Southern Company

Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors

Westinghouse eVinci™ Micro-Reactor Licensing Modernization Project Demonstration

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List of Abbreviations

ANS	American Nuclear Society
AOO*	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBE*	Beyond Design Basis Event
CBS	Core Block Subsystem
CC	core challenge
CCS	Canister Containment Subsystem
CDS	Control Drum Subsystem
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
DBA*	Design Basis Accident
DBE*	Design Basis Event
DID*	defense-in-depth
DOE	Department of Energy
EPA	Environmental Protection Agency
ESS	Emergency Shutdown Subsystem
F-C*	Frequency-Consequence
F-C Target*	Frequency-Consequence Target
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
H_2	Hydrogen Release
HLR	High Level Requirement
HRUPT	Heat Pipe Rupture-Multiple
IE*	Initiating Event

LBE*	Licensing Basis Event
LMP	Licensing Modernization Project
NEI	Nuclear Energy Institute
non-LWR	non-light water reactor
NRC	Nuclear Regulatory Commission
NSRST*	Non-Safety-Related with Special Treatment
NST*	Non-Safety-Related with No Special Treatment
ORNL	Oak Ridge National Laboratory
PCS	Power Conversion Subsystem
PHXTR	Passive Heat Exchanger Tube Rupture
PRA	Probabilistic Risk Assessment
PSF*	PRA Safety Function
PWR	Pressurized Water Reactor
RIPB*	risk-informed and performance-based
RSF*	Required Safety Function
SEQ	Sequence
SR*	Safety Related
SSC	Systems, Structures, and Components
SVS	Secure Vault Subsystem
TEDE	Total Effective Dose Equivalent
U.S.	United States

*These terms have special meanings defined in NEI 18-04.

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1 INTRODUCTION

The Westinghouse eVinci™ Micro-Reactor Demonstration Project described in this document directly supports the Licensing Modernization Project (LMP), a Southern Company-led, Department of Energy- (DOE-) supported effort to achieve the desired technology-inclusive and risk-informed and performance-based (RIPB) pathway to licensing of advanced non-light water reactors (non-LWRs). The Demonstration Project's purpose, scope, objectives, and deliverables represent products supporting the LMP effort and are summarized below.

1.1 Purpose

The purpose of the eVinci Micro-Reactor LMP Demonstration Project was to exercise key processes as described in the LMP Guidance Document[1] to gain insights on adaptability of the LMP proposal to a micro reactor technology and identify potential areas of improvements and enhancements for facilitating ease of user application. Given the previous DOE and industry work which is foundational to the LMP, it was not the purpose of the eVinci Micro-Reactor LMP Demonstration Project to determine whether the proposed process is feasible to implement or to justify the process by producing particular results; affirmative answers to those questions have long been observed and documented as reflected in various documents associated with the Modular High Temperature Gas-cooled Reactor, Next Generation Nuclear Plant, and the American Nuclear Society (ANS) design standard on modular helium cooled reactors (ANS 53.1). Additionally, it should be noted that each of the constituent components of the Guidance Document process had been employed in previous DOE and industry initiatives with positive results; the Demonstration Project performed with Westinghouse is one of several LMP demonstration projects that span the spectrum of advanced non-LWR technologies including a pebble bed high temperature gas-cooled reactor,[2] a sodium-cooled fast reactor,[3] a fluoride-cooled high temperature reactor, and a molten-salt reactor. This eVinci Micro-Reactor LMP Demonstration Project provides the opportunity to evaluate the applicability of the LMP methodology to a micro-reactor concept.

Finally, it is envisioned that the output of this Demonstration Project can be used to improve the regulatory predictability of the Nuclear Regulatory Commission (NRC) review of the design and its associated safety design approach. Additionally, output of the Demonstration Project provided insights to the eVinci Micro-Reactor design-specific regulatory strategy.

1.2 Background on LMP

The LMP is a Southern Company-led, DOE-supported industry effort that describes acceptable processes for selection and evaluation of Licensing Basis Events (LBEs); safety classification of Systems, Structures, and Components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy applicable to a technology-inclusive array of advanced non-light water reactor designs. The scope of the LMP is focused on establishing guidance for advanced non-light water reactor designs so license applicants can develop inputs that can be used to comply with key regulatory requirements, including but not limited to the following:

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- 10 Code of Federal Regulations (CFR) 50.34(a) describes the content required in the Preliminary Safety Analysis Report for a Construction Permit application.
- 10 CFR 50.34(b) describes the content required in the Final Safety Analysis Report (FSAR) for an Operating License application.
- 10 CFR 52.47 describes the required information for a FSAR associated with a Standard Design Certification application.
- 10 CFR 52.79 describes the required information for a FSAR associated with a Combined License application.

The goal is to provide predictability in the regulatory process. It is envisioned that, where appropriate, the NRC and applicants will leverage the decreased societal risk of advanced reactors to streamline the NRC licensing process to be commensurate with that reduced risk by enabling shorter NRC reviews and reducing the required information in an application while ensuring the adequacy of information to support NRC's regulatory obligations.

1.3 Scope

The scope of this Demonstration Project will cover Tasks 1 through 7 described in Figure 1-1 taken from Section 3.2.2, "Licensing Basis Event Selection Process," in the LMP Guidance Document, as well as additional tasks associated with SSC safety classification in Figure 1-2. The level of effort applied to each task was defined by Westinghouse based on the level of information available for the current state of the reactor design. Where design detail is not complete, Westinghouse has applied appropriate simplistic or bounding technical assumptions to allow the completion of the Demonstration Project. Such simplifications are necessary and common to all of these limited scope LMP demonstration projects.

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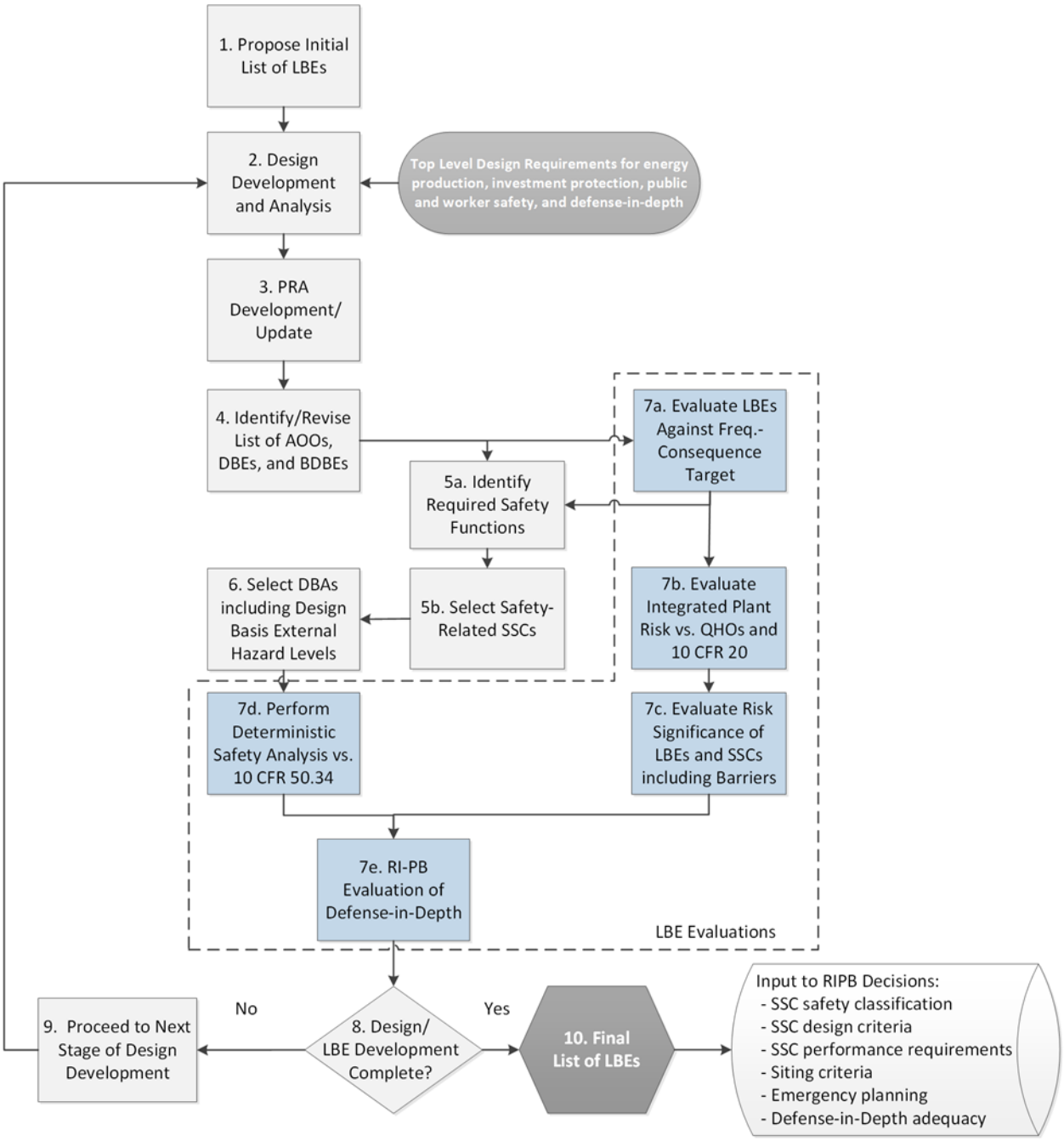

Figure 1-1. LMP Tasks for Selection and Evaluation of Licensing Basis Events[1]

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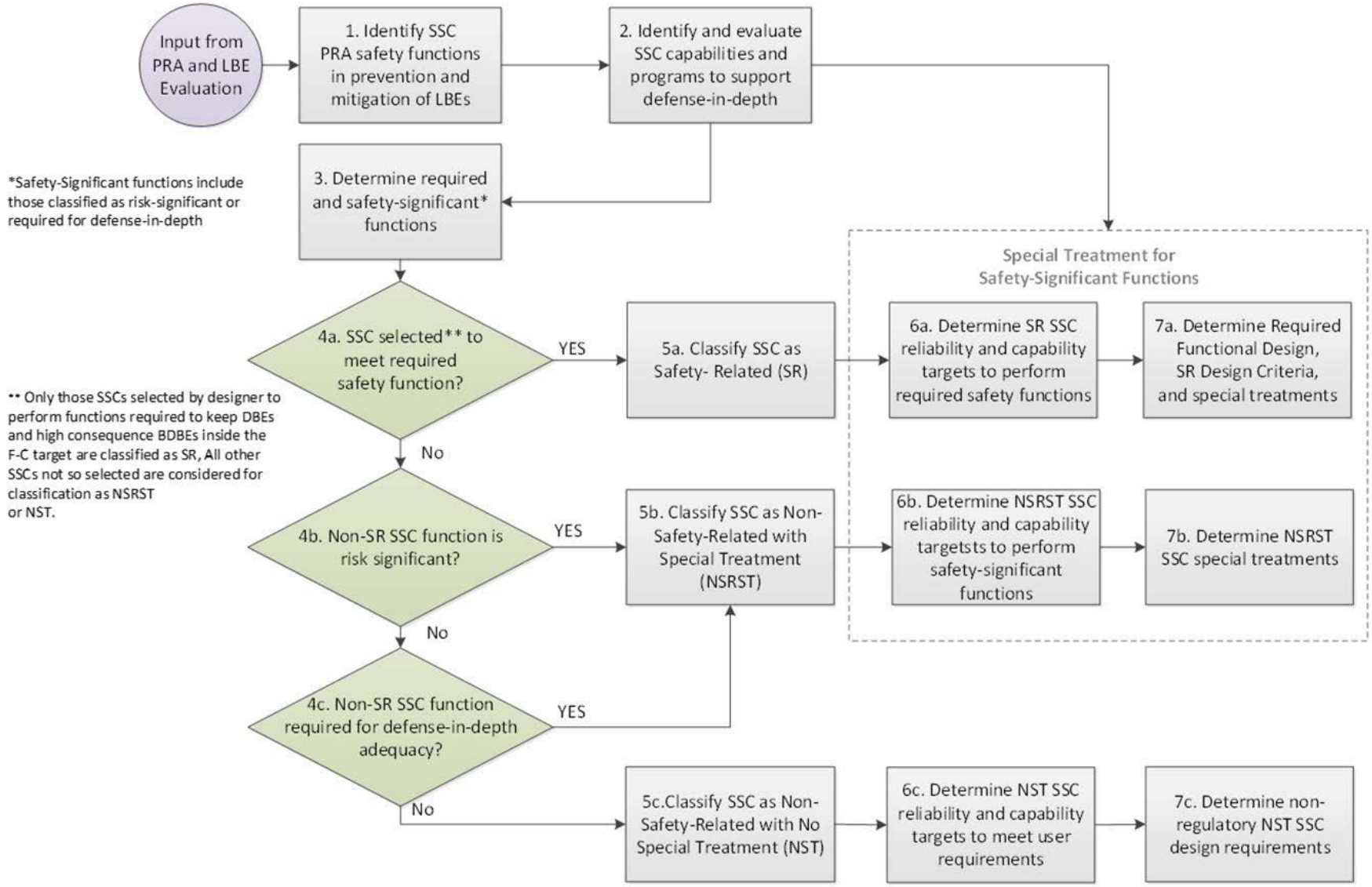

Figure 1-2. LMP SSC Function Safety Classification Process[1]

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1.4 Objectives

The objectives of the LMP eVinci Micro-Reactor Demonstration Project were to:

- Introduce the LMP process to Westinghouse.
- Demonstrate key processes of the LMP Guidance Document as applied to the eVinci Micro-Reactor design.
- Provide an opportunity for Westinghouse to develop a licensing strategy that leverages the LMP process to improve the regulatory certainty of eVinci Micro-Reactor design and safety case.

1.5 Deliverables

The deliverables of the Demonstration Project described in this report include:

- Preliminary Failure Modes and Effects Analysis (FEMA) of the eVinci Micro-Reactor Design
- A limited scope probabilistic risk assessment (PRA) model to support LMP demonstration tasks
- LBE selection and evaluation, involving Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), and Beyond Design Basis Events (BDBEs)
- Preliminary identification of the Required Safety Functions
- Preliminary Safety Classification of SSCs
- Preliminary list of Design Basis Accidents
- Documentation of a partial evaluation of DID adequacy

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2 BACKGROUND AND LINKAGE TO LMP

The U.S. commercial nuclear power industry has long sought a broadly applicable, NRC-accepted, RIPB licensing framework. Incremental advances, accelerated recently by increased interest in licensing advanced non-LWRs and Congressional interest, have resulted in an opportune time to pursue NRC endorsement of an RIPB framework. That framework is being advanced currently by the LMP, a Southern Company-led, DOE-supported effort to achieve the desired technology-inclusive, RIPB pathway to licensing of non-LWRs specifically. This Demonstration Project is an applied execution of the RIPB processes proposed by the LMP.

The LMP, working with the Nuclear Energy Institute (NEI), is currently developing a stand-alone Guidance Document describing the LMP processes for selection and evaluation of LBEs, SSC safety classification and performance requirements, and evaluation of DID adequacy.[1] The Guidance Document extracts important regulatory insights from a series of documents covering the same topics which describe the technical bases for the performance of RIPB decisions associated with designing and licensing advanced non-LWRs. The Guidance Document is intended to be endorsed by the NRC in the form of a Regulatory Guide for licensing advanced non-LWRs.

2.1 LMP Documents

Probabilistic Risk Assessment Approach

The PRA draft white paper approach document contains the historical background, technical justifications and supporting information, and implementation guidance for creating a PRA computer model fit for providing insights into plant behavior for a given phase of design development. The PRA approach is reactor technology inclusive and makes use of technology inclusive risk metrics. The PRA can be introduced at an early stage of design to incorporate risk insights into early design decisions. The PRA models are initially limited in scope and of a coarse level of detail as constrained by available supporting information. The scope and level of detail of the PRA models are increased as design and site information are available. The RIPB decisions supported by the PRA and deterministic safety approaches are reviewed and revised as the risk model definition is brought into focus. This document is available as Reference [4] and will be revised as part of the LMP to incorporate feedback from this and other planned LMP demonstrations.

Selection and Evaluation of Licensing Basis Events

The key to building the safety case of any reactor design is identifying, selecting, and evaluating LBEs, including Design Basis Accidents (DBAs). The LMP proposed approach is designed to identify LBEs that reflect the reactor design and technology-specific issues and challenges associated with each reactor's safety design approach. A systematic and prescriptive process is used to determine the safety functions required to meet risk targets, whose process provides the developer with options to select the safety-related SSCs that will be used to demonstrate satisfaction of requirements for the DBAs. This process builds on the PRA model and is tightly linked with the safety classification of SSCs. This report is available as Reference [5] and will be revised as part of the LMP to incorporate feedback from this and other planned LMP demonstrations.

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Safety Classification and Performance Criteria for Structures, Systems, and Components

Criteria are provided to classify SSCs into three safety classes: SSCs are Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST) or Non-Safety-Related with no Special Treatment (NST). Based on the SSC safety functions in the performance of both prevention and mitigation functions, the developer assigns reliability and performance targets which help ensure that selected special treatment requirements are performance-based. This LMP report is available as Reference [6] and will be revised as part of the LMP to incorporate feedback from this and other planned LMP demonstrations.

Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy

The concept of DID has long been an expressed philosophy of commercial nuclear power design, licensing, and operation. This LMP white paper proposal document seeks to systematically evaluate DID adequacy for the plant capabilities and programs that comprise DID, incorporate needed layers of defense to address uncertainties in the design and operation of the plant, and establish a fixed baseline of DID adequacy. This document is available as Reference [7] and will be revised as part of the LMP to incorporate feedback from this and other planned LMP demonstrations.

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3 DEMONSTRATION OVERVIEW

3.1 Summary of Demonstration Activities

During the demonstration planning phase, a cross-functional, multi-company core team consisting of Westinghouse, Southern Company and various industry experts involved with developing the LMP methodology was assembled. This team included subject matter experts on PRA, RIPB processes, technical project execution, licensing, and eVinci Micro-Reactor design and safety analyses. A training session for Westinghouse team members was conducted by the LMP technical leads covering the three phases of the LMP process including LBE selection and evaluation, SSC safety classification, and evaluation of DID adequacy. The core Westinghouse team began the project by executing the various tasks described in the LMP Guidance Document. Several teleconference meetings were held between Westinghouse and LMP team members where interim results were presented and discussed. These meetings included a review of the FMEA that was conducted to provide the knowledge base for the PRA, a review of the conceptual design PRA, and a review of the steps through the LMP demonstration.

During the execution of the project, Southern Company and industry consultants provided guidance regarding the application of the LMP RIPB process and led authoring of this report. The outputs, lessons, and conclusions from this demonstration effort are part of the project closeout phase and are included in this document. Insights from this demonstration will be incorporated into the final closeout report for the LMP to DOE.

3.2 eVinci Micro-Reactor Design

The eVinci Micro-Reactor is a high temperature heat pipe reactor. The core design is comprised of a solid monolithic block with three types of channels that accommodate fuel, neutron moderators and heat pipes. The monolith encapsulation of the fuel in fuel channels provides the first barrier to fission product release. The surrounding monolith block forms the second barrier. There are no moving or mechanical parts, except for reactivity control drums, which surround the monolithic block and allow absorber material to passively turn inward toward the core if power is lost, as well as on demand. A thick radial neutron reflector surrounds the monolithic core block and reactivity control drums, which, in turn, is surrounded by a neutron shield, followed by a gamma shield. A Canister Containment Subsystem (CCS) encases the entire core and each of these fission product barriers, thereby providing a third barrier to fission product release, as shown in Figure 3-1. The CCS is located within a fortified Secure Vault Subsystem (SVS) which provides protection from external events. The SVS is “sealed” to limit leaks from inside to the outside or vice versa.

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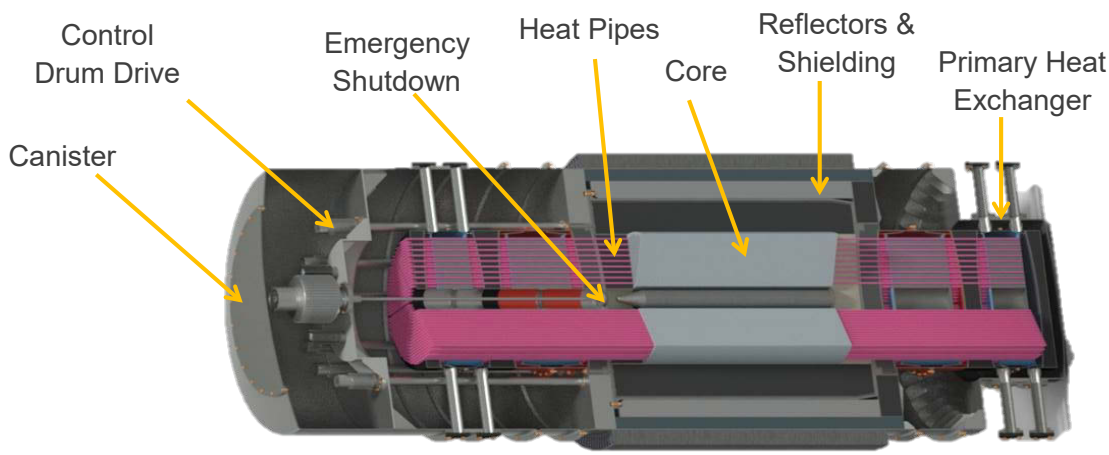

Figure 3-1. eVinci Micro-Reactor Cutaway

Each heat pipe contains a small amount of sodium liquid as the working fluid to move heat from the core to a heat exchanger and is fully encapsulated in a sealed channel. Unlike traditional sodium-cooled reactor designs, in which large volumes of sodium are pumped around the core, the eVinci Micro-Reactor requires very small amounts of sodium to serve as the coolant, almost all of which is entrained in the wicks of the heat pipes. There are no mechanical pumps, valves, or large diameter primary loop piping. Heat is transferred through the heat exchanger to a secondary side Power Conversion Subsystem (PCS) which includes equipment necessary to convert the heat into electricity for transmission.

The reactor core is itself subcritical; it cannot achieve criticality without both the neutron moderator and the neutron reflector. In the unlikely event that an emergency shutdown is necessary multiple means to do so are included in the design. The reactivity control drums, which are the primary component of the Control Drum Subsystem (CDS), have the capability to passively shutdown the reactor on loss of power. A passively actuated Emergency Shutdown Subsystem (ESS) provides alternate means of shutdown. A loss of moderation was initially considered a third possible means of shutdown; however, this option has since been discounted.

When the PCS is available, heat removal during an event is accomplished in the same manner as at-power heat removal. Heat can also be removed directly from the containment canister via a natural circulation-driven external to the CCS. Buoyancy-driven air channeled from the outside allows this heat to be transferred to the surrounding air environment, which acts as the ultimate heat sink. The components of this system are sized such that it is capable of removing heat at a rate greater than that generated by the core shortly after reactor shutdown.

It should be noted that the design utilized in the LMP demonstration described herein reflects a point in the design development of the product. The design has evolved since the majority of the input supporting the demonstration was developed, and will continue to evolve, as a result of continued design, analysis, and testing. Downstream implications on the LMP related activities will be incorporated and packaged as necessary to support various regulatory approvals; however, this LMP demonstration report will not be updated.

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3.3 FMEA Summary

The failure modes and their resulting effects on the design were evaluated by an expert panel comprised of representatives from various disciplines including design, licensing, probabilistic risk assessment, and deterministic analysis. The FMEA was limited to the functional failures resulting from at-power internal events which have the potential to result in challenges to nuclear safety. The results of the FMEA are utilized in the subsequent LMP steps as well as to inform design iterations.

3.4 PRA Overview

The American Society of Mechanical Engineers (ASME) and ANS Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants[7] provides technical requirements for the development of a PRA in support of the design and operation of non-LWR nuclear power plants. This standard is used to develop the framework of the PRA support to the eVinci Micro-Reactor LMP Demonstration Project. In addition, NEI 18-04 identifies this standard as one acceptable approach to establish PRA technical adequacy for implementing the LMP methodology.

For PRAs performed early in the design, the PRA models are typically limited to a single plant operating state associated with full-power operation and reactor core source of radioactive material. In addition, the scope and level of detail of the PRA models are consistent with the information available to describe the design features of the plant. This is the case for the PRA model developed to support to the eVinci Micro-Reactor LMP Demonstration Project that includes at-power internal events for a single reactor module. It does not include internal flooding events, internal fire events, seismic events, or external hazards. Because of the absence of water-based systems in the design of the eVinci Micro-Reactor, it is expected that the flood hazard is essentially eliminated by design. Because of the limited number of components in the eVinci Micro-Reactor design that could be considered fire ignition sources, and the small size of the plant, it is expected that a fire resulting from one of the few fire ignition sources in the plant would have an impact very similar to the loss of the individual component. Seismic events and external hazards are highly dependent on the site location which was not part of the scope of the eVinci Micro-Reactor LMP Demonstration Project.

Initiating Events (IEs) were defined for the eVinci Micro-Reactor, and event trees were developed for the IEs that addressed the key functions of reactivity control, decay heat removal, and containment and the event sequences associated with combinations of successes and failures to perform these functions. System fault trees were developed for the systems providing the key functions. These initial system models are simplified to match the design state of the eVinci Micro-Reactor. The resulting sequence end states were combined with associated dose releases to plot the results on an event sequence frequency/total effective dose graph similar to the example shown in Figure 3-2. Details of the PRA model and results are presented in Section 4.

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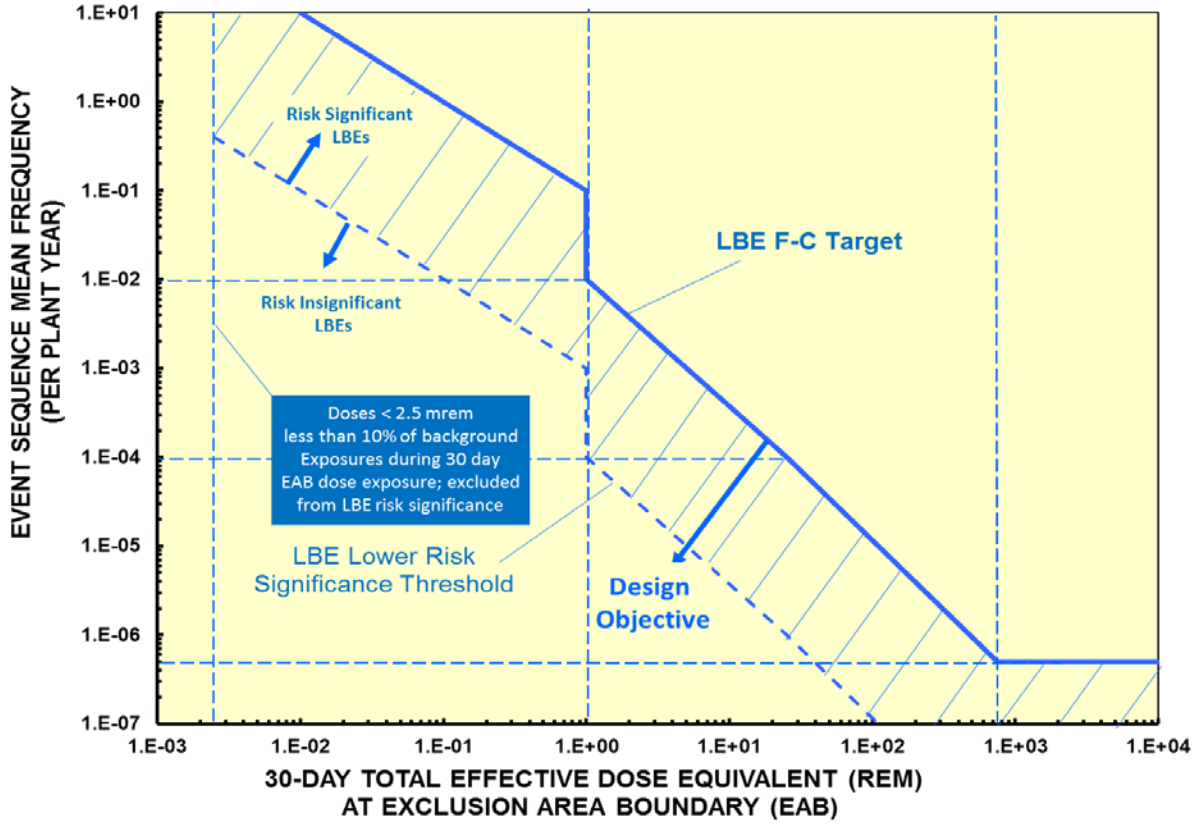

Figure 3-2. Frequency-Consequence Evaluation Criteria Proposed for LMP from Reference [1]

3.5 LBE Selection and Evaluation

The LBE selection process follows the approach described in the NEI 18-04 LMP Guidance Document. The proposed Frequency-Consequence (F-C) Target used for the eVinci Micro-Reactor LMP Demonstration Project is the same as that found in NEI 18-04 and shown in Figure 3-2. A design objective of the eVinci Micro-Reactor is to keep the LBEs well within the F-C Target with sufficient margins that can support the eventual demonstration of DID adequacy.

Section 5 provides the current efforts that have been performed for the eVinci Micro-Reactor for the LBE selection. In the PRA model, a grouping of IEs was performed into dose cases, which represent the initial set of LBEs evaluated. These dose cases were then grouped into AOOs, DBEs, and BDBEs and plotted on the F-C curve to evaluate the identified LBEs. This evaluation was then used as input into the Safety Classification process, described in Section 6, which was performed only on the system level of the eVinci Micro-Reactor design. As the plant design and licensing for the eVinci Micro-Reactor matures, classification on a component level and a more in-depth evaluation of DID will be performed. Definition of DBAs and the final LBE selection will also follow as the design matures.

The LMP LBE selection and evaluation process was implemented as indicated in the tasks on the flow chart shown in Figure 1-1 covering Tasks 1 through 7 to varying degrees of completeness and simplification as appropriate for a limited scope application in a demonstration project.

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4 PRA DEVELOPMENT

ASME/ANS RA-S-1.4-2013[7] provides technical requirements for the development of a PRA in support of the design and operation of non-LWR nuclear plants. This standard is used to develop the framework of the PRA support to the eVinci Micro-Reactor LMP Demonstration Project. Reference [7] is structured in a number of Technical Elements, High Level Requirements (HLRs) and technical Supporting Requirements that cover the entire possible scope of a PRA in terms of design stages, plant operating states, hazards and risk metrics. Because of the early stage in the design of the eVinci Micro-Reactor, not all the Technical Elements, HLRs and technical Supporting Requirements can be meaningfully addressed for the eVinci Micro-Reactor LMP Demonstration Project. This approach of selecting a subset of requirements to match the scope and level of detail of a design stage PRA is consistent with the PRA applications concept that is described in Section 3 of the standard. In the future as the design matures, the scope of application of the standard's requirements may be expanded to match the evolving design. Such an iterative process is entirely consistent with the LMP methodology described in NEI 18-04.

The eVinci Micro-Reactor PRA was developed starting with a review of possible IEs as described in Section 4.1. Event trees were developed based on the IEs identified for the eVinci Micro-Reactor and the critical functions identified in Section 4.1. Success criteria were qualitatively assessed based on the reactor design, critical functions, and discussions with the design team. System fault trees were developed for systems supporting the critical functions. The detail included in the system fault trees was limited due to the stage of design development.

Only one operator action is currently included in the eVinci Micro-Reactor PRA model and that is to trip the CDS. A conservative screening value was used for the human error probability, based on the expected simplicity of the action; as this operator action is not driving the current risk estimates, the assumed screening value is considered appropriate at this level of the design. The PRA model was developed as a large fault tree and quantified for dose cases as discussed in Section 4.4. The PRA model developed to support to the eVinci Micro-Reactor LMP Demonstration Project addresses at-power internal events.

The eVinci Micro-Reactor PRA was quantified for all sequences and dose cases were defined for each sequence of the modeled event trees. Because of the limited design details, all the sequences of the eVinci Micro-Reactor PRA were ultimately grouped into 11 dose cases covering both success sequences and sequences with failure of mitigation functions. The reliability data for the eVinci Micro-Reactor components is currently a large source of uncertainty. Generic data sources were used for generic components. Parametric uncertainties were not addressed at this stage of the eVinci Micro-Reactor design due to the fact that epistemic uncertainties are currently expected to dominate the risk insights. While only the "base case" PRA is currently used for the equipment classification, every assumption made in the development of the eVinci Micro-Reactor PRA was reviewed and agreed upon with the design team and a fairly large number of sensitivities were performed to address the associated epistemic uncertainties (i.e., model uncertainties), which are expected to be progressively reduced with the evolution of the design.

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4.1 Selection of Initiating Events

The IE analysis for the eVinci Micro-Reactor PRA was performed to provide a reasonably complete identification of IEs for at-power operation and reactor core source of radioactive material for a one reactor module plant, in a manner commensurate with the current plant design stage.

Initiating events that challenge normal plant operation, when the plant is at power, and that require successful mitigation to prevent a release of radioactive material are identified using a structured, systematic process that accounts for plant-specific features.

Because of the limited design information and the lack of operating procedures, existing lists of known initiators for pressurized water reactor designs are used as starting point, with an initial assessment of the applicability to the eVinci Micro-Reactor design. Each pressurized water reactor event was investigated for the possibility of functionally equivalent events that could impact the eVinci Micro-Reactor design.

A review was performed of the IEs identified in NUREG/CR-3862,[9] NUREG/CR-5750,[10] and NUREG/CR-6928[11] to determine if they are applicable to the eVinci Micro-Reactor design. Note that these documents define events for light water reactor technology, but the review examined whether a similar type of event was possible for the eVinci Micro-Reactor design. Shutdown-related IEs from Reference [8] (2015 update) were not addressed because at this time the eVinci Micro-Reactor PRA is only developed for at-power operation.

A FMEA (Section 3.3) was performed for the initial design of the eVinci Micro-Reactor and was used to supplement the identification of IEs and to identify events that are specific and unique to the eVinci Micro-Reactor.

The identified IEs were grouped for similar consequences resulting in the following four IE categories:

- Spurious reactor trip
- PCS failures generating an IE
- Primary heat exchanger tube rupture
- Multiple (more than three) heat pipe seal ruptures

Initiating event frequencies were developed using existing nuclear industry data, using engineering judgement, and making assumptions about failure rates that were reviewed by the design team, and quantifying a system fault tree.

4.2 Definition of Event Sequences and Event Sequence Families

A simplified functional event tree was generated for the eVinci Micro-Reactor. This functional event tree was used as a vehicle for a systematic event sequence analysis for the IEs identified in Section 4.1.

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The functional event tree is built starting from three critical functions as shown in Table 4-1:

1. Reactivity control
2. Heat removal
3. Containment of radioactive material

Table 4-1. Functional Event Tree Top Events

Initiating Event	Reactivity Control	Heat Removal	Containment	End State
The IEs identified in Section 4.1 above are processed through the functional event tree	The evaluated eVinci Micro-Reactor design has three strategies for reactivity control: • CDS • ESS • The passive release of hydrogen from the moderator*	The evaluated eVinci Micro-Reactor design includes two strategies for heat removal: • Heat removal via the secondary side system • Conduction through the core block to the canister with natural draft heat removal from the outside surface of the canister to an air duct system that channels air to the surrounding environment.	The evaluated eVinci Micro-Reactor design relies on the CCS for the containment function.	As the figure of merit for the evaluated eVinci Micro-Reactor risk assessment is a release frequency, even success states can result in limited releases.

Note: As the reference design changed during the course of the eVinci Micro-Reactor LMP Demonstration Project, it was assumed that the third reactivity control mechanism is not present.

The simplicity of the eVinci Micro-Reactor design allows the four event trees that were initially envisioned to address each of the IEs to be collapsed into one single event tree structure, with all the system dependencies directly captured in the mitigation logic (i.e., the loss of the PCS system as an IE is directly reflected in the mitigation logic but the event tree for loss of PCS maintains a PCS header). Figure 4-1 illustrates this concept.

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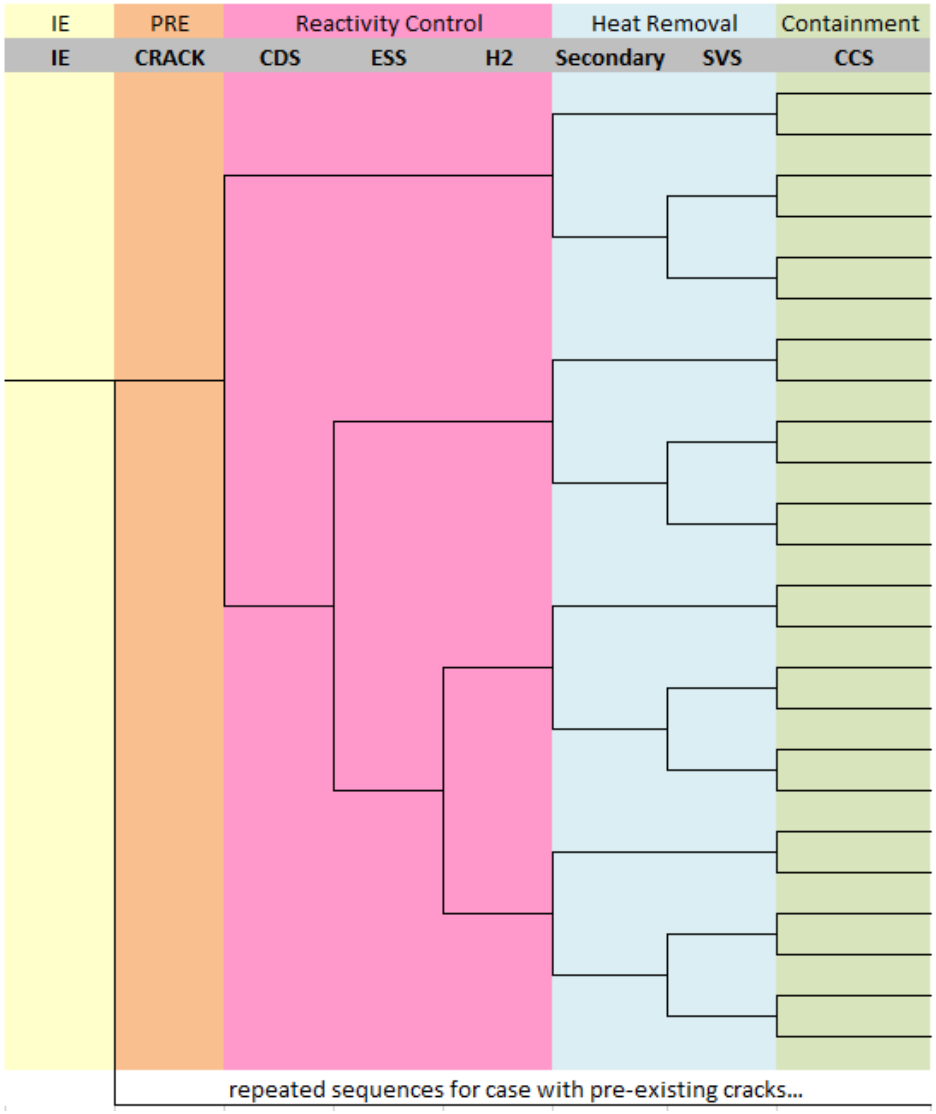

Figure 4-1. Generic Functional Event Tree for eVinci Micro-Reactor

For two IEs (i.e., failure of the heat exchanger and multiple ruptures of the heat pipe channels), the event tree is repeated to address two situations:

- A nominal failure situation (i.e., failure of the heat exchanger interface or of the heat channel impacting an intact heat channel)
- The potential for a situation where the IE impacts a faulted heat pipe channel (i.e., a heat pipe channel with a potential pre-existing condition where the monolith fabrication resulted in imperfections that resulted in a fuel channel pre-existing leak to a heat pipe channel). This pre-existing leak has the potential to impact the release dose estimate, even for sequences that are successfully mitigated.

An assumption has been made that the eVinci Micro-Reactor would continue to operate with one failed heat pipe channel; therefore, the possibility of a pre-existing failure that results in up to four fuel channels in communication with a single heat channel is included in the event tree

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sequence analysis. A pre-existing failure of this type would result, at a minimum, in different releases for the success sequences for certain IEs.

For the eVinci Micro-Reactor design, there is no applicable definition of core damage per se, and therefore the risk metric selected for the evaluation goes directly to a release frequency (i.e., dose release frequency). Different levels of potential challenge to the core are envisioned based on the temperature transients that are envisioned, which may or may not result in actual "damage" of the core in a classical definition. A cumulative uncontrolled core challenge scenario is therefore defined when all decay heat removal capabilities are lost or there is a loss of reactivity control function.

System fault trees were developed for the CDS, ESS, PCS, SVS, and the CCS to support the PRA model quantification.

Due to the passive nature of the eVinci Micro-Reactor design primary mitigation strategies, the fault tree modelling for the systems did not screen out any failure mode based on low conditional probabilities. Some conservative failure modes (e.g., plugging of air duct openings) were retained while the design of the features is being finalized. For a number of components in the eVinci Micro-Reactor design, reliability data is not readily available, and generic failure rates (and/or common cause failure parameters) were used with applicability caveats and large expected uncertainties on the numerical values.

The current PRA quantifies the frequency of individual sequences and the frequency of a specific dose release.

4.3 Estimation of Source Terms and Consequences

4.3.1 Scoping Mechanistic Fission Product Source Term

A measure of the consequences of a reactor accident is the offsite dose at the site boundary. An important component of the dose calculation is the fission product source term that defines the magnitude, composition, and timing of the radionuclide release to the environment. The purpose of this section is to describe the scoping mechanistic fission product source term from the fuel matrix that is used to perform initial estimates of the offsite releases and consequences for licensing basis events for the eVinci Micro-Reactor.

4.3.1.1 Methodology

The fission product release fractions are calculated as a function of fuel temperature using the Oak Ridge National Laboratory (ORNL) Booth model. This model has been used previously for developing fission product releases from light water reactors in severe accident analysis codes such as MAAP4, MAAP5,[12]and MELCOR. The analysis assumes that the fuel is uranium dioxide.

The ORNL-Booth model is the latest diffusion fission product release model from ORNL using the classical single-atom diffusion equations and diffusion coefficients which are based on a wide range of experimental fission product release data. The model implementation to the

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eVinci Micro-Reactor is performed as described in the FPRELC subroutine in the MAAP5 user's manual.[12]

A fuel channel fission product gap fraction is calculated for the normal operational temperature of 650°C. For accident source terms, the fractional releases are summed up over heat-up and cooldown transients to give the final overall release fractions of the fission product groups. The heat-up and cooldown transients were defined using a computational fluid dynamics (CFD) model of the eVinci Micro-Reactor core[13] as discussed in Section 4.3.1.2.

4.3.1.2 Core Accident Temperature Transients

Fission product releases from the fuel matrix to the fuel channel gap are calculated for the normal operational temperature (normal gap fraction) and for three heat-up and cooldown transients. The eVinci Micro-Reactor core loss of heat sink accident temperature transient was calculated using the Reference [12] CFD model. The CFD temperature transient, repeated here as Figure 4-2, defines nominal heat-up and cooldown rates for the eVinci Micro-Reactor core. Peak temperatures are assumed for three shutdown modes: insertion of the melting shutdown rod (750°C), release of the hydrogen moderator (850°C), and a high temperature shutdown at 950°C. The heat-up and cooldown rates are assumed to be constant for each of the assumed peak temperatures. As heat transfer calculations typically have an uncertainty of approximately +/- 20%, temperature transients were calculated assuming heat-up and cooldown rates +/- 20% of the nominal heat-up and cooldown rates to develop uncertainty bands on the source term.

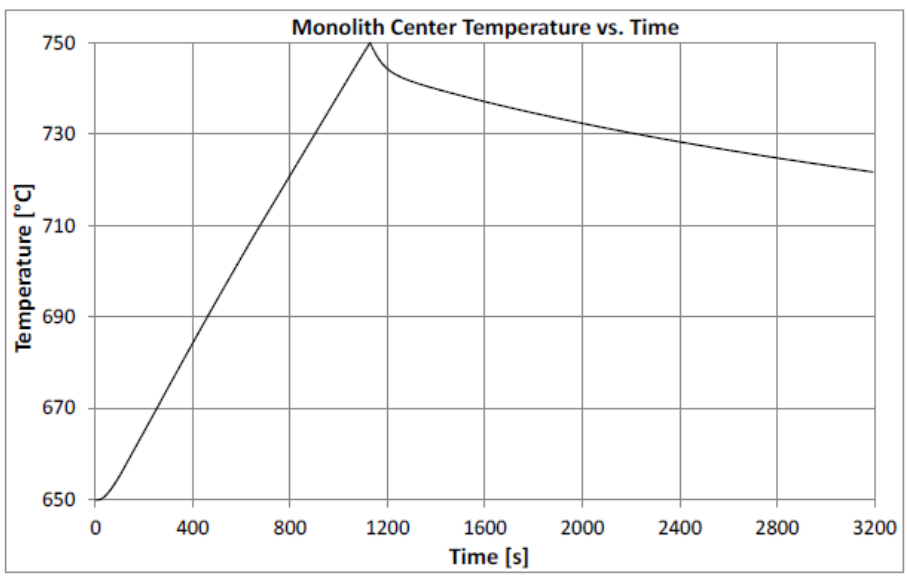

Figure 4-2. Monolith Temperature Transient During Loss of Secondary Cooling Accident

4.3.1.3 Fission Product Source Terms

The fission product source terms are defined as the fraction of the total fission product inventory that is released from the fuel matrix into the fuel channel gap. Shown in Table 4-2, these inventories are used to calculate the offsite releases for the dose calculation, as described in Section 4.3.2, based on the accident scenario temperature transient, the number of available fission product barriers, and their respective leak rates.

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Table 4-2. eVinci Micro-Reactor Fission Product Source Terms

	Gap Fraction	Heat-Up and Cooldown Release Fractions including the Gap Fraction								
		Peak Temp = 750°C			Peak Temp = 850°C			Peak Temp = 950°C		
		Min	Nom	Max	Min	Nom	Max	Min	Nom	Max
Species	Duration	2.3 hrs	2.8 hrs	3.5 hrs	4.3 hrs	5.3 hrs	6.4 hrs	6.2 hrs	7.6 hrs	9.3 hrs
Nobles	1.7E-04	5.0E-04	5.7E-04	6.6E-04	3.1E-03	3.6E-03	4.5E-03	1.8E-02	2.1E-02	2.7E-02
I	1.4E-04	4.0E-04	4.5E-04	5.3E-04	2.4E-03	2.9E-03	3.6E-03	1.4E-02	1.7E-02	2.1E-02
Cs	1.4E-04	4.6E-04	5.3E-04	6.2E-04	3.0E-03	3.5E-03	4.4E-03	1.8E-02	2.1E-02	2.6E-02
Sr	1.7E-06	5.0E-06	5.7E-06	6.6E-06	3.1E-05	3.6E-05	4.5E-05	1.8E-04	2.2E-04	2.7E-04
Mo	4.4E-05	1.3E-04	1.4E-04	1.7E-04	7.6E-04	9.1E-04	1.1E-03	4.5E-03	5.4E-03	6.7E-03
Ba	3.5E-06	1.0E-05	1.1E-05	1.3E-05	6.1E-05	7.3E-05	8.9E-05	3.6E-04	4.3E-04	5.4E-04
La	3.5E-08	1.0E-07	1.1E-07	1.3E-07	6.1E-07	7.3E-07	9.0E-07	3.6E-06	4.3E-06	5.4E-06
Ce	3.5E-08	1.0E-07	1.1E-07	1.3E-07	6.1E-07	7.3E-07	9.0E-07	3.6E-06	4.3E-06	5.4E-06
Sb	8.7E-05	2.5E-04	2.8E-04	3.3E-04	1.5E-03	1.8E-03	2.2E-03	9.0E-03	1.1E-02	1.3E-02
Te	1.4E-04	4.0E-04	4.5E-04	5.3E-04	2.4E-03	2.9E-03	3.6E-03	1.4E-02	1.7E-02	2.1E-02
Ru	8.7E-06	2.5E-05	2.8E-05	3.3E-05	1.5E-04	1.8E-04	2.2E-04	9.1E-04	1.1E-03	1.4E-03

4.3.2 Radiological Consequences Calculations

4.3.2.1 Description of Analyses

Using the fission product source terms described in Section 4.3.1, upper bound radiological consequences were evaluated using the assumptions in Table 4-3. Varying combinations of assumed release paths and activity releases were included in the evaluation such that the radiological consequences associated with the range of IEs and combinations of available mitigation could be determined. The activity release scenarios also considered two thermal power output levels, 1 MWt and 14 MWt, to support analysis on the sensitivity of the parameter. The release path combinations are those identified in Table 4-4 with the activity release scenarios as follows:

a) Temperature Excursion to Peak of 950°C, all fuel channels release

b) Temperature Excursion to Peak of 850°C, all fuel channels release

c) Temperature Excursion to Peak of 750°C, all fuel channels release

d) Temperature Excursion to Peak of 750°C, four fuel channels release (applies to both 1 MWt and 14 MWt)

e) Gap Release from four Fuel Channels (applies to both 1 MWt and 14 MWt)

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Table 4-3. Summary of Dose Assumptions

Description	Value (units)	Notes
Nuclide Information		
Core Nuclide Inventories	See notes	Inventories at Shutdown were generated for a 1 MWt nominal core. These are increased by a factor of 14 for the 14 MWt design.
Nuclide Chemical Forms	See notes	All nuclides were assumed to be particulate, except noble gases. Iodine volatilizes from aqueous solutions in the presence of low pH (excess hydrogen) and excess oxygen. The eVinci Micro-Reactor has no water, so iodine oxidation and subsequent volatilization is judged to not be a concern.
Dose Conversion Factors		Dose conversion factors are taken from US Environmental Protection Agency (EPA) Federal Guidance Reports 11[14] and 12[15] consistent with analysis that follows RG 1.183.[16]
Atmospheric Assumptions		
Atmospheric Dispersion Factor (X/Q)	1 sec/m^3	The assumed value of 1 sec/m^3 is a physical maximum, representing no dispersion. This is representative of <1 meter of distance between the source and the dose receptor.
Breathing Rate	3.5E-04 m^3/s	Standard NRC Dose Analysis Assumption
Geometry Assumptions		
Monolith Volume	1 ft^3	Arbitrary value as the leak rate is 1%/day.
Maximum Monolith Leak Rate	0.001 wt%/day	It is anticipated that the Canister is maintained at a positive pressure relative to the Monolith, even during accident conditions. Thus, any leakage out of the Monolith is a non-physical conservative assumption.
Monolith Activity Removal Rate	N/A	No removal was assumed in the monolith.
Canister Volume	1 ft^3	Arbitrary value as the leak rate is 1%/day.
Maximum Canister Leak Rate	0.001 wt%/day	The standard value for a large dry containment at a 60 psig design pressure is 0.1 weight-%/day. This is expected to be conservative with respect to the leak rate from the Canister.
Maximum Canister Activity Removal Rate	N/A	No removal was assumed in the Canister.
SVS Volume	1 ft^3	Arbitrary value as the leak rate is 1%/day.
Maximum SVS Leak Rate	0.001 wt%/day	The standard value for a large dry containment at a 60 psig design pressure is 0.1 wt%/day. This is expected to be conservative with respect to the leak rate from the SVS.
Maximum SVS Activity Removal Rate	N/A	No removal was assumed in the SVS.
Fuel Damage Assumptions – Scenario 1		
Activity Release Fractions	See notes.	Fission product release fractions were calculated for postulated monolith isothermal heat up using the ORNL-Booth model for releases from uranium dioxide fuel. Peak temperatures were assumed at 750°C, 850°C, and 950°C, along with minimum, nominal, and maximum heat up and cooldown rates.
Number of Fuel Channels	378 (1 MWt), 4219 (14 MWt)	The number of fuel channels does not scale linearly with power.

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Table 4-4. Release Path Combinations

	Case 1	Case 2	Case 3
Fission Product Barriers	1 barrier	2 barriers	3 barriers
Monolith to Canister	N/A	N/A	0.001%/day
Canister to SVS	N/A	0.001%/day	0.001%/day
SVS to Environment	0.001%/day	0.001%/day	0.001%/day

Each of the activity release scenarios was also examined considering the effects of variations in heat-up and cooldown rates discussed in Section 4.3.1.2. Note that the gap releases are the same for all scenarios.

4.3.2.2 Results

The resulting radiological consequences determined based on the approach discussed in Section 4.3.2.1 are shown in Table 4-5 and Table 4-6 for the 1-MWt and 14-MWt cases, respectively. It should be noted that the numerical results therein are in units of rem (Total Effective Dose Equivalent [TEDE]) and are calculated for 30 days with no environmental dispersion (i.e., representative of a distance between the source and the dose receptor of <1 m).

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Table 4-5. Doses for 1-MWt Reactor

	Case 1			Case 2			Case 3		
FP Barriers	1 barrier			2 barriers			3 barriers		
Leak Rate	0.001%/day			0.001%/day			0.001%/day		
Section 4.3.1.2 Temperature Cases	Minimum	Nominal	Maximum	Minimum	Nominal	Maximum	Minimum	Nominal	Maximum
Activity Release Scenario	30-day Dose at <1 m (rem TEDE)			30-day Dose at <1 m (rem TEDE)			30-day Dose at <1 m (rem TEDE)		
a	3.85E+00	4.82E+00	5.59E+00	5.00E-04	6.44E-04	7.25E-04	4.74E-08	6.15E-08	6.84E-08
b	6.25E-01	7.22E-01	1.14E+00	8.15E-05	9.59E-05	1.54E-04	7.73E-09	9.16E-09	1.48E-08
c	1.07E-01	1.15E-01	1.49E-01	1.41E-05	1.50E-05	2.00E-05	1.35E-09	1.42E-09	1.91E-09
d	1.43E-03	1.53E-03	1.99E-03	1.88E-07	1.99E-07	2.66E-07	1.80E-11	1.89E-11	2.55E-11
e	5.11E-04	5.11E-04	5.11E-04	6.65E-08	6.65E-08	6.65E-08	6.33E-12	6.33E-12	6.33E-12

Table 4-6. Doses for 14-MWt Reactor

	Case 1			Case 2			Case 3		
FP Barriers	1 barrier			2 barriers			3 barriers		
Leak Rate	0.001%/day			0.001%/day			0.001%/day		
Section 4.3.1.2 Temperature Cases	Minimum	Nominal	Maximum	Minimum	Nominal	Maximum	Minimum	Nominal	Maximum
Activity Release Scenario	30-day Dose at <1 m (rem TEDE)			30-day Dose at <1 m (rem TEDE)			30 Day Dose at <1 m (rem TEDE)		
a	5.39E+01	6.75E+01	7.83E+01	7.00E-03	9.01E-03	1.01E-02	6.63E-07	8.62E-07	9.58E-07
b	8.75E+00	1.01E+01	1.60E+01	1.14E-03	1.34E-03	2.15E-03	1.08E-07	1.28E-07	2.07E-07
c	1.50E+00	1.61E+00	2.09E+00	1.98E-04	2.09E-04	2.79E-04	1.89E-08	1.99E-08	2.68E-08
d	1.43E-03	1.53E-03	1.99E-03	1.88E-07	1.99E-07	2.66E-07	1.80E-11	1.89E-11	2.55E-11
e	5.11E-04	5.11E-04	5.11E-04	6.65E-08	6.65E-08	6.65E-08	6.33E-12	6.33E-12	6.33E-12

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4.4 Estimation of Event Sequence Frequencies

The event sequences discussed in Section 4.2 have been quantified for each of the consequence analysis cases calculated in Section 4.3. Details of the dose case quantifications are shown in Table 4-7.

Table 4-7. Dose Case Quantification

Dose Case	Consequence Case*	Sequences	Description
1	e-3	SEQ-01, SEQ-03	Any IE followed by successful trip via CDS and successful heat removal via PCS. Containment is intact (nominal leakages only).
2	e-2	SEQ-02, SEQ-04, SEQ-21, SEQ-23	Any IE followed by successful trip via CDS and successful heat removal via PCS. Containment is faulted (i.e., spurious opening of safety valves). This includes cases with and without a pre-existing crack in the fuel to heat pipe channel, for which the associated fission products may be released.
3	b-2	SEQ-05, SEQ-11, SEQ-25, SEQ-31	Any IE followed by successful trip via CDS or ESS with failed decay heat removal from both PCS and SVS(2). Containment is intact (nominal leakages only). This includes cases with and without a pre-existing crack in the fuel to heat pipe channel, for which the associated fission products may be released.
4	b-1	SEQ-06, SEQ-12, SEQ-26, SEQ-32	Any IE followed by successful trip via CDS or ESS with failed decay heat removal from both PCS and SVS. Containment is faulted (i.e., spurious opening of safety valves). This includes cases with and without a pre-existing crack in the fuel to heat pipe channel, for which the associated fission products may be released.
5	c-3	SEQ-07, SEQ-09	Any IE followed by successful trip via ESS (CDS fails) with successful decay heat removal from either PCS or SVS. Containment is intact (nominal leakages only). These cases do not have a pre-existing crack in the fuel to heat pipe channel.
6	c-2	SEQ-08, SEQ-10	Any IE followed by successful trip via ESS (CDS fails) with successful decay heat removal from either PCS or SVS. Containment is faulted (i.e., spurious opening of safety valves). These cases do not have a pre-existing crack in the fuel to heat pipe channel.
7	a-2	SEQ-19, SEQ-39	Any IE followed by failure of reactivity control (both CDS and ESS, no credit for hydrogen release). Containment is intact (nominal leakages only). This includes cases with and without a pre-existing crack in the fuel to heat pipe channel, for which the associated fission products may be released.
8	a-1	SEQ-20, SEQ-40	Any IE followed by failure of reactivity control (both CDS and ESS, no credit for hydrogen release). Containment is faulted (i.e., spurious opening of safety valves). This includes cases with and without a pre-existing crack in the fuel to heat pipe channel, for which the associated fission products may be released.
9	e-1	SEQ-22, SEQ-24	Any IE followed by successful trip via CDS with successful decay heat removal via either PCS or SVS. Containment is faulted (i.e., spurious opening of safety valves). These cases have a pre-existing crack in the fuel to heat pipe channel.
10	d-2	SEQ-27, SEQ-29	Any IE followed by successful trip via ESS (CDS fails) with successful decay heat removal via either PCS or SVS. Containment is intact (nominal leakages only). These cases have a pre-existing crack in the fuel to heat pipe channel.
11	d-1	SEQ-28, SEQ-30	Any IE followed by successful trip via ESS (CDS fails) with successful decay heat removal via either PCS or SVS. Containment is faulted (i.e., spurious opening of safety valves). These cases have a pre-existing crack in the fuel to heat pipe channel.

*See Section 4.3 for a discussion on the consequence cases.

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The quantification results are summarized in Table 4-8 for each dose case.

Table 4-8. Dose Case Frequency

Dose Case	Frequency
1	1 E-00
2	3 E-04
3	1 E-03
4	1 E-07
5	1 E-03
6	1 E-07
7	1 E-07
8	1 E-11
9	1 E-10
10	3 E-10
11	4 E-14

Each of the dose cases was then plotted on a consequence versus frequency diagram as input to the classification task discussed in Section 5. Figure 4-3 shows the base case dose cases quantified from the eVinci Micro-Reactor PRA. A more refined quantification is possible that breaks apart each dose case in specific IEs and in specific sequences.

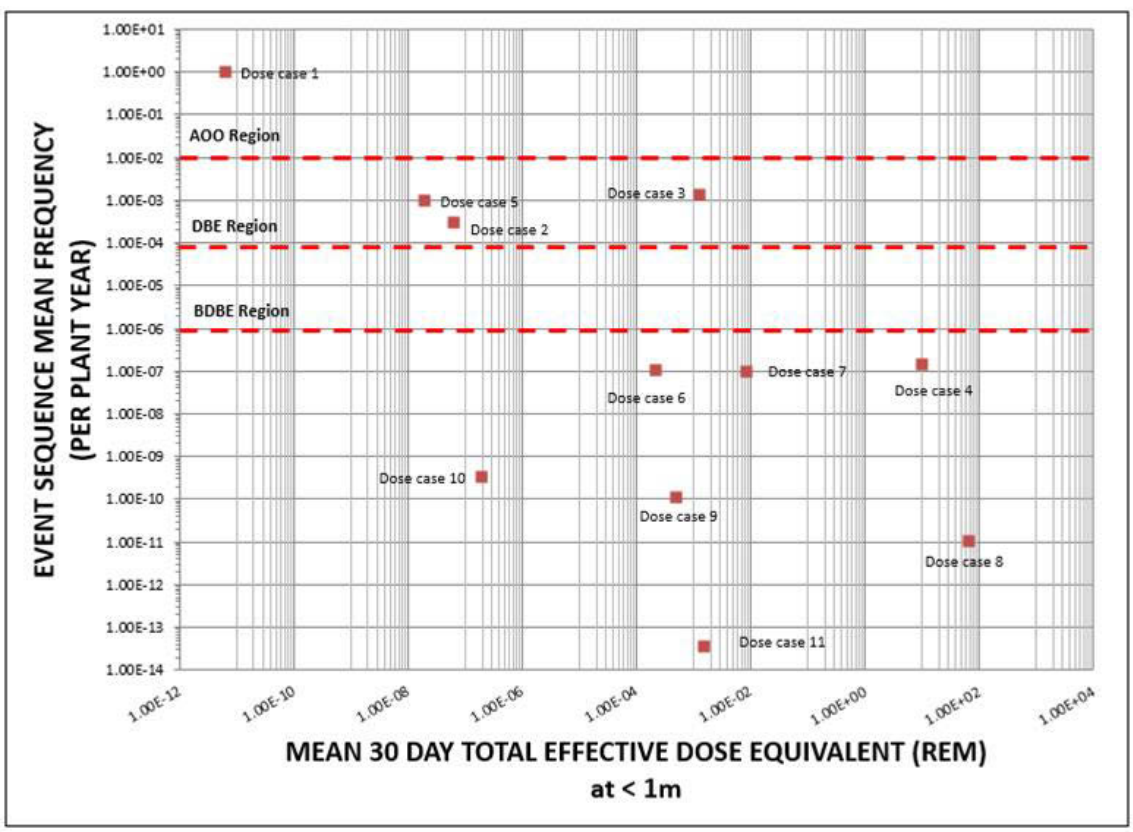

Figure 4-3. F-C Targets (Base Case Only)

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4.5 PRA Model Assumptions, Limitations, and Uncertainty Assessment

Due to the stage of design development for the eVinci Micro-Reactor LMP Demonstration Project, a large number of key assumptions were made during the PRA model development to allow for a working model to be generated. These assumptions spanned across every PRA technical element, from component failure modes to IE frequencies, from control and protection logic design to component failure consequences, from system design features to operator actions, maintenance frequencies, and latent failures. Each assumption made by the PRA developer is reviewed and agreed upon with the design team and an estimation of the potential associated uncertainty is discussed.

Sensitivity cases were quantified to gain an initial understanding of some of the uncertainties inherent in the analysis. The sensitivity cases included failing modeled operator actions, changing the redundancy of the ESS, increasing the redundancy of the SVS ducting, modifying the IE frequency for the multiple heat pipe seal rupture event, decreasing the IE frequency for the spurious reactor trip event, changing the CCS relief strategies by using different kind of devices, setting the IE frequency for the loss of the PCS at a system reliability target, modifying the surveillance frequency for the SVS duct air intake, increasing the probability of a pre-existing crack in one core channel, modifying the IE frequency for the primary heat exchanger tube rupture event, and adding a potential failure mode to the ESS. The overall result of the sensitivities is that each point in the F-C diagram presented in the previous Figure 4-3 is modified into a frequency band for each dose case, as shown in Figure 4-4.

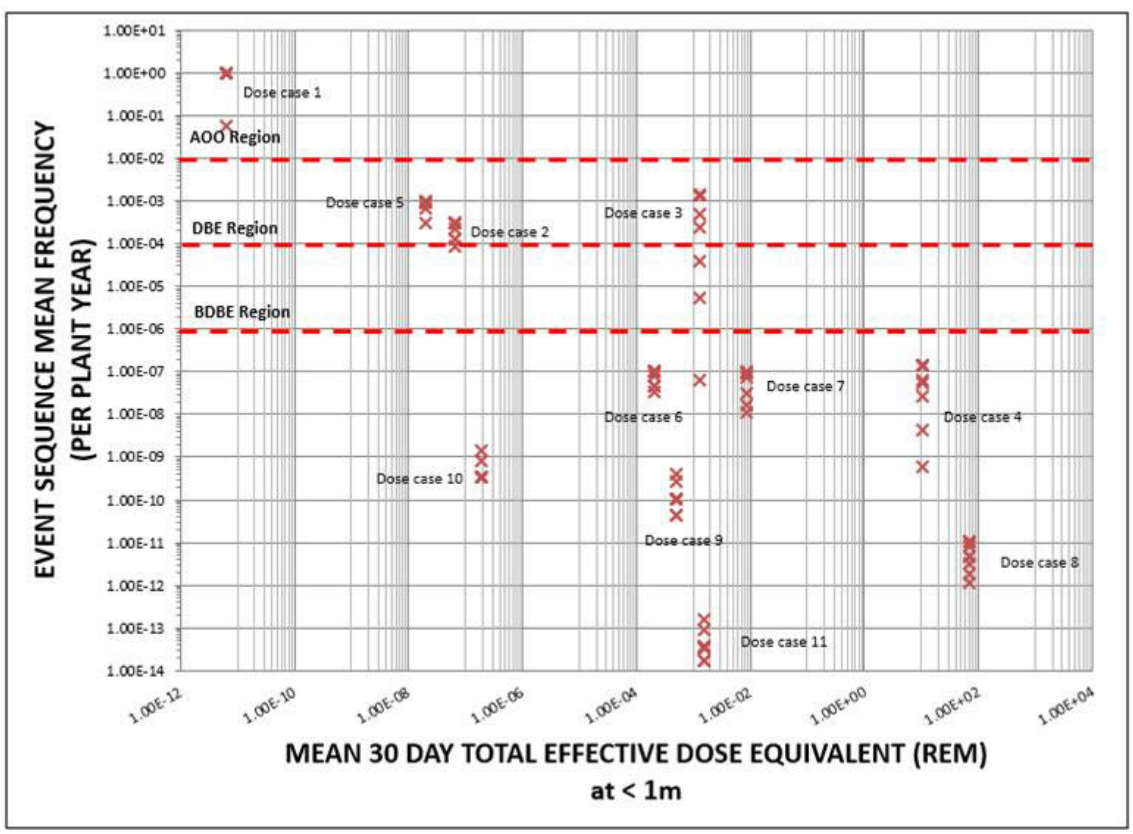

Figure 4-4. F-C Targets (Frequency Uncertainties)

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Similar uncertainties are also tracked in the dose calculations, where uncertainties are captured by different behaviors associated with the temperature transients. As such, these uncertainties are essentially constant for each dose case and, therefore, they are not reported explicitly in Figure 4-4.

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5 SELECTION AND EVALUATION OF LICENSING BASIS EVENTS

5.1 Identification of AOOs, DBEs, and BDBEs

As discussed in Section 4, a grouping of IEs was performed with four unique IEs selected and explicitly modelled in the PRA. Due to the simplicity and the early design phase of the eVinci Micro-Reactor, some plant behavior differences cannot be recognized yet. Therefore, only one event tree has been developed capturing all of the accident sequences originating from the four IEs. From this event tree, individual accident sequences have been grouped into 11 dose cases (Table 5-1) that represent LBEs based on the specific radiological consequences from the IE and sequence of failures.

Table 5-1. Dose Case Groupings into AOOs, DBEs, and BDBEs

AOO	DBE	BDBE
Dose case 1	Dose case 2	Dose case 4
	Dose case 3	Dose case 8
	Dose case 5	Dose case 6
		Dose case 7
		Dose case 9
		Dose case 10
		Dose case 11

Section 5.2 describes and depicts the mapping of individual LBEs on the frequency vs. consequence diagram. The AOOs, DBEs and BDBEs thresholds used for eVinci use the same numerical values suggested in Reference [1]:

- AOOs: LBEs with mean frequencies greater than 1×10^{-2}/plant-year
- DBEs: LBEs with mean frequencies between 1×10^{-4} and 1×10^{-2}/plant-year
- BDBEs: LBEs with frequencies less than 1×10^{-4} but with upper bound frequencies greater than or equal to 5×10^{-7}/plant-year

5.2 Evaluation of LBEs Against F-C Target

As part of the PRA development efforts (Section 4), an estimation of the source term and dose consequences from failures of certain safety functions was performed. With the dose consequence for each defined dose case evaluation of the LBEs against the F-C Target shown in Figure 3-2 can be performed as shown in Figure 5-1.

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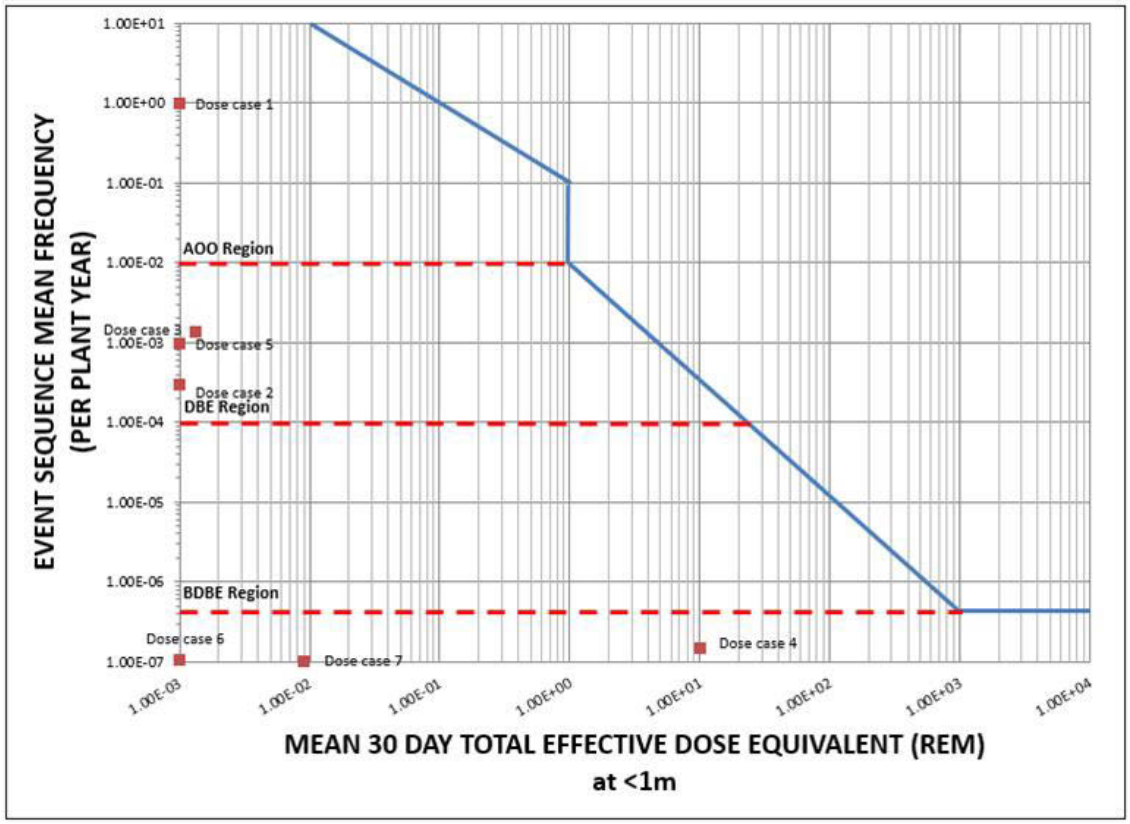

Figure 5-1. Frequency-Consequence Plot for the eVinci Micro-Reactor (in LMP Resolution)

There are some observations to note from this plot.

- In the LMP, frequencies and consequences are truncated at physically meaningful values (i.e., consequences greater than 1E-03 rem and frequencies greater than 1E-07). Figure 5-1 is reported with this resolution. With this resolution, the majority of the dose cases fall on the "zero dose" or on the "zero frequency" region.

- Only Dose Case 3 and Dose Case 4 have meaningful values in the region of interest of the diagram within the LMP context. As a conservative approach for this evaluation, Dose cases are categorized in the DBE category if the frequency of the event meets the criterion, although the dose may be lower than the criterion for consideration in LMP. For the BDBE cases, any dose case with an event frequency below the DBE threshold is evaluated as a BDBE, even if the event frequency is lower than the threshold for BDBE categorization. This meets the intent of the NRC position in DG-1353[17] discussing that the F-C Target figure does not depict acceptance criteria or regulatory limits and that the target provides an approach for use within a broader, integrated approach to support safety classification.

5.3 Identification of Required Safety Functions

Due to the simplicity of the eVinci Micro-Reactor design, the PRA Safety Functions (PSFs) identified in the PRA effort (Section 4) are minimal and the PSFs are also identified as the Required Safety Functions (RSFs). Therefore, the identified RSFs are:

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- Reactivity Control
- Decay Heat Removal
- Containment of Radioactive Material

The evaluations performed and presented in Section 6.3 establish that these RSFs are needed to keep the DBEs and high consequence BDBEs inside the F-C Target and to prevent core challenge.

5.4 Selection of Safety-Related SSCs

The selection of SR SSCs is discussed in Section 6 of this report.

5.5 Definition of DBAs

Section 4.5 describes that the PRA model developed provides an initial conservative representation for at-power internal events only . For full LMP implementation, the model requires an update to address final design features of the eVinci Micro-Reactor in addition to consideration of the full set of IEs (including external hazards). As a result, definition of the full set DBAs will be defined later as the plant design and licensing for the eVinci Micro-Reactor matures.

5.6 Identification of Risk Insights Regarding eVinci Micro-Reactor Design Features

As discussed in Section 4, epistemic uncertainties (i.e., model uncertainties) are currently expected to be dominating the risk insights. A number of sensitivities were performed to address these associated uncertainties, which are discussed in further detail in Section 4.5.

5.7 LBE Selection and Evaluation Assumptions and Limitations

Final selection and evaluation of LBEs separate from the LBEs defined from the PRA model in Section 4 will be performed as the plant design and licensing for the eVinci Micro-Reactor matures. Major assumptions and limitations for the current set of LBEs are discussed in Section 4.5.

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6 SSC SAFETY CLASSIFICATION AND PERFORMANCE CRITERIA

This section describes the efforts performed to classify the eVinci Micro-Reactor systems, structures, and components (SSCs) into one of the following safety classification categories:

- 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 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—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).

The classification of SSCs into these categories used the process shown in Figure 1-2, which involves seven tasks using input from the PRA and LBE evaluation (Sections 4 and 5).

6.1 Identification of SSC Functions in the Prevention and Mitigation of LBEs

The three critical functions identified and evaluated in the PRA (Section 4):

- Reactivity Control
- Decay Heat Removal
- Containment of Radioactive Material

The PRA identified that these functions are capable of mitigating the LBEs identified.

6.2 Identification and Evaluation of SSC Capabilities and Programs to Support DID

At this stage of the design, SSC capabilities to perform the required safety functions are identified on a system and subsystem level and not a component level. Table 6-1 shows the available SSCs in the PRA that perform the PSFs identified in Task 1 (Section 6.1).

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Table 6-1. SSCs Performing PRA Safety Functions

PRA/SSC Function	Available SSCs
Reactivity Control	Control Drum Subsystem Emergency Shutdown Subsystem The passive release of hydrogen from the moderator
Decay Heat Removal	The use of the heat channels (Core Block Subsystem [CBS]) through the secondary side system (PCS) The reliance on conduction of heat through the CBS to the canister and heat exchange through CCS air ducts per natural draft of the air over the surface of the canister through the SVS to the environment
Containment of Radioactive Material	Canister Containment Subsystem

It is noted that while only one SSC is considered for availability in performing the containment of radioactive material function, the eVinci Micro-Reactor design described in Section 3.2 provides three barriers to ensure events do not allow the radioactive release of material:

1. Monolith Encapsulation of Fuel
2. Solid Core Block
3. Canister Containment Subsystem

The CCS is the only system for which the PRA addresses specific failure modes that are included in the logic model. Uncertainties associated with the behavior of the other two completely passive barriers (i.e., the monolith encapsulation of the fuel, and the solid core block) are implicitly addressed in the dose release assumptions and calculations discussed in Section 4.5.

6.3 Identification of Required Safety Functions

In the PRA (Section 4), event sequences were developed and quantified with the system failures assumed in the analysis for the overall Cumulative Core Challenge Scenario Frequency along with each of the consequence analysis cases. The identified accident sequences were assembled into top events associated with specific radiological consequences. These are identified as "Dose Cases" in the PRA. Section 5.1 describes the PRA efforts to identify the dose cases that fall into the AOO, DBE or BDBE thresholds for frequencies. Section 5.2 then compares each dose case's frequency and consequence. Figure 5-1 shows the F-C Target results for each of the dose cases identified.

The different DBE dose cases and high consequence BDBE cases can be evaluated to determine the required safety functions to meet the F-C Target. The different cases evaluated for RSF identification are shown in Table 6-2. This table identifies core challenge (CC) as an unacceptable event. Even though the F-C Target may still be met if a CC occurs, it is assumed in this Demonstration Project that RSFs are required to prevent CC.

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Table 6-2. Credited Systems for PRA DBEs and High Consequence Dose Cases / Sequences

Dose Case	Sequence #	Reactivity Control		Heat Removal		Containment		PRA Result
		CDS	ESS	H_2	PCS	SVS	CCS	
2	SEQ-02	X			X			OK
2	SEQ-04	X				X		OK
3	SEQ-05	X					X	CC
4	SEQ-06 (BDBE)	X						CC
3	SEQ-11		X				X	CC
4	SEQ-12 (BDBE)		X					CC
5	SEQ-07		X		X		X	OK
6	SEQ-08 (BDBE)		X		X			OK
5	SEQ-09		X			X	X	OK
6	SEQ-10 (BDBE)		X			X		OK
8	SEQ-20 (BDBE)			X				CC

For Dose Case 3, SEQ-05 demonstrates that reactivity control from CDS plus containment is not sufficient to eliminate the DBE consequence. To eliminate the DBE consequence, decay heat removal is required, since core damage is the result of the sequence. SEQ-02 and SEQ-04 (Dose Case 2) demonstrate that either PCS or SVS (with CDS), mitigates the dose consequence to below the DBE threshold (without crediting containment). As shown in Table 6-2, SEQ-02 credits the PCS and SEQ-04 credits heat removal via the SVS ducts.

For Dose Case 3, SEQ-11 demonstrates that reactivity control from ESS plus containment is not sufficient to eliminate the DBE consequence. To mitigate the DBE consequence, decay heat removal is required. SEQ-08 and SEQ-10 (Dose Case 6) demonstrate that either PCS or SVS (with ESS) mitigates the dose consequence; however, the dose consequence remains within the DBE threshold without credit for containment.

As shown in Table 6-2, SEQ-08 credits the PCS and SEQ-10 credits the SVS for decay heat removal. To eliminate the DBE dose consequence, containment is required. Per Table 6-2, SEQ-07 and SEQ-09 (Dose Case 5) demonstrate that with ESS and decay heat removal by either PCS or SVS, the dose consequence is below the DBE threshold with containment credited.

From review of the PRA sequences and the evaluation performed, it is concluded that the following functions are RSFs:

- Reactivity Control
- Decay Heat Removal
- Containment of Radioactive Material

As discussed in Section 5.3, the PSFs identified in the PRA effort are minimal due to the simplicity of the eVinci Micro-Reactor design. Therefore, the PSFs identified are also identified as the RSFs capable of mitigating the LBEs.

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6.4 Selection and Classification of Safety-Related SSCs

To identify the SR SSCs, each of the RSFs identified in Section 6.3 is evaluated in the following subsections. To evaluate the safety-significance of the systems which perform each RSF, individual sequence information from the PRA (Section 4) is utilized.

6.4.1 Reactivity Control Function Evaluation

Reactivity control can be performed by three systems:

- Control Drum Subsystem
- Emergency Shutdown Subsystem
- The passive release of hydrogen from the moderator

The ESS is a Passive Category B system per IAEA-TECDOC-626.[18] Due to the reliability of its passive features, it is selected as the safety-related system to perform the RSF of reactivity control. Therefore, it must be demonstrated in the PRA results that the ESS is capable of performing this RSF for all DBEs.

Dose Case 3 is identified as a DBE and represents the failure of the decay heat removal systems to mitigate an IE. Only reactivity control and containment are available for mitigation. Using SEQ-05 and SEQ-11 from Dose Case 3, CDS and ESS can be compared as shown in Table 6-3. SEQ-05 credits CDS and containment, while SEQ-11 credits ESS and containment. Since these sequences are grouped in the same dose case, there is not a significant difference in dose between the sequences. In evaluating the frequency by initiator for SEQ-11, which credits ESS, the event frequencies are all lower than those in SEQ-05, where the CDS is credited. This demonstrates the higher reliability of the passive safety system, ESS, for mitigation of a DBE.

Table 6-3. Comparison of Dose and Event Sequence Frequencies for Dose Case 3 Sequences

SEQ	Dose (TEDE rem)	Trip only	PHXTR only	CPS IE only	HRUPT only	CDS	ESS	H_2	PCS	SVS	CCS
05	1.54E-04	2.9E-6	2.7E-6	1.3E-3	9.8E-15	X					X
11	1.54E-04	1.0E-9	9.6E-10	4.7E-7	0		X				X

Notes:
PHXTR—Passive Heat Exchanger Tube Rupture
HRUPT—Heat Pipe Rupture-Multiple
H_2—Hydrogen Release

Dose Case 4 is identified as a high consequence BDBE. SEQ-06 credits the CDS for reactivity control, no decay heat removal, and no containment. SEQ-12 credits the ESS, with no decay heat removal and no containment. In evaluating the frequency by initiator for SEQ-12, which credits ESS in a similar manner Dose Case 3 sequences were compared, the event frequencies are all lower than those in SEQ-06. This demonstrates the higher reliability of the passive safety system, ESS, for mitigation of a BDBE.

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Dose Case 8 is identified as a high consequence BDBE. The passive release of hydrogen from the moderator is only credited in SEQ-19 and SEQ-20. This passive release of hydrogen coupled with containment has not demonstrated successful mitigation of core challenge. Therefore, this system cannot be credited as the only safety-related means of reactivity control based on the available PRA data. Future PRA evaluations of the passive hydrogen release with decay heat removal and containment are recommended to further evaluate the safety-classification of the ESS. Thus, the ESS is identified as the safety-related system for reactivity control.

6.4.2 Decay Heat Removal Function Evaluation

Decay Heat Removal can be provided by the following systems:

- The use of the heat channels (CBS) through the secondary side system (Power Conversion System [PCS])
- The reliance on conduction of heat through the CBS to the canister and heat exchange through air ducts per natural draft of the air over the surface of the canister (SVS)

There are no PRA cases which evaluated the success of a transient with the reactivity control systems failed, the containment failed and only the decay heat removal systems used for mitigation of the transient, so there cannot be a direct comparison of just the PCS to SVS for decay heat removal.

The ESS system has been identified for safety-related reactivity control as it is a passive system. The preference for safety-related heat removal is also the passive heat removal by the SVS. Therefore, Dose Case 5, which is identified as a DBE by its frequency, is evaluated. SEQ-07 credits ESS, PCS, and CCS. SEQ-09 credits the ESS, SVS, and CCS. The dose consequence from either SEQ-07 or SEQ-09 is approximately zero dose.

In comparison of the event frequencies for SEQ-07 with PCS and SEQ-09 with SVS (shown in Table 6-4), the frequency of the IEs for SEQ-09 are not less than those for SEQ-07 for all IEs.

Table 6-4. Comparison of Dose and Event Sequence Frequencies for Dose Case 5 Sequences

SEQ	Dose (TEDE rem)	Trip only	PHXTR only	CPS IE only	HRUPT only	CDS	ESS	H2	PCS	SVS	CCS
07	1.91E-09	2.8E-4	0	0	9.5E-13		X		X		X
09	1.91E-09	1.5E-6	1.4E-6	6.8E-4	4.9E-15		X			X	X

As noted in the PRA (Section 4), the failure of the heat removal functionality of the SVS has been conservatively modeled and is a significant contribution to the unrealistically high core damage frequency. The failure mode for plugging the SVS intake is retained in the model. There are no filters in the SVS intake, and it is expected that the opening will be large enough to avoid significant plugging. The failure rate (associated with filters) is higher than what would be realistic to assume for this design. This is the ultimate heat sink, and is therefore addressed conservatively at this point, while design refinements are implemented.

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The SVS is a Passive Category B system per Reference 17 and is therefore selected as the safety-related system for the decay heat removal function.

6.4.3 Containment Function Evaluation

As shown in Table 6-2, the CCS is the SSC that performs the PSF of containment. Therefore, the significance of the operation of CCS is evaluated using a comparison of individual sequences with and without CCS. Using these comparisons, the significance of CCS operation can be evaluated. The relevant sequence comparisons evaluated are as follows:

- SEQ-05 and SEQ-06: Reactivity Control using CDS, no Decay Heat Removal
- SEQ-11 and SEQ-12: Reactivity Control using ESS, no Decay Heat Removal
- SEQ-19 and SEQ-20: Reactivity Control with H_2, no Decay Heat Removal

Using the first comparison (SEQ-05 and SEQ-06), it is concluded that the CCS is needed to prevent the consequences of a DBE to within the F-C Target. SEQ-05 has a dose consequence of Dose Case 3 and SEQ-06 has a dose consequence of Dose Case 4. Figure 5-1 can be used to show that if CCS was unavailable for Dose Case 3, it would challenge the consequence to remain within the F-C Target using the dose consequence from Dose Case 4. Figure 6-1 illustrates this concept. Therefore, it is concluded that the containment function (CCS) is needed to maintain dose consequences within the F-C Target.

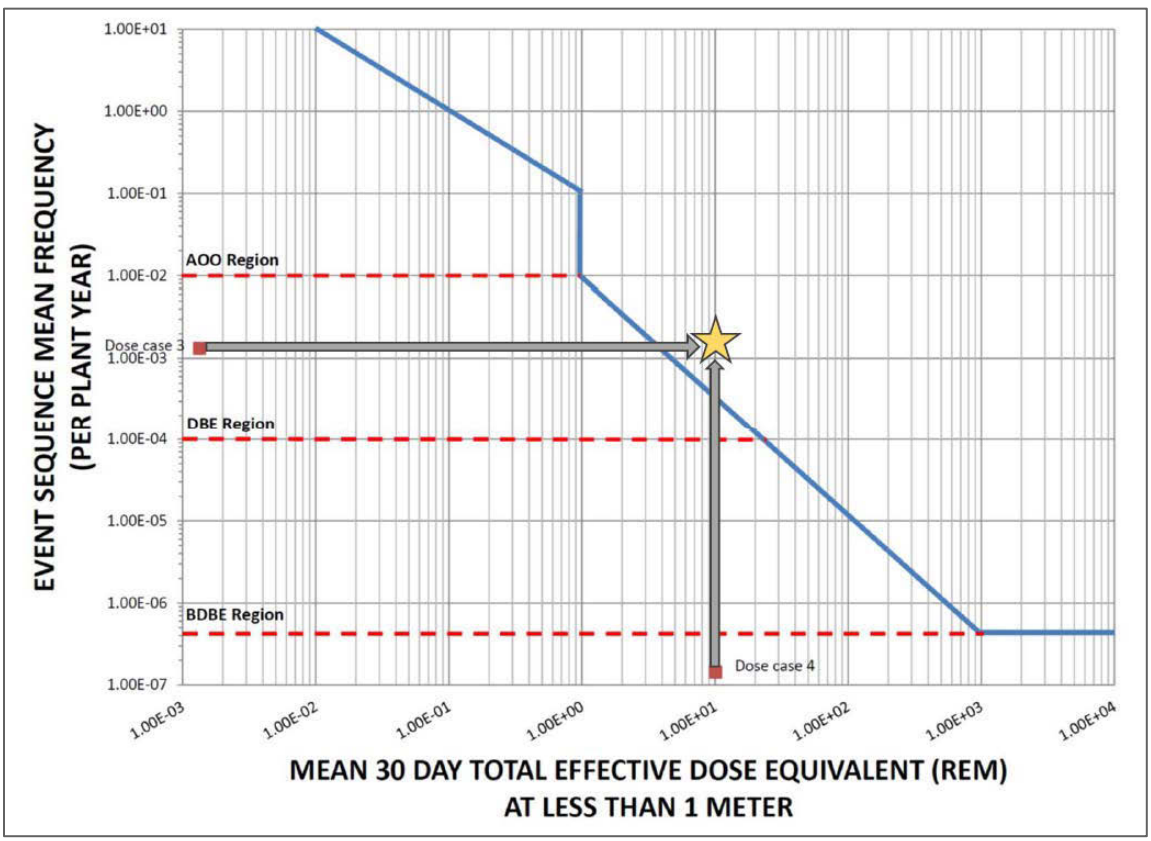

Figure 6-1. Comparison of Dose Case 3 and Dose Case 4 Showing Consequence without CCS

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6.4.4 Conclusion for Selection of Safety-Related SSCs

In Task 4A, each of the DBEs and any high-consequence BDBEs (those with doses above 10 CFR 50.34 limits) are examined to determine which SSCs are available to perform the RSF that covers all the DBEs and high-consequence BDBEs. These specific SSCs are classified as Safety-Related in Task 5A and are the only ones included in DBA analyses. All remaining SSCs are evaluated further in Tasks 4B and 4C.

At this stage of the PRA development, the following systems are identified as Safety-Related:

- Emergency Shutdown Subsystem
- Secure Vault Subsystem (air ducts)
- Canister Containment Subsystem

As demonstrated in SEQ-09, the combination of ESS, SVS, and CCS results in successful mitigation of all IEs considered.

The safety classification is provided at the system level at this stage of development and will progress to the SSC level as the eVinci design evolves and the PRA incorporates system level functions. It is expected that there will be components within these systems that will not be safety-related and therefore will require additional review by SSC classification Tasks 4B/5B and 4C/5C.

6.5 Identification and Classification of NSRST and NST SSCs

The following systems evaluated are not identified as Safety-Related in Section 6.4:

- Control Drum Subsystem
- Power Conversion Subsystem

These SSCs require additional review in Tasks 4B/5B and 4C/5C to determine if any SSCs are identified as NSRST or if they are identified as NST.

At this stage in the design/PRA development, there is not sufficient available information to perform a meaningful assessment on the component level. A DID analysis has not yet been performed for the current eVinci design.

As demonstrated in SEQ-09, the combination of ESS, SVS, and CCS results in successful mitigation of any of the IEs, by using the safety-related systems.

It is demonstrated in PRA sequence SEQ-01 that the mitigation by the CDS, PCS, and CCS successfully mitigates a core challenge for any of the IEs. Therefore, the CDS and PCS provide a defense-in-depth means of performing the functions of reactivity control and decay heat removal.

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6.5.1 Control Drum Subsystem Evaluation

6.5.1.1 CDS Evaluation for Safety Classification Task 4B and 5B: Risk-Significance Determination

To determine the risk significance of the CDS, the following question is addressed: Does the CDS maintain one or more LBEs within F-C Target?

With the containment intact, there is very little difference in the dose consequences if the CDS is credited as the reactivity control system versus the ESS. This is shown when evaluating the individual sequences within Dose Case 3, specifically SEQ-05 versus SEQ-11. SEQ-05 credits the CDS and SEQ-11 credits the ESS as shown in Table 6-3.

The ESS is a passive system and is shown in Table 6-3 that it has a higher reliability than the CDS for reactivity control. Therefore, the CDS is concluded as not risk significant.

The CDS shall be evaluated for DID adequacy in Safety Classification Task 4C and 5C.

6.5.1.2 CDS Evaluation for Safety Classification Task 4C and 5C: DID Adequacy of the CDS

Due to the unique nature of the ESS performance, which is based on a gate melting for passive actuation, the CDS may be identified as a DID system to prevent the ESS initiation.

The CDS operation provides reactivity control, which will prevent the ESS gate melting and actuation of the system. Therefore, the CDS does provide DID adequacy by providing a means of reactivity control which precludes the need for the ESS to perform the Safety-Related function of reactivity control.

Therefore, it is recommended that the SSCs within the CDS that perform this DID function are classified as NSRST. The function and safety classification of individual CDS components will be addressed later as the plant design and licensing for the eVinci Micro-Reactor matures.

6.5.2 Power Conversion Subsystem Evaluation

6.5.2.1 PCS Evaluation for Safety Classification Task 4B and 5B: Risk-Significance Determination

To determine the risk significance of the PCS, the following question is addressed: Does the PCS maintain one or more LBEs within F-C Target?

With the containment intact, there is very little difference in the dose consequences if the PCS is credited as the decay heat removal system versus heat removal via the SVS ducts. This is shown when evaluating the individual sequences within Dose Case 5, specifically SEQ-07 versus SEQ-09. SEQ-07 credits the PCS and SEQ-09 credits the SVS as shown in Table 6-4.

Heat removal via the SVS ducts relies on a passive system, and therefore, reliability of the system is expected to be higher than the PCS. In the current PRA evaluation, it is noted that the SVS failure rate is conservatively higher than expected. This is due to assumptions made in the PRA due to the preliminary nature of the air duct design. Once detail of the air ducts is further

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identified, it is expected that the SVS air ducts will provide a more reliable means of performing the decay heat removal function. Therefore, the PCS is concluded as not risk significant.

The PCS shall be evaluated for DID adequacy in Safety Classification Task 4C and 5C.

6.5.2.2 PCS Evaluation for Safety Classification Task 4C and 5C: DID Adequacy of the PCS

The PCS does provide a diverse means of decay heat removal. It is noted that operation of the PCS precludes the need for heat removal to be performed via the SVS ducts and decreases the overall reliance on the SVS ducts if operating. Additionally, operation of the PCS may preclude operation of the ESS, which will be further evaluated in future DID evaluations in future eVinci Micro-Reactor design efforts.

Therefore, it is recommended that the SSCs within the PCS that perform this DID function are classified as NSRST. The function of individual PCS components will be addressed later as the plant design and licensing for the eVinci Micro-Reactor matures.

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7 EVALUATION OF DID ADEQUACY

The DID aspects of the systems evaluated in the PRA have been initially considered as described in Section 6. This includes consideration of the Reference [1] guidance and process for incorporation and evaluation of DID, which is depicted in Figure 7-1. As the eVinci Micro-Reactor design matures, a detailed DID evaluation will be performed in future stages of the design development project.

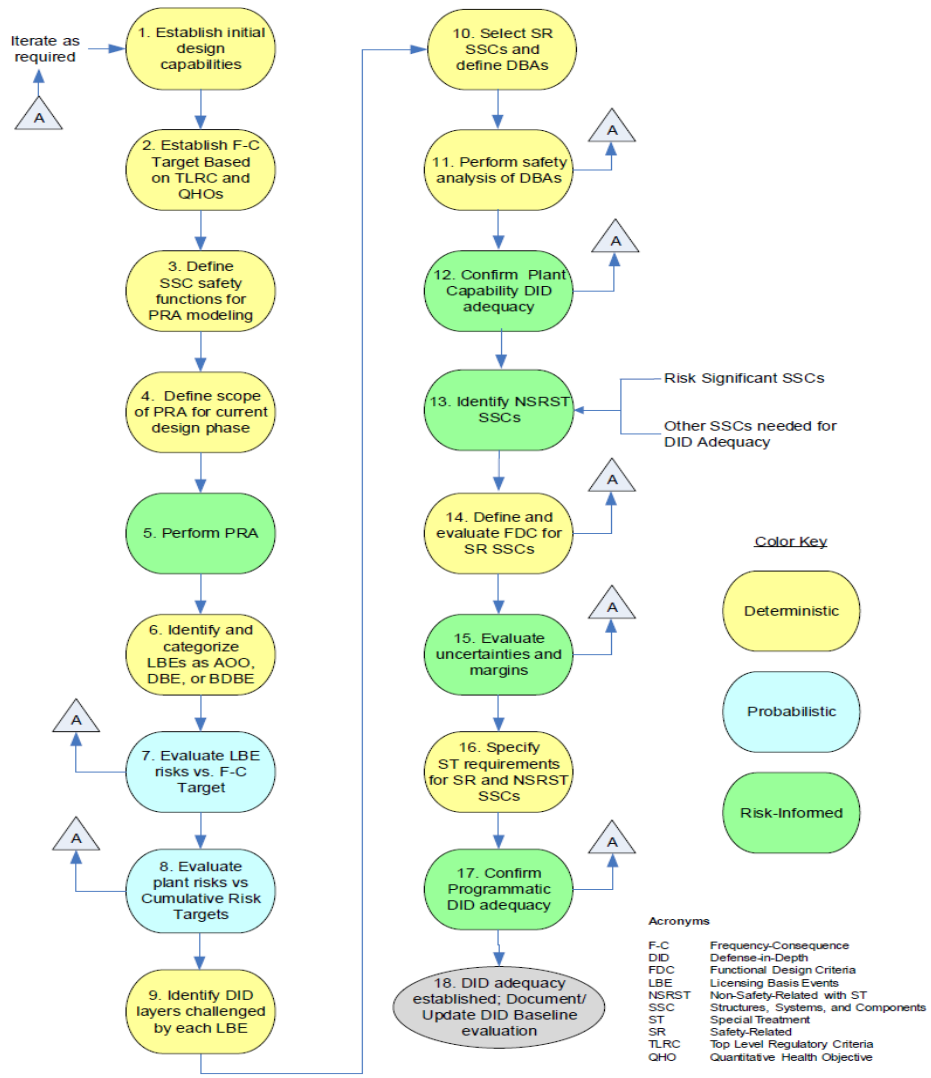

Figure 7-1. LMP Integrated Process for Incorporation and Evaluation of Defense-in-Depth[1]

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8 CONCLUSIONS

8.1 Overall Conclusions

The Demonstration Project met the objectives and deliverables as discussed in pre-demonstration meetings and summarized in this report. Specifically, the following objectives have been met:

1. Introduce the LMP process to Westinghouse—Through the course of several interactions, the LMP team was able to explain the LMP process to the Westinghouse eVinci Micro-Reactor Demonstration participants in a manner that was clearly understood.
2. Demonstrate key processes of the LMP Guidance Document as applied to the eVinci Micro-Reactor design—Significant progress was made to demonstrate the selection and evaluation of LBEs based on information obtained from the eVinci Micro-Reactor PRA for event sequences, combined with performance-based targets for frequency and radiological dose, reflecting the conceptual design and additional efforts to estimate offsite radiological doses for each LBE. The process of defining the required safety functions using design-specific examples was also demonstrated. Options for selecting SR SSCs for each required safety function were identified in these examples. However, it should be noted that classification of SSCs as safety-related will ultimately be made by the designer. An introduction to the process of evaluating plant capability DID adequacy was also reviewed. Other elements of the DID adequacy determination were not exercised within the bounds of the Demonstration Project.
3. Provide an opportunity for Westinghouse to develop a licensing strategy that leverages the LMP process to improve the regulatory certainty of eVinci Micro-Reactor design and safety case The Demonstration provided insights into how the technology-inclusive, RIPB LMP process can be leveraged in the licensing process to provide:
    a. A flexible, systematic, and technically defendable process for developing the foundation for the eVinci Micro-Reactor safety case
    b. Minimization of unnecessary regulatory burden
    c. Potential to increase regulatory certainty, specifically in the areas of LBE selection and evaluation, identification of required safety functions, SSC classification, and DID adequacy once the LMP proposed methodology is endorsed

Although, the design of the eVinci Micro-Reactor has evolved since the majority of the input supporting the demonstration was developed, the process utilized in this demonstration can now be leveraged to support preparations for regulatory interaction as the design of the eVinci Micro-Reactor continues to develop. The breadth and depth of the FMEA, PRA and radiological consequence analysis that provide the foundation of the demonstration will continue to progress in step with the development of design detail. The conservative assumptions and uncertainties

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reflected currently, for example, will likely be reduced. The result will be a robust safety case that uses the NEI 18-04 guidance as an integral part of a fully developed identification of required safety functions, safety classification of SSCs, identification of LBEs / DBAs, and evaluation of the DID adequacy.

8.2 Possible LMP Process Optimizations

The documentation associated with the LMP process was found to be relatively clear, and implementation was fairly straightforward. However, a few points of added clarification, which are described below, could be made relative to the guidance associated with Figure 1-1 to further improve the overall process.

Task 1

Task 1 is fairly straightforward and does not require particular explanation. It is understood that this task is to some extent in parallel with the development of the PRA (i.e., Task 3), and some wording may be added in the description of the task and in the flowchart to suggest the concept that Tasks 1 and 3 are mostly likely performed in parallel. Some clarification may be warranted in the differentiation between the concept of Initiating Events and the concept of LBEs. In a performance-based approach, the LBEs include the potential failure of components and/or specific functions, so the list of LBEs is indeed a result of the PRA rather than an input of the PRA. In this perspective, this task is more an initial identification of Licensing Basis Initiators rather than Events. Therefore, it could be envisioned that Task 1 should come after the current Task 2 and be labeled as "Initial/Updated List of LBEs," as shown in Figure 8-1, as it seems counterintuitive to start defining the list of LBEs even before having at least a pre-conceptual design.

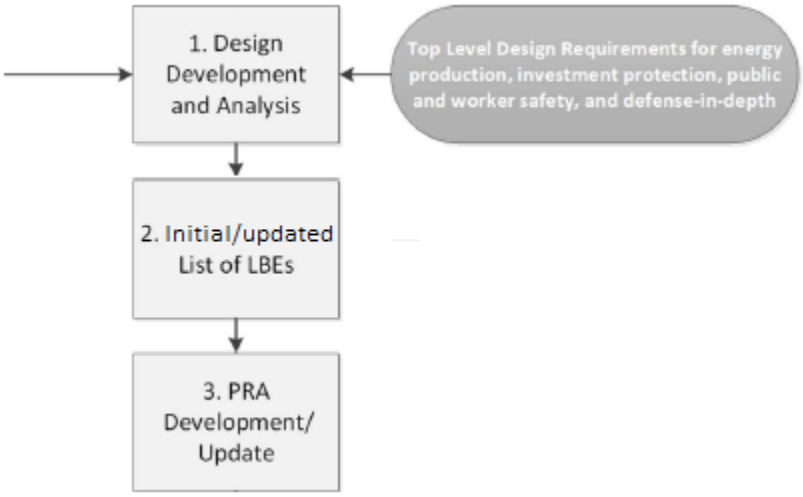

Figure 8-1. Alternative Task 1, 2, and 3 Order

Task 2

A discussion on the role and especially management of assumptions in this task may be appropriate.

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Task 3

It may be worth noting in the LMP documentation that, while the primary focus of any new design pays particular attention to the reactor system, this system normally plays a very small part in the PRA. The development of a working PRA, albeit possibly heavily relying on assumptions, requires a minimal amount of information on the power conversion system to be available. For the eVinci Micro-Reactor, the reactor system is so simplified that the early design dedicated enough attention to the power conversion system. This allowed enough basic information to be generated so that a PRA could be developed.

Task 4

It would be appropriate to expand the discussion on the uncertainties to capture the concept of modelling of epistemic uncertainties, which are much more relevant than aleatory uncertainties, especially in early phases of the design with a lot of design alternatives still being considered.

Task 7

Modelling uncertainties are significantly more important at the early stage of the design, and additional discussion should be provided on that aspect. The PRA standard relies on sensitivities to address model uncertainties, which is what the eVinci Micro-Reactor PRA uses.

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From:	Afzali, Amir
To:	Reckley, William; Cubbage, Amy; Segala, John
Cc:	Jim C. Kinsey Jr; Wayne Moe (wayne.moe@inl.gov); Jill Monahan (monohajs@westinghouse.com); Smith, Matthew C; Shirley, Barry J.
Subject:	[External_Sender] WEC eVinci Report
Date:	Monday, August 12, 2019 8:48:37 PM
Attachments:	EMR_LTR_190010 Westinghouse Approval of eVinci Micro Reactor Licensing Modernization Project Demonstration.pdf

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Dear Mr. Reckley, Ms. Cubbage, and Mr. Segala,

It is my pleasure to submit for your information the Westinghouse Electric Company (WEC) report on eVinci Micro Reactor Licensing Modernization Project (LMP) Demonstration. This collaborative work between Westinghouse and Southern Company LMP team demonstrated the serviceability/adoptability of the LMP proposals, as documented in NEI 18-04, to this micro reactor technology. We are grateful to WEC for investing considerable level of effort in producing such a high quality report in support of LMP which is an important crosscutting industry effort.

Best regards,

Amir

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