents involved in producing the ultimate Project deliverable product, NEI 18-04 "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development." This report contains a wealth of references to documents that future users of the LMP RIPB process may find useful. Tables within the report provide references to the

NRC Agencywide Document Management System (ADAMS) Accession Numbers of many industry and NRC documents that future permit and license applicants may wish to reference in their own applications.

LMP Lessons Learned, Best Practices, and Frequently Asked Questions and Responses

The LMP team produced a report based on the experiences of early adopters of the LMP RIPB process which includes best practices, lessons learned, and frequently asked questions and responses. This report provides guidance to reactor designers on how to efficiently implement the LMP RIPB processes within their own organization and answers to 32 frequently asked questions from reactor designers.

B.2 Frequently Asked Questions

Probabilistic Risk Assessment Frequently Asked Questions

PRAQ1. How can the use of PRA technology to risk-inform the licensing of advanced non-LWRs be justified given the lack of operating experience with these reactors?

PRAQ2. How to develop adequate PRA data for initiating events and frequencies, component failure rates, maintenance unavailability, and other PRA data needs?

PRAQ3. What is the role of the PRA in the SSC safety classification process and how does safety classification influence the PRA models and data?

PRAQ4. What is the role of absolute and relative risk significance criteria in the LMP methodology?

PRAQ5. What is the applicability of 10 CFR 50 Appendix B to PRA in the LMP methodology?

PRAQ6. What is the available guidance for the systematic search for initiating events for the PRA on advanced non-LWRs?

PRAQ7. How does the LMP methodology identify and evaluate “cliff edge” effects?

PRAQ8. How does the structure of the PRA event tree logic impact the identification of the Required Safety Functions and the selection of the SR SSCs?

PRAQ9. How can the PRA standard requirements be met during the design stage when as-built and as-operated information is not available?

PRAQ10. What is the available guidance on how RSFs are determined, how they relate to FSFs?

PRAQ11. What guidance is available on the PRA treatment of safety functions provided via passive means and utilizing inherent reactor features?

PRAQ12. How can the LMP methodology be applied using dynamic PRA method?

PRAQ13. How does LMP address events that are not modeled in the PRA?

Licensing Basis Events Frequently Asked Questions

LBEQ1. What is the available guidance for how to develop mechanistic source terms using the PRA and supporting deterministic processes?

LBEQ2. How is the safety classification and special treatment of SSCs influenced by the placement of LBEs as AOOs vs. DBEs or BDBE?

LBEQ3. Is there additional information available on the selection of the F-C Target anchor points for evaluating the risk-significance of LBEs?

LBEQ4. What insights were obtained for using the F-C charts from the LMP tabletop exercises and from discussions with the NRC Staff regarding DG-1353 and SECY-19-0117?

SSC Classification Frequently Asked Questions

SSCQ1. What guidance is available on how to select among candidates for SR SSCs and possible conflicts with ARDCs?

SSCQ2. What guidance is available for how to classify NSRST SSCs and how to come up with STs.

SSCQ3. What guidance is available for how to consider whether an SSC is classified as NSRST as necessary for adequate DID?

SSCQ4. What guidance is available for how to address the full scope of SSCs in a plant including I&C, support systems, active SSCs, passive SSCs relying on inherent features, and SSCs necessary to implement safety significant operator actions?

SSCQ5. What guidance is available for how to consider the need to protect SR SSCs against DBEHLs and how to consider the requirements for NSR and NSRST SSCs?

SSCQ6. What guidance is available to discuss how SSC classification flows down from RSFs to major components and subcomponents to establish SRDC at the lowest level?

SSCQ7. What guidance is available on how to set reliability and capability targets for safety significant SSCs?

SSCQ8. What is the relationship between the Maintenance Rule scope and the LMP SSC approach to assuring reliability and capability targets for NSRST and NSR components?

SSCQ9. IEEE standards for I&C design only consider two safety classifications, 1E or non-1E. 1E is for safety functions or supporting systems that perform safety functions. Software QA for 1E is very complex and expensive. 1E V&V is also complex and difficult (i.e. exploration for unintended functions and behavior). The same concept of existing industrial codes and standards having binary rules for safety-related and non-safety-related SSC, but not addressing the "middle" NSRST, is encountered often across standards development organizations. Should equipment classified by LMP as NSRST be treated as 1E or non-1E (or, as safety-related or non-safety-related) and why?

Defense-in-Depth Frequently Asked Questions

DIDQ1. What guidance is available on how to examine the results, limitations, uncertainties, and omissions from the PRA for making IDP decisions that impact SSC safety classification and ST or deciding on practical compensatory actions?

DIDQ2. What guidance is available on how to organize the IDP and update the DID baseline through design iterations?

DIDQ3. What is the distinction between the IDP and the IDPP and why is it important?

DIDQ4. What additional guidance is there regarding the evaluation of Plant Capability DID for low dose or no dose (zero consequences) LBEs and the determination of NSRST SSCs?

Project Management Frequently Asked Questions

PMQ1. What guidance is available for how to manage the iterative process of design development, PRA development, and selection of codes and standards for SSCs?

PMQ2. How does a designer know that they are completely done implementing the LMP RIPB process with a reactor design? What is the definitive "pencils down" "finish line" event?