INL/RPT-22-65633-Revision-0

# Utilization of the LMP Methodology in Support of the VTR Conceptual Safety Design Report

January 2022

Changing the World's Energy Future

David Grabaskas, Ben Chen, Matthew Bucknor, Jonathon Li, Dennis Henneke, Matthew Warner, Glen Seeman, Jason P Andrus, Alison Wells, Troy P Reiss, Doug Gerstner

Idaho National Laboratory

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**Idaho National Laboratory**
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**http://www.inl.gov**

**Prepared for the**
**U.S. Department of Energy**
**Under DOE Idaho Operations Office**
**Contract DE-AC07-05ID14517**

VTR
VERSATILE
TEST REACTOR

INL/RPT-22-65633

MPL: A17

# VERSATILE TEST REACTOR

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1/17/2022

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REVISION 0

# UTILIZATION OF THE LMP METHODOLOGY IN SUPPORT OF THE VTR CONCEPTUAL SAFETY DESIGN REPORT

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# SYSTEM ABSTRACT

The Versatile Test Reactor (VTR) is a fast spectrum test reactor currently being developed in the United States under the direction of the US Department of Energy (DOE), Office of Nuclear Energy. The VTR is utilizing a risk-informed performance-based (RIPB) approach for design support and authorization by the DOE, derived from recent efforts by the US industry led Licensing Modernization Project (LMP). This document contains an overview of the implementation of the LMP approach in support of the VTR Conceptual Safety Design Report (CSDR). The work reported here is the result of studies supporting a VTR conceptual design, cost, and schedule estimate for DOE-NE to make a decision on procurement. As such, it is preliminary.

The VTR RIPB authorization approach utilizes information from the probabilistic risk assessment (PRA), coupled with deterministic analyses, to aid in decision-making regarding the identification and categorization of safety basis events (SBEs), the classification of structures, systems, and components (SSCs), and the evaluation of defense-in-depth (DID) adequacy. As part of initial reactor design efforts, a VTR conceptual design PRA was developed to support the RIPB process, which focused on at-power internal events, with scoping analyses for seismic and sodium fire hazards.

In addition to supporting numerous design studies, preliminary results from the RIPB approach and the VTR conceptual design PRA were utilized as the basis of the VTR CSDR. The initial identification and categorization of SBEs, SSC classification, and DID evaluation were contained within the CSDR, which was submitted to DOE in 2019 as part of the CD-1 submittal package. Following review, DOE approved the CSDR in April 2020 and the CD-1 package in late 2020.

Valuable experience was gained through the implementation of the RIPB approach for design and authorization during the VTR conceptual design phase, which is summarized in this document. To the extent possible, this experience has been shared with the advanced reactor industry, through publications and participation in licensing tabletops, in addition to informing DOE:NE advanced reactor regulatory development efforts. Furthermore, the approval of the CSDR by the DOE as part of CD-1 represents a significant milestone in the use of RIPB approaches for advanced reactor licensing.

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# REVISION SUMMARY

| Rev # | Section Modified | Revision Summary |
|---|---|---|
| 0 | All | Initial Release |
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# The Utilization of the LMP Methodology in Support of the VTR Conceptual Safety Design Report

## 1.0 INTRODUCTION

The Versatile Test Reactor (VTR) is a fast spectrum test reactor currently being developed in the United States under the direction of the US Department of Energy (DOE), Office of Nuclear Energy. The mission of the VTR is to enable accelerated testing of advanced reactor fuels and materials required for advanced reactor technologies. The conceptual design of the 300 MWth sodium-cooled metallic-fueled pool-type fast reactor has been led by US National Laboratories in collaboration with General Electric-Hitachi and Bechtel National Inc. The VTR is utilizing a risk-informed performance-based (RIPB) approach for design support and authorization by the DOE, derived from recent efforts by the US industry led Licensing Modernization Project (LMP) [1]. This document contains an overview of the implementation of the LMP approach in support of the VTR Conceptual Safety Design Report (CSDR). The work reported here is the result of studies supporting a VTR conceptual design, cost, and schedule estimate for DOE-NE to make a decision on procurement. As such, it is preliminary.

### 1.1 Purpose

The conceptual design phase of the VTR project represents one of the first applications of the LMP methodology to an advanced reactor undergoing design and authorization. The use of RIPB approach represents a fundamental shift in reactor authorization and presents new opportunities to realistically account for plant risk as part of key design and authorization activities. The RIPB insights provide a pathway to justify the authorization treatment of event sequences, the classification of structures, systems, and components (SSCs), the evaluation of defense-in-depth (DID), and the assessment of future plant operations and modifications. Given the importance of the LMP methodology to the advanced reactor industry, the current document was prepared to share insights into its implementation in support of the VTR CSDR and key lessons learned during the process.

### 1.2 Structure

The report has been structured in a fashion to provide information on several key areas of the LMP process implementation for VTR. Section 2.0. provides background information regarding the LMP methodology and modifications that were made to the process to align with applicable DOE regulations and guidance. Section 3.0 describes the development of the VTR conceptual design PRA, which forms the basis of the risk-informed analyses conducted as part of the LMP approach. In Section 4.0, a review of the implementation of the LMP analyses is provided. Key lessons learned and other insights are reviewed in Section 5.0, with overall conclusions in Section 6.0.

## 2.0 ADAPTED LMP APPROACH

The LMP process represents a RIPB approach for nonlight water reactor (non-LWR) licensing under the U.S. Nuclear Regulatory Commission (NRC). The LMP approach has been endorsed by the NRC in RG 1.233 [2], and further efforts to outline licensing documentation based on the methodology are being conducted by the Technology Inclusive Content of Application Project (TICAP) [3] and Advanced Reactor Content of Application Project (ARCAP). Given its recent development and endorsement, the decision was made by the VTR project to adopt the LMP approach to support reactor authorization by the DOE. The following section describes the LMP process and how it was adapted to the DOE authorization structure.

### 2.1 Background

The engineering design and potential construction VTR are to the requirements of DOE O 413.3B, “Program and Project Management for the Acquisition of Capital Assets” [4]. As shown in Figure 2-1, authorization of facilities under DOE O 413.3B is a phased process based on critical decision (CD) levels. In accordance with DOE O 413.3B, safety must be integrated into the design process for new or major modifications to DOE hazard category 1 nuclear facilities, such as VTR. Requirements provided in DOE O 413.3B and DOE O 420.1C Change 2, “Facility Safety,” [5] and the expectations of DOE-STD-1189-2016, “Integration of Safety into the Design Process,” [6] provide for identification of hazards early in the project and use of an integrated team approach to design safety into the facility.

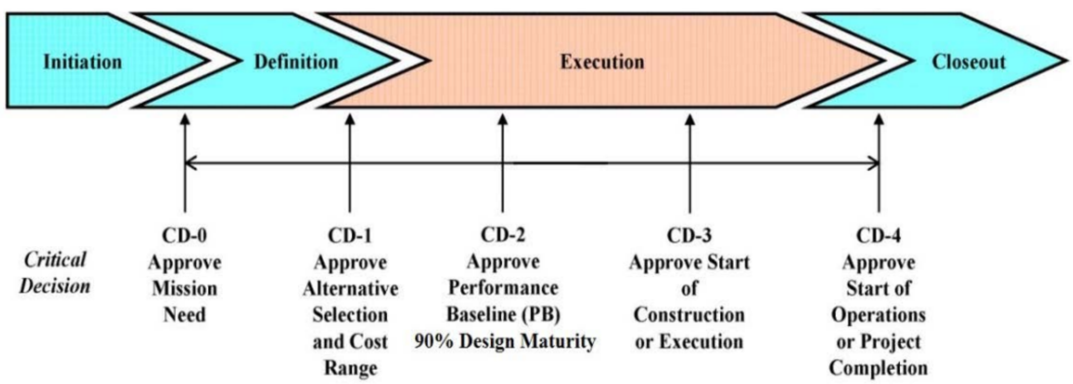


**Figure 2-1: DOE Critical Decision Levels [4]**

Major safety basis document submittals are expected at certain CD levels, detailed in Table 2-1. Note that VTR currently plans to couple the CD-2 and CD-3 levels. The output of the conceptual design phase of reactor development is the CSDR, the purpose of which is to summarize the hazards analysis efforts and safety-in-design decisions incorporated into the VTR conceptual design along with any identified project risks associated with the selected strategies. This CSDR is prepared in accordance with the suggested format and content provided in DOE-STD-1189-2016.

**Table 2-1: Authorization Safety Documents at CD Levels**

| Critical Decision Level | Authorization Safety Document |
|---|---|
| CD-1 | Conceptual Safety Design Report (CSDR) |
| CD-2/3 | Preliminary Safety Analysis Report (PSAR) |
| CD-4 | Final Safety Analysis Report (FSAR) |

In terms of project structure, an integrated project team (IPT) leads reactor development and consists of a cross-functional group of individuals organized for the specific purpose of delivering a project where the technical, management, budgetary, safety, and security interests are met. The IPTs are the primary tool for breaking down the walls that can exist between different organizations, different professions, and different levels within the command structure. A successful IPT brings the diverse elements together to form a unit that is willing to share information and balance priorities and ideologies in efforts to successfully execute the project mission while achieving the overall safety strategy.

The safety design integration team[1] (SDIT) supports the IPT to ensure the integration of safety into the design process. The composition of this team is adjusted as necessary to ensure the proper technical representation, including traditional worker safety disciplines, emergency management, and safeguards and security commensurate with the analyzed hazards and the specific project phase. The SDIT ultimately supports decisions to be made by the Federal Project Director. Core members of the SDIT, responsible for the implementation of safety-in-design for the project, and their corresponding responsibilities are identified in the SDIT charter.

## 2.2 Process

To fulfill the requirements associated with reactor authorization under the DOE, it was necessary to select a systematic and reproducible framework for the analysis of reactor safety and risk. This includes the incorporation of related information into the design and authorization decisions, such as the selection of safety basis events (SBEs), the classification of SSCs, and the determination of the adequacy of the VTR DID strategy. The LMP approach represents a RIPB method to satisfy these goals. The following information regarding the approach was documented in the VTR Safety Design Strategy (SDS), which is a formal submittal to the DOE as part of the authorization process.

The overall LMP process, depicted in Figure 2-2, was adopted for VTR and is described below. Note that individual steps may be iterative based upon information gained at any step impacting previous steps.

1. This task defines an initial list of SBEs based on past experience and preliminary analyses.
2. This task defines key elements of the safety design approach, the design approach to meet the top-level design requirements for energy production and investment protection, and analyses to develop sufficient understanding to perform a probabilistic risk assessment (PRA) and the deterministic safety analyses. This task is performed in phases (conceptual, preliminary, and final design) and may include iterations within phases.
3. A PRA model is developed and then updated as appropriate for each phase of the design. Prior to the first introduction of the PRA, it is necessary to develop a technically sound understanding of the potential failure modes of the reactor concept, how the reactor plant

[1] More information on the role and composition of the SDIT can be found in DOE-STD-1189-2016 [6].

would respond to such failure modes, and how protective strategies will be incorporated into formulating the safety design approach.

4. The event sequences modeled and evaluated in the PRA are grouped into accident families, each having a similar initiating event, challenge to the plant safety functions, plant response, end state, and mechanistic source term if there is a radiological release. Each of these families will be assigned to an SBE category based on event sequence frequency of occurrence.
5. The full set of SBEs will be examined to identify that the safety functions necessary and sufficient to ensure that the consequence criteria are met. For each of these required safety functions, a decision will be made on which SSCs: 1) perform the required safety functions, 2) are available on all the SBEs, and 3) should be classified as safety class.
6. For each SBE identified in Task 5, a deterministic DBA will be defined that includes the required safety function challenges represented in the SBE, but assumes that the required safety functions are performed exclusively by safety-related SSCs, and all non-safety SSCs that perform these same functions are assumed to be unavailable. These DBAs will then be used for supporting the conservative deterministic safety analysis of the SAR.
7. This task will involve the deterministic and probabilistic safety evaluations that are performed for the full set of SBEs.
8. The purpose of this task will be to decide if additional design development is needed, either to proceed to the next logical stage of design, or to incorporate feedback from the SBE evaluation that design, operational, or programmatic improvements should be considered.

The remaining tasks in the figure involve the completion of the SSC safety classification process and the formulation of performance requirements for SSCs. In this approach, risk information from the PRA is utilized to inform a number of activities:

- Supporting and evaluating the development of the VTR design.
- Identifying the spectrum of SBEs to be considered.
- Evaluating the risk significance of SBEs against frequency-consequence evaluation criteria.
- Safety classification of SSCs.
- Development of performance criteria for the reliability and capability of SSCs in the prevention and mitigation of accidents.
- Determining integrated plant performance margins compared to performance-based objectives.
- Exposing and evaluating sources of uncertainty in the identification of SBEs and in the estimation of their frequencies and consequences, and providing key input to the evaluation of the adequacy of DID.
- Providing risk and performance-based insights into the evaluation of the design DID adequacy.
- Supporting other VTR RIPB decisions.

Although these activities are risk-informed, they are not risk-based, as complementary deterministic analyses and expert reviews are conducted throughout the process. The following subsections detail each step in the process and the analyses performed.

When comparing the LMP approach described in NEI 18-04 [1] to the VTR process discussed here, it is important to note that differences in nomenclature due to the alignment with DOE terminology. A comparison between the VTR approach terminology and comparable terms from the LMP process is provided in Table 2-2. Although many of the terms are similar, the definitions may not necessarily be identical. For example, safety significant (SS) SSCs under the DOE may not have the same requirements as non-safety related with special treatment (NSRST) under the NRC.

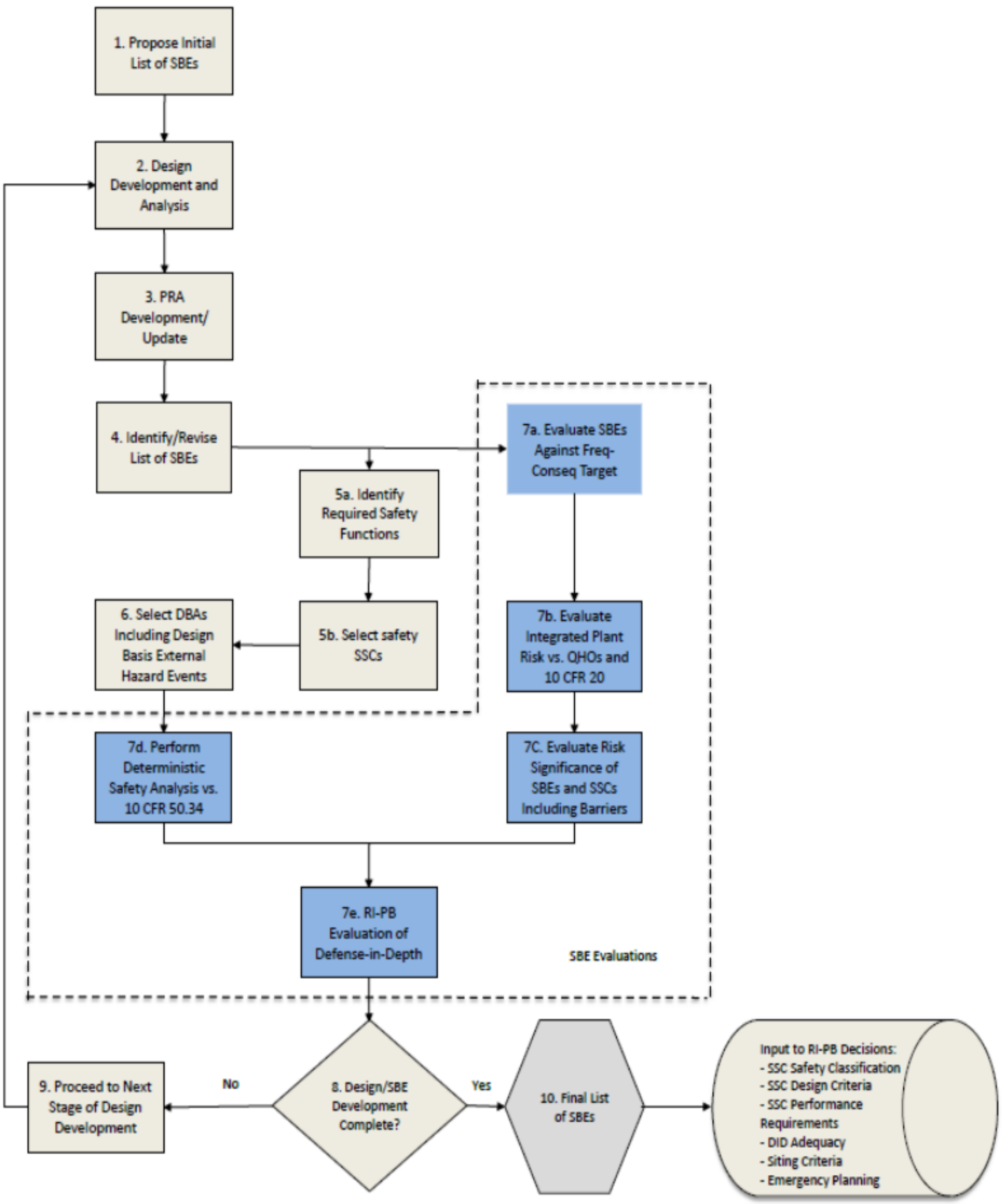


**Figure 2-2: VTR Adapted LMP Analysis Methodology**

**Table 2-2: Comparison of LMP and VTR Terminology**

| VTR Term | Comparable LMP Term |
|---|---|
| Safety Basis Event (SBE) Categories:<br>• Anticipated<br>• Unlikely<br>• Extremely Unlikely | Licensing Basis Event (LBE) Categories:<br>• Anticipated Operational Occurrence (AOO)<br>• Design Basis Event (DBE)<br>• Beyond Design Basis Event (BDBE) |
| SSC Classes:<br>• Safety Class (SC)<br>• Safety Significant (SS)<br>• Non-Safety (NS) | SSC Classes:<br>• Safety Related (SR)<br>• Non-Safety Related with Special Treatment (NSRST)<br>• Non-Safety Related with No Special Treatment (NST) |

## 2.3 SBE Categorization and Evaluation

Fundamental to the VTR authorization process is the identification of SBEs, which are defined as the entire collection of event sequences considered in the design and safety basis of the plant. The PRA informs a set of bounding, unique, and representative SBEs that will undergo deterministic consequence analyses to demonstrate compliance with the identified frequency consequence guidelines. This event selection process is common practice in development of DOE facilities and is consistent with DOE-STD-3009-2014, “Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,” [7].

In the SBE categorization process, event sequences (or event sequence families, as will be discussed in section 4.1) from the PRA are placed into categories according to their estimated frequency of occurrence and the guidelines in Table 2-3. The placement of SBEs into the designated categories has an impact on the type of analyses performed and their role in certain safety basis decisions, like SSC classification.

**Table 2-3: VTR Radiological Consequence Guidelines for SBE Categories**

| SBE Category | Frequency Range (/yr) | Radiological Consequence Guideline (TED - rem) | | |
|---|---|---|---|---|
| | | Offsite | Onsite | Worker |
| Anticipated | $F \geq 10^{-2}$ | <5 | <5 | N/A |
| Unlikely | $10^{-2} > F \geq 10^{-4}$ | <5 | <25 | <25 |
| Extremely Unlikely | $10^{-4} > F \geq 10^{-6}$ | <25 | <100 | <100 |
| Beyond Extremely Unlikely | $F < 10^{-6}$ | No Criteria | No Criteria | No Criteria |

Both the LMP process and the VTR approach utilize a frequency-consequence (F-C) curve, which links event sequence frequency with potential consequence, to aid in decision-making. Satisfying the F-C curve does not necessarily imply satisfaction of regulatory criteria for reactor authorization or licensing, but the curve provides guidance for the determination of event sequence categorization, SSC classification, and DID evaluation. The LMP F-C curve, shown in Figure 2-3, couples event sequence frequency and offsite consequence (30-day total effective dose equivalent - TEDE - at the exclusion area boundary). The limits of the LMP F-C curve are based on CFR, EPA, and NRC regulatory guidelines. For VTR, the DOE has a different set of regulatory guidelines, with separate consequence limits for offsite individuals and collocated workers, as shown in Figure 2-4 and Figure 2-5. DOE F-C curve limits are not linear in log-space, as in LMP, but are constant for each category of SBEs. In general, this results in more conservative offsite consequence limits at low frequencies of the SBE categories for the VTR approach.

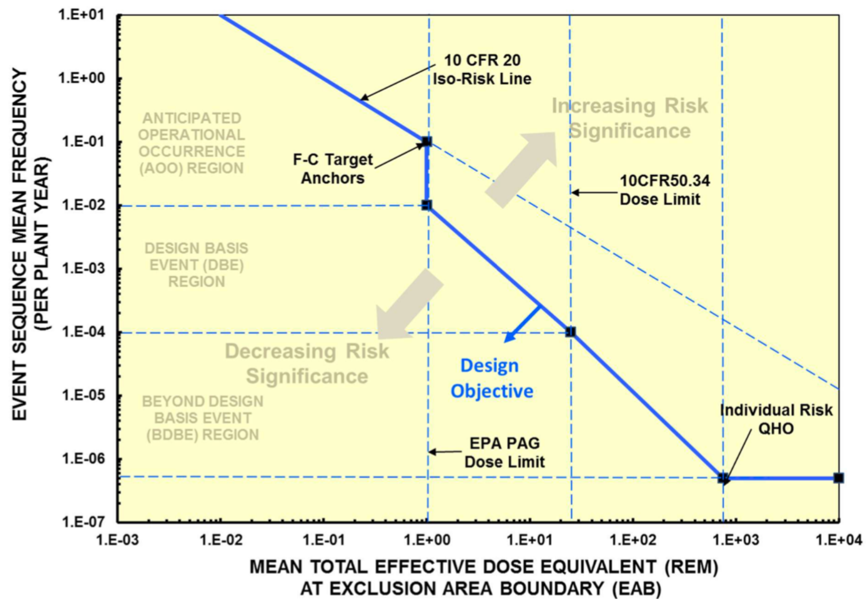


**Figure 2-3: LMP Frequency versus Consequence Curve[2] [1]**

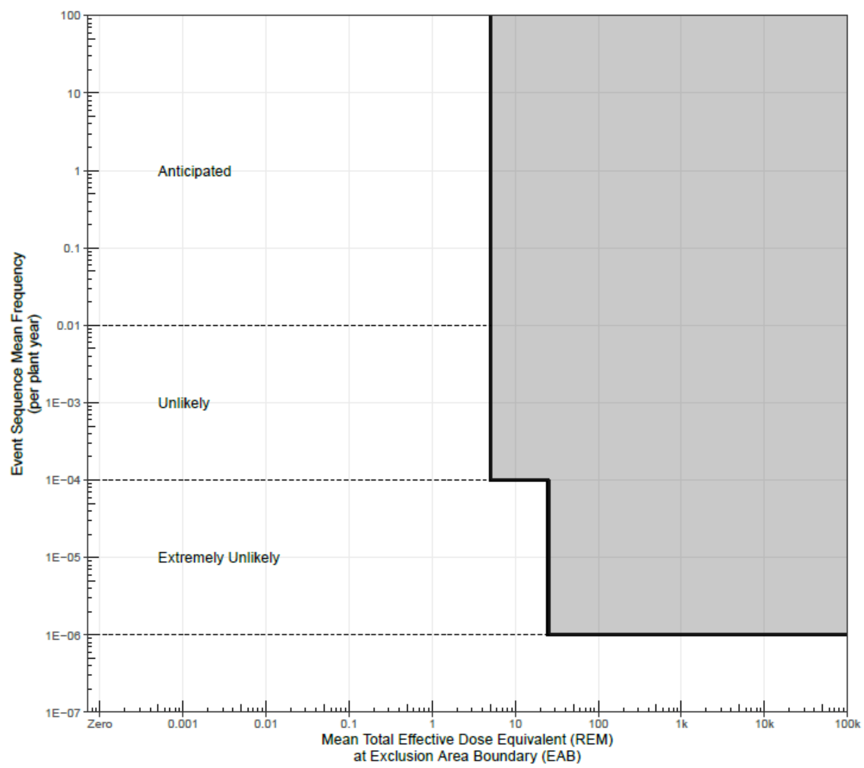


**Figure 2-4: VTR Public Frequency versus Consequence Curve**

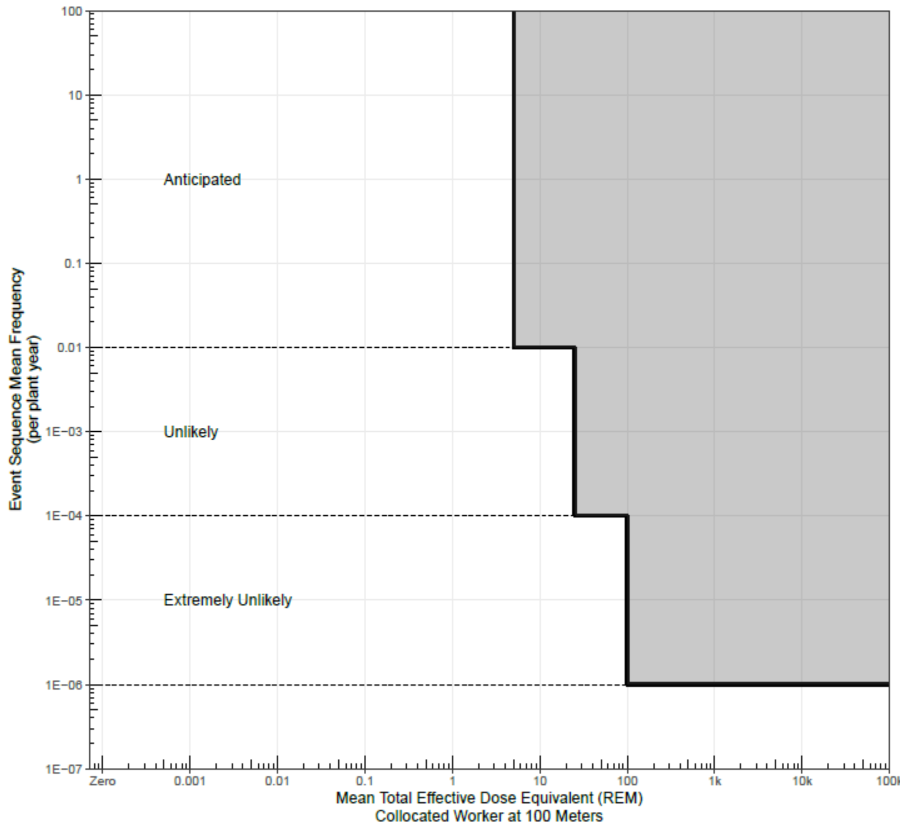


**Figure 2-5: VTR Collocated Worker Frequency versus Consequence Curve**

## 2.4 SSC Classification

VTR SSC classification is performed based upon the relative importance of the SSC to plant safety. Ultimately, the purpose of the designation of these systems is to assure their functionality to prevent or mitigate the release of radioactive materials to the public, workers, or environment. Three classifications of equipment are utilized for VTR SSCs based upon their importance in preventing or mitigating events that could lead to release of uncontrolled radioactive material. These classifications are consistent with current DOE SSC classification schemes in DOE O 420.1C, which allows for appropriate integration into other DOE requirements for system engineering, maintenance, operations, and incident reporting. The classifications and a description of criteria resulting in their classification are outlined in Table 2-4, which contains both risk-informed and deterministic criteria.

**Table 2-4: VTR SSC Classification Criteria**

| SSC Classification | Criteria Type | Description |
|---|---|---|
| Safety Class (SC) | Risk-Informed | • Offsite F-C Curve[1]:<br>○ For SBEs greater in frequency than $10^{-6}$/yr, an SSC is SC if its removal causes the SBE to violate the F-C curve, when considering a one-by-one removal of SSCs, crediting all remaining SSCs, regardless of safety classification, at appropriate reliability levels (and with appropriate accounting of common cause failure).<br>○ For DBAs derived from SBEs in the "Unlikely" category (<$10^{-2}$ to >$10^{-4}$/yr), an SSC is SC if it is necessary for the DBA to satisfy the 25 rem consequence limit when utilizing deterministic, prescriptive analysis of the event sequence and crediting *only* SC SSCs. |
| | Deterministic or Defense-in-Depth | • If the SSC is required to ensure integrity of the primary coolant boundary.<br>• If the SSC is required to ensure reactor shutdown.<br>• If SC classification is determined necessary for the SSC based on IDP and SDIT review to address uncertainties or assumptions within the PRA analysis or specific, high-consequence DID adequacy. |
| Safety Significant (SS) | Risk-Informed | • Offsite F-C Curve[1]:<br>○ For SBEs in the "Extremely Unlikely" region (<$10^{-4}$ to >$10^{-6}$/yr), an SSC is SS if its removal causes the SBE to violate the F-C curve when considering *only* SC and SS SSCs appropriate for the SBE.<br>• Collocated Worker F-C Curve[1]:<br>○ For SBEs greater in frequency than $10^{-6}$/yr, an SSC is SS if its removal causes the SBE to violate the F-C curve when considering *only* SC and SS SSCs appropriate for the event sequence.<br>• SSC performs risk significant function, where risk significant is defined as:<br>○ If the SSC makes a significant contribution (>1% of the limit value) to the cumulative risk metrics. |
| | Deterministic or Defense-in-Depth | • If SS classification is determined necessary for the SSC based on IDP and SDIT review to address uncertainties or assumptions within the PRA analysis or DID adequacy.<br>• SSC is necessary to protect public or workers from a chemical hazard above DOE limits. |
| Non-Safety (NS) | N/A | All other facility systems not classified as SC or SS are *de facto* classified as non-safety. |

[1] For the treatment of uncertainties, all comparisons to frequency and consequence limits of the F-C curve are performed utilizing the 95th-percentile of the uncertainty distributions associated with the SBE frequency and consequence. For preliminary analyses, additional margin to the limits may be applied in substitute for detailed uncertainty analyses.

SC classification of SSCs (SC SSC) is determined based upon their necessity to perform a RSF or limit the public risk from an identified SBE to within the approved SBE evaluation guidelines in Table 2-3. The risk-informed criteria gauge the importance of the SSC in limiting radionuclide releases to the public, utilizing the offsite dose F-C guidelines. First, an SSC is considered SC if its removal results in the consequence of any SBE of frequency greater than 1E-6 per year violating the consequence limits specified in Table 2-3. The second risk-informed SC criterion examines DBAs, which are event sequences derived from those SBEs in the *unlikely* region. These SBEs are re-evaluated utilizing deterministic, prescriptive analyses and only crediting SC SSCs. Therefore, all SSCs required to assure that the consequences of DBAs remain below 25 rem offsite are considered SC.

There are two deterministic SC SSC criteria. First, all SSCs that compose the primary coolant boundary are considered SC. Second, SSCs required to ensure reactor shutdown are considered SC. Note that the second requirement does not apply to all reactivity control systems if redundancy is available. The last SC criterion allows the inclusion of additional SSCs as SC if determined necessary during the review by the integrated decision panel (IDP) and/or the SDIT, discussed further in Section 2.8. The IDP or SDIT may determine it necessary to include additional SC SSCs to address uncertainties within the PRA or high-consequence DID adequacy. It is important to note that while the IDP and SDIT may elevate the classification of an SSC, *they may not reduce the classification of an SSC if it meets one of the designated SSC classification criteria*. The IDP and SDIT may also consider factors such as risk, cost, and compliance for determining SC SSCs in instances where redundant or diverse SSCs are available to perform a similar safety function.

SS classification is determined either by SSC performance as an important DID function to SC SSCs or their necessity to limit collocated worker risks from SBEs to within the appropriate collocated worker guidelines specified in Table 2-3. First, an SSC is considered SS if its removal results in the consequence of any SBE in the extremely unlikely category violating the offsite dose limits, when crediting only SS and SC SSCs. The second SS criterion is similar to the first SC criterion, but utilizes the collocated worker F-C guidelines rather than the offsite limits. For all SBEs greater than 1E-6 per year in frequency, an SSC is considered SS if its removal results in the SBE violating the collocated worker guidelines, when crediting other SS and SC SSCs.

In addition to the two risk-informed criteria above, an SSC may also be designated SS if it is considered “risk significant,” which is discussed further in Section 2.5. An SSC may also be designated as SS if determined necessary by the IDP and/or SDIT to address uncertainties within the PRA or DID adequacy. Lastly, an SSC may be designated as SS if it is required to protect the public or workers from chemical hazards above acceptable DOE limits.

All other SSCs not classified as SC or SS are de facto classified as Non-Safety (NS).

## 2.5 Risk Significance

The determination of risk significance is important for several elements of VTR authorization. First, the identification of risk significant SSCs is necessary for their proper classification. Risk significant SSCs are identified utilizing two approaches. An SSC is risk significant if its failure would result in an SBE lying at a value above the acceptable limits of the F-C curves (similar to the criteria in Table 2-4) or if it makes a significant contribution to the satisfaction of the cumulative risk metrics. A significant contribution to each cumulative risk metric limit is satisfied when the total frequency of all SBEs with failure of the SSC exceeds 1% of the cumulative risk metric limits, discussed in Section 2.6.

The second use of risk significance is the identification of risk significant event sequences. This is a necessary step for the satisfaction of requirements within the ASME/ANS Non-LWR PRA standard, discussed in Section 3.1.2, as the standard mandates different analysis methodologies for risk significant event sequences. For example, state-of-knowledge correlation within modeling uncertainty must be treated at a higher level of detail for risk significant event sequences.

The VTR PRA utilizes absolute risk measures, rather than relative metrics, for the determination of risk significance SBEs. This decision is based on lessons learned from the PRISM PRA update [8], where several problems were encountered with relative risk measures. First, unlike LWR PRAs, which typically utilize a single measure of risk (cumulative core damage frequency), advanced reactor PRAs utilize the offsite consequences from many release categories. This makes the comparison of relative risk significance difficult, as the offsite consequences are not contributors to a single encompassing risk measure. Second, advanced reactors are purposefully designed to reduce overall system risk, which can render relative measures of risk significance impractical for the determination of important event sequences and SSCs.

The VTR PRA risk significance criteria for SBEs are similar to those developed by the LMP but are modified to satisfy DOE. Risk significant SBEs are determined by their relationship to the frequency versus consequence limit curve. SBEs that lie within an area close to the limit curve utilizing 95th percentile values for the frequency and consequence of the SBE are considered risk significant. The risk significance area was determined based on several factors, as described below:

- ***Anticipated* Region**: Risk significance region is determined from the limit curve line of 5 rem to a lower limit of 0.0025 rem, which is 10% of the normal 30-day dose from background sources. The 0.0025 rem limit is considered a de minimis consequence criteria for the consideration of risk significant SBEs, as chosen by the LMP [1].
- ***Unlikely* through the *Extremely Unlikely* Region**: Risk significance region is determined based on an area created between the limit curve (5 rem for the Anticipated and Unlikely regions and 25 rem for the Extremely Unlikely region) and a log-line with points at 1E-2 per year frequency with 0.0025 rem consequence and 1E-6 per year frequency with 5 rem consequence. The use of a sloped line, rather than a stairstep curve, for the lower limit is to account for the reduction in risk as frequency is reduced (as risk is frequency multiplied by consequence). The first point on the line delineating the lower bound of the risk significance region was determined based on the risk significance area of the *anticipated* region. The second point at 5 rem and 1E-6 per year was chosen since SBEs below this consequence level could increase by two orders of magnitude in frequency and still meet the limit curve.

In addition to the risk significance region, there is an additional “cliff-edge” effect examination area at very low frequencies (1E-6 to 1E-8 per year). The purpose of this region is to ensure that the event sequences that have potentially large offsite consequences are sufficiently understood and properly categorized. This includes a bounding estimate of uncertainties to confirm that the event sequences could not potentially cross into the Extremely Unlikely region. The reasoning for the shape of this area is as follows:

- ***Beyond Extremely Unlikely* Region**: Cliff-edge examination region is bound between frequencies of 1E-6 and 1E-8 per year, with consequence greater than a log-line with points at 5 rem at 1E-6 per year and 25 rem at 1E-8 per year. SBEs with an offsite dose lower than this line could incur a frequency increase of greater than two orders of magnitude and still satisfy the limit curve.

### 2.6 Integrated Risk

Integrated or cumulative risk metrics are utilized for additional risk insights and to assist decision-making as part of the VTR process, but they are not explicit authorization criteria. The first two cumulative risk metrics are provided in DOE Policy 420.1 [9], which contains two quantitative safety objectives for nuclear safety:

> *Having as its Safety Goal to conduct its operations such that (a) Individual members of the public be provided a level of protection from the consequences of DOE operations such that individuals bear no significant additional risk to life and health to which members of the general population are normally exposed, and (b) DOE*

*workers' health and safety are protected to levels consistence with or better than that achieved for workers in similar industries.*

*The following two quantitative safety objectives for public protection are established as "aiming points" (not requirements) in support of the Safety Goal that guides the development of DOE's nuclear safety requirements and standards:*

- *The risk to an average individual in the vicinity of a DOE nuclear facility for prompt fatalities that might result from accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the population are generally exposed. For evaluation purposes, individuals are assumed to be located within one mile of the site boundary.*
- *The risk to the population in the area of a DOE nuclear facility for cancer fatalities that might result from operations should not exceed one-tenth of one percent (0.1%) of the sum of all cancer fatality risks resulting from all other causes. For evaluation purposes, individuals are assumed to be located within 10 miles of the site boundary.*

These safety objectives are identical to the quantitative health objectives (QHOs) of the NRC [10], which translate to the following quantitative limits on offsite consequence [11]:

- **Prompt fatality QHO**: $5\text{x}10^{-7}$ per year for an average individual within 1 mile
- **Latent cancer fatality QHO**: $2\text{x}10^{-6}$ per year for an average individual within 10 miles

In addition to these two cumulative risk metrics, a third risk metric is identified in 10 CFR 20 [12], which limits the total effective dose of members of the public to 0.1 rem in a year. This metric is utilized to evaluate the impact of smaller but potentially more frequent releases.

## 2.7 Defense in Depth Adequacy

The VTR safety design approach implements the DID strategy by adopting the traditional five layers of DID, as follows:

Layer 1: Prevention of abnormal operation and failures

Layer 2: Control of abnormal operation and detection of failures

Layer 3: Control of accidents within the design basis

Layer 4: Control of severe facility conditions

Layer 5: Mitigation of radiological consequences.

For the layer 1, accident prevention is the first priority. This level focuses on reliable normal operation and accident prevention through features of the plant design, construction, availability, operation, and maintainability, and includes reliability enhancement through redundancy, QA, testability, inspectability, and simplified fail safe system design. The first layer of defense treats events of the lowest severity. The objective for provisions of Layer 1 is to control small plant disturbances and transients. Success in this objective results in prevention of off-normal operation and anticipated events. Principles of redundancy, diversity, and independence will be observed to ensure that no single failure or removal from service of an active or passive component can result in loss of a safety function. Performance of safety and security functions will not depend on a single element of design, construction, maintenance, or operation. At least two redundant, diverse, and independent means for generating signals to effect reactor shutdown and decay heat removal will be provided, and there will be at least two barriers to fission product release.

The second layer of safety prevents accident propagation, recognizing that accidents may occur despite the care taken in design, construction, and operation associated with Layer 1 of DID. The second layer of defense treats events that fall into the anticipated frequency category. The objective of DID Layer 2 is to detect and control anticipated events, including identification of their cause and taking corrective actions.

The objective of the third layer of defense is to control unlikely and extremely unlikely event plant conditions within the design basis. This is accomplished by engineered safety features that are capable of leading the facility to a safe controlled state. This is achieved in VTR by conservative design and engineered safety systems for reactor shutdown, and decay heat removal. Success in meeting the objectives in this Layer occurs when required safety functions have been performed for the DBAs. A central component of DID is the use of successive, multiple physical barriers for protection against release of radioactivity and hazardous materials. Multiple, diverse, and independent means are provided to accomplish safety functions. This layer's objective is to control severe plant conditions and mitigate beyond extremely unlikely event consequences.

The fourth layer is to control severe plant conditions and mitigate beyond extremely unlikely event consequences. The proposed VTR design will be capable of accommodating various beyond extremely unlikely basis accident initiators without producing conditions that might lead to a severe accident. Successful operation of Layer 4 involves maintaining critical safety functions for the retention of radioactive or hazardous material.

The fifth layer of defense applies to severe accidents where significant releases of radiological or hazardous material occurs. The objective of Layer 5 is to mitigate accident doses to workers and the public by employing anticipatory emergency planning and off-site accident management. It serves to ensure that even in the extremely unlikely event of a severe accident, adverse impacts on health and safety are still avoided. Successful operation of this layer prevents adverse health and safety impacts.

Utilizing the LMP guidance, the adequacy of the five layers of DID from a plant capabilities perspective are evaluated utilizing the qualitative and quantitative criteria outlined in Table 2-5. Programmatic DID considerations, such as ensuring adequate margin and providing adequate assurance, are completed as part of the IDP review process. Section 4.4 will further discuss the implementation of these criteria for the VTR conceptual design phase.

**Table 2-5: LMP Plant Capabilities DID Criteria[3] [1]**

<table>
<tr><th rowspan="2">Layer[a]</th><th colspan="2">Layer Guideline</th><th colspan="2">Overall Guidelines</th></tr>
<tr><th>Quantitative</th><th>Qualitative</th><th>Quantitative</th><th>Qualitative</th></tr>
<tr><td>1) Prevent off-normal operation and AOOs</td><td colspan="2">Maintain frequency of plant transients within designed cycles; meet owner requirements for plant reliability and availability[b]</td><td rowspan="5">Meet F-C Target for all LBEs and cumulative risk metric targets with sufficient[d] margins</td><td rowspan="5">No single design or operational feature,[c] no matter how robust, is exclusively relied upon to satisfy the five layers of defense</td></tr>
<tr><td>2) Control abnormal operation, detect failures, and prevent DBEs</td><td>Maintain frequency of all DBEs < $10^{-2}$/plant-year</td><td>Minimize frequency of challenges to SR SSCs</td></tr>
<tr><td>3) Control DBEs within the analyzed design basis conditions and prevent BDBEs</td><td>Maintain frequency of all BDBEs < $10^{-4}$/plant-year</td><td>No single design or operational feature[c] relied upon to meet quantitative objective for all DBEs</td></tr>
<tr><td>4) Control severe plant conditions and mitigate consequences of BDBEs</td><td rowspan="2">Maintain individual risks from all LBEs < QHOs with sufficient[d] margins</td><td rowspan="2">No single barrier[c] or plant feature relied upon to limit releases in achieving quantitative objectives for all BDBEs</td></tr>
<tr><td>5) Deploy adequate offsite protective actions and prevent adverse impact on public health and safety</td></tr>
<tr><td colspan="5">Notes:<br>[a] The plant design and operational features and protective strategies employed to support each layer should be functionally independent.<br>[b] Non-regulatory owner requirements for plant reliability and availability and design targets for transient cycles should limit the frequency of Initiating Events and transients and thereby contribute to the protective strategies for this layer of DID. Quantitative and qualitative targets for these parameters are design specific.<br>[c] This criterion implies no excessive reliance on programmatic activities or human actions and that at least two independent means are provided to meet this objective.<br>[d] The level of margins between the LBE risks and the QHOs provides objective evidence of the plant capabilities for DID. Sufficiency will be decided via the IDP.</td></tr>
</table>

## 2.8 Integrated Decision Panel (IDP)

A critical aspect of the implemented LMP approach is the review performed by the IDP as part of the decision-making process regarding the reactor design and safety basis. The IDP involves knowledgeable individuals representing engineering, risk and safety analysis, operations, and maintenance perspectives to understand the plant safety case and guide the RIPB decision-making process. The Nuclear Energy Institute (NEI) has developed procedures and guidelines for the makeup and responsibilities of such panels for other risk-informed applications. Specifically, NEI 00-04 [13] provide useful guidance on the composition of a panel and the associated output documentation. In many instances, these individuals are also members of the SDIT.

The IDP has several important roles:

1) Confirming the SBE identification, categorization, and evaluation against the F-C curves.
2) Confirming the SSC classification (or selecting from sets of potential SSC classification).
3) Evaluating DID adequacy.

Each IDP member is responsible for the following:

- Serves as a member of a multi-disciplinary review panel collectively having broad knowledge of plant design, licensing requirements, operating and maintenance practices, risk and experience.
    - Each IDP member is responsible for providing input to the decision-making process from their respective discipline area (e.g., Nuclear Safety, System Engineering, etc.) while recognizing the needs and constraints of other organizations.
- Ensures all attributes of the evaluation presented to them are fully addressed to provide a valid risk-informed conclusion or decision that addresses the regulatory and system requirements as well as maintenance of DID and adequate safety margin.
- Evaluates PRA risk insights, passive risk insights, and qualitative risk insights to reach a consensus-based categorization recommendation for system functions and components.
- Evaluates recommended changes to categorization resulting from changes to the plant, PRA model updates, changes to operational practices, as well as other applicable changes.

The IDP is run as a consensus panel is facilitated by the IDP Chair or a designated facilitator, who acts as the Chair during the IDP. The facilitator coordinates the IDP sessions and ensures the IDP decisions are documented and retained as a quality record. This function is critical to future decision-making regarding plant changes which have the potential to affect DID. The facilitator should attempt to reach a unanimous decision on specific recommendations for the above process steps, but a consensus is needed to finalize the recommendations. Any differing opinions by IDP members should be documented. Further detail regarding the implementation of the IDP is provided in Section 4.5.

# 3.0 VTR PRA DEVELOPMENT

The following section details the development of the VTR conceptual design PRA, which served as the basis for the LMP implementation during the conceptual design phase. An overview of standard guidance, the PRA methods, and scope are provided. The VTR PRA builds on the experience of past SFR PRAs, including the Clinch River Breeder Reactor PRA [14], EBR-II PRA [15], SAFR PRA [16], and two versions of the PRISM PRA [17] [18]. In addition, from 2014-2015 GE-Hitachi and Argonne collaborated on an update/modernization of the PRISM PRA, with the goal to create an advanced reactor PRA that satisfied the requirements of the new ASME/ANS Advanced Non-LWR PRA Standard [19]. The techniques and methodologies developed by that project, along with the lessons learned from the experience, are reflected in the VTR conceptual design PRA.

## 3.1 Standard Guidance

Given the importance of the PRA to the safety basis and authorization of VTR, ensuring adequate confidence in the PRA development process and subsequent analyses was a central focus. Two standards were utilized to guide the development of the VTR PRA: DOE-STD-1628-2103 and the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard for advanced non-LWR power plants.

### 3.1.1 DOE-STD-1628-2013

DOE-STD-1628-2013 (*Development of a Probabilistic Risk Assessment for Nuclear Safety Applications)* [20] outlines the requirements associated with the use of a PRA as part of the authorization of a DOE nuclear facility. The standard focuses on the planned uses of the PRA within the nuclear safety application and the assurance of technical adequacy, rather than dictating technical methods. A central tenet of the standard is the requirement to develop and submit a formal PRA plan to the DOE as part of the authorization process. The PRA plan must outline the following:

- — PRA approach
  - Detailed assumptions
  - Methodology for parameter estimation and analysis
  - Methodology description
  - Schedule and resources
- — Anticipated outcomes and intended use of information
  - Outcomes
  - Interpretation of results
  - Impact on safety basis
- — PRA technical adequacy and peer review approach
  - PRA team and review personnel
  - Completeness and transparency of documentation
  - Procedures
  - Configuration control and performance monitoring
  - Quality assurance requirements
  - Technical and peer reviews

Prior to the development of the VTR conceptual design PRA, a VTR PRA plan was developed and submitted to the DOE through reference as part of the VTR SDS. The VTR PRA plan was approved as part of the DOE approval of the VTR SDS.

### 3.1.2 ASME/ANS Non-LWR PRA Standard

DOE STD-1628-2013 refers to other industry standards regarding technical PRA requirements. For non-light water reactors (non-LWRs), the standard refers to the ASME/ANS non-LWR PRA standard [19]. The ASME/ANS standard, which was released for trial use[4] in 2013, is an integral standard covering PRA technical elements from initiating events to offsite consequence and internal and external hazards. Table 3-1 outlines the 18 technical elements included within the standard. The ASME/ANS standard provides requirements regarding *what* must be done to develop a PRA but does not specify *how* it should be done.

**Table 3-1: ASME/ANS Non-LWR PRA Standard Elements**

| PRA Elements | Scope of Groups | | |
|---|---|---|---|
| | Internal Events | Internal Hazards | External Hazards |
| Plant Operating State Analysis (POS) | ✖ | ✖ | ✖ |
| Initiating Events Analysis (IE) | ✖ | ✖ | ✖ |
| Event Sequence Analysis (ES) | ✖ | ✖ | ✖ |
| Success Criteria (SC) | ✖ | ✖ | ✖ |
| Systems Analysis (SY) | ✖ | ✖ | ✖ |
| Human Reliability Analysis (HR) | ✖ | ✖ | ✖ |
| Data Analysis (DA) | ✖ | ✖ | ✖ |
| Internal Flood PRA (FL) | | ✖ | |
| Internal Fire PRA (FI) | | ✖ | |
| Seismic PRA (S) | | | ✖ |
| Other Hazards Screening Analysis (EXT) | | | ✖ |
| High Winds PRA (W) | | | ✖ |
| External Flooding PRA (XF) | | | ✖ |
| Other Hazards PRA (X) | | | ✖ |
| Event Sequence Quantification (ESQ) | ✖ | ✖ | ✖ |
| Mechanistic Source Term Analysis (MS) | ✖ | ✖ | ✖ |
| Radiological Consequence Analysis (RC) | ✖ | ✖ | ✖ |
| Risk Integration (RI) | ✖ | ✖ | ✖ |

## 3.2 VTR PRA Development Phases

As described in Section 2.1, under the DOE, reactor authorization is a phased approach based on CD levels. Because of this structure and the use of risk-insights in the VTR design process, the development of the VTR PRA is also divided into phases to align with major authorization safety document submittals, as outlined in Table 3-2.

[4] DOE STD-1628-2013 references the 2013 trial use version of the ASME/ANS non-LWR PRA standard. However, a new version of the standard has been developed based on trial use feedback, including that from the VTR project, and was recently approved by the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) and the American National Standards Institute (ANSI). The revised version is now available as ANSI/ASME/ANS RA-S-1.4-2021 [21]. The updated version of the standard will be utilized for all future VTR PRA development efforts.

**Table 3-2: VTR PRA Development Phases**

| PRA Development Phase | Critical Decision Level | Authorization Document |
|---|---|---|
| Conceptual Design Phase | CD-1 | Conceptual Safety Design Report (CSDR) |
| Preliminary through Final Design Phase | CD-2/3 | Preliminary Safety Analysis Report (PSAR) |
| Start of Operation Phase | CD-4 | Final Safety Analysis Report (FSAR) |
| Operational Phase | Operation | (VTR Living PRA) |

For the conceptual design phase, VTR PRA development focused on the analysis of full-power internal events, with preliminary, scoping analyses for select hazards, as presented in Table 3-3. Radionuclide sources considered in the analysis include the active core, spent fuel stored within the vessel, and sources associated with ex-vessel purification systems. The VTR conceptual design PRA serves several purposes. First, the PRA provides insights for risk-informed design decision-making, including an understanding of approaches to mitigate hazard risks. Second, the PRA contributes to the risk-informed activities of the LMP process described in Section 2.2. At the conceptual design stage, this includes an initial identification and categorization of SBEs and the initial classification of SSCs. Such information is also utilized to inform decisions regarding DID and the design requirements of the SSCs. The results of these analyses support the development of VTR CSDR.

**Table 3-3: VTR Conceptual Design PRA Attributes**

| Phase | Scope | Plant Operating States | Purpose |
|---|---|---|---|
| Conceptual Design Phase | • Internal Events<br>• Preliminary Internal Hazards<br>• Preliminary External Hazards | Full Power | • Initial identification and categorization of SBEs<br>• Initial classification of SSCs<br>• Design requirements of SSCs<br>• Risk-informed design decision-making (including mitigating hazard risks) |

## 3.3 PRA Structure

Regarding the structure of the PRA, VTR follows the guidance from the ASME/ANS non-LWR PRA standard and does not follow the Level I, II, III PRA nomenclature of LWR PRAs. Given the nature of advanced reactors and the requirements of risk-informed licensing and authorization approaches, an integrated analysis is utilized, as shown in Figure 3-1, which covers from initiating events to offsite consequences.

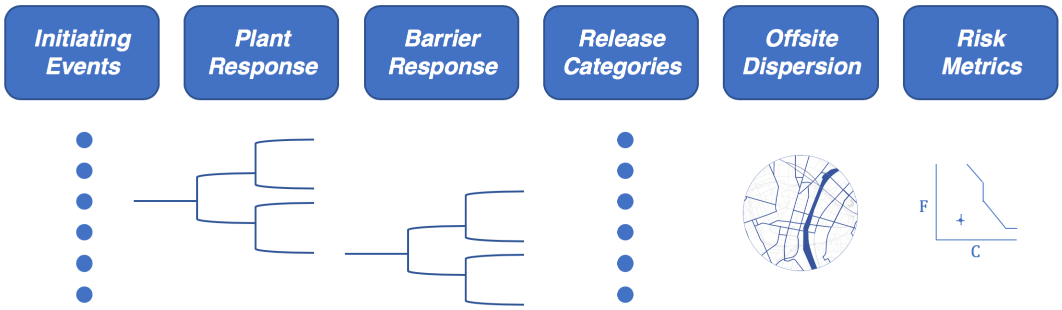


**Figure 3-1: Overview of VTR PRA Structure**

The process begins with the selection of initiating events (IEs), which feed into plant response event trees. These event trees model the response of two safety functions, reactivity control and heat removal, as shown in Figure 3-2. If the transient sequence results in potential damage to reactor fuel (or other radionuclide containing components), then the sequence transfers to a barrier response event tree. This event tree examines the radionuclide retention safety function, including the extent of fuel damage and response of radionuclide barriers, such as the primary vessel boundary. The results of the barrier response event tree sequences are a series of release categories, which describe radionuclide release characteristics for the event sequences. The release categories are then utilized to perform offsite dispersion calculations that determine dose to individuals within the plant site boundary and to the public. In the final step, results of the offsite dispersion calculation are coupled with frequency results of the event sequence to provide a point of comparison to risk metrics, such as the F-C curves. The following subsections provide additional detail on each portion of the analysis.

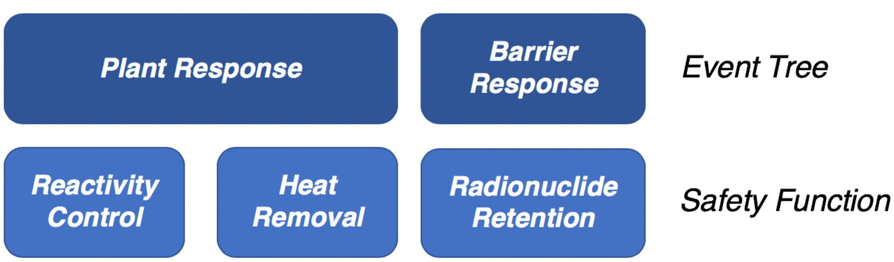


**Figure 3-2: VTR PRA Event Tree and Safety Function Linking**

## 3.4 Initiating Event Analysis

One of the first and basic steps in a PRA is the identification and quantification of the IEs, which are perturbations to the plant that challenge plant control and/or safety systems. IEs have historically been broadly classified as internal events, internal hazards, and external hazards. The VTR conceptual design PRA is limited to at-power internal events, which includes those IEs occurring during power operation either as a direct result of equipment failure, or as the result of errors while performing maintenance, testing, or any other operator action. Internal and external hazards (e.g., seismic events, internal fire/flood) and IEs during shutdown were out of scope for the conceptual design PRA. However, due to their importance in the conceptual design, scoping analyses of sodium fire and seismic hazards have been included and are discussed in Section 3.7.

The IE analysis includes two major steps: IE identification and IE grouping and quantification. A systematic approach was used to identify events that challenge at-power VTR plant operation and require successful mitigation to prevent radionuclide release. This includes an evaluation of light water reactor (LWR) and non-LWR IEs for applicability to VTR, and an assessment of the failure modes and effects of systems that are unique to the VTR design. As defined by the ASME/ANS Non-LWR PRA Standard, internal event IEs are divided into four major categories:

- Transients
- Reactor Coolant Boundary Breaches
- Interfacing Systems Loss of Coolant Accidents (ISLOCAs)
- Special initiators.

“Transients’’ is a broad category that includes events ranging from a plant SCRAM to loss-of-flow or heat sink. Non-LWR specific considerations are included for the IEs and IE groups within these broad categories. For example, the VTR pool-type design generally precludes ISLOCAs and also greatly reduces the potential for reactor (primary) coolant boundary breaches.

Applicable IEs were grouped according to similar attributes, such as plant response and system success criteria (discussed in Section 3.5). The IEs within each group are of adequate similarity such that differentiators (such as specific event timing, effect on operability, and effect on mitigating systems) are considered insignificant at the conceptual design stage of the VTR PRA. Combining IEs into groups reduces the number of event trees that need to be developed and quantified. Through this process, almost 100 initiating events were combined into 11 groups, with some further delineation within groups as needed to capture specific impacts.

## 3.5 Event Sequence Analysis

In PRA, the event sequence (ES) analysis depicts the potential plant response to IEs, which is accomplished through event tree modeling. Each event sequence within the event tree represents a possible plant response to an IE with varying functionality of mitigating systems. Event trees developed for the VTR PRA are based on the IE groups and the necessary plant response to prevent or mitigate radionuclide release. In this section, the event tree structure is discussed first, followed by the methods utilized to determine system success criteria.

### 3.5.1 Event Tree Structure

For the VTR PRA, there are three general groups of event trees:

- **Protected**: Examines the plant response for event sequences in which reactivity control is successful via control rod insertion.
- **Unprotected**: Examines the plant response for event sequences in which control rod insertion has failed.
- **Confinement**: Examines radionuclide barrier performance for both protected and unprotected event sequences that may result in radionuclide release.

As shown in Figure 3-3, IEs are first modeled with the protected event tree. If control rod insertion is unsuccessful, the event sequences transfer to the unprotected event trees. Both protected and unprotected event trees may have event sequences that could potentially result in the release of radionuclides from the fuel. These event sequences are transferred to the confinement event tree, which assesses radionuclide barrier performance. Each event sequence is assigned a release category, which describes the characteristics associated with any potential radionuclide release to the environment. Event sequences with no possibility of radionuclide release are assigned to the OK end state. The outcome of the event sequences were based on the results of simulations performed using the SAS4A/SASSYS-1 code [22].

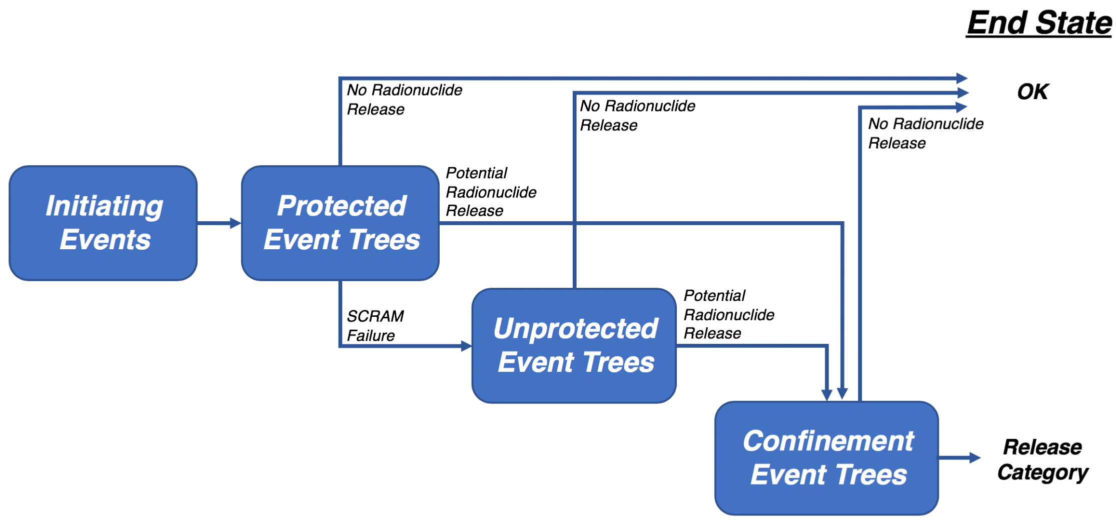


**Figure 3-3: VTR Event Tree Structure**

### 3.5.2 Success Criteria Analysis

For top events in the event trees, two general approaches were utilized for the determination of success criteria. For active systems, the traditional PRA approach of fault tree analysis was used to develop failure probabilities for different event sequences. For the VTR conceptual design PRA, the fault trees varied in level of detail depending on the maturity of the system under consideration. As will be discussed further in Section 5.2.4, for some systems, such as the sodium-to-air heat exchangers, the various potential levels of performance led to the use of non-binary event tree branching to properly capture the impact on system behavior.

For passive system behavior, such as the performance of RVACS or the behavior of inherent reactivity feedback, a mechanistic reliability approach was adopted, as is mandated by the ASME/ANS non-LWR PRA standard. In this approach, system behavior is modeled through computer code simulation (SAS4A/SASSYS-1 in the case of the two systems discussed), with variation of the input parameters based on assessments of likely values for the events sequences considered. Through this detailed uncertainty analysis, the likelihood of the system successfully achieving its desired role is determined.

A detailed overview of the passive system analysis process is shown in Figure 3-4, which integrates the passive system reliability analysis with the determination of system success criteria. The procedure begins with the identification of the system, then bifurcates into the system and success criteria analyses. The system analysis identifies the system mission (such as preventing core damage during scenarios with loss of normal heat removal), then moves into failure modes and influential parameter identification. In parallel, the success criteria analysis utilizes the system mission to identify success metrics, such as peak clad temperature (PCT) limits. Then results of the system analysis parameter selection and screening are used to determine the operational space of the success criteria analyses, including associated uncertainty distributions. Next, the probability of the system meeting the success criteria is determined through modeling using a best-estimate code and uncertainty propagation techniques. In step four, results of the functional failure analysis are combined with results of the traditional physical component failure analysis to create an integrated picture of passive system reliability, before being integrated into the PRA in step five.

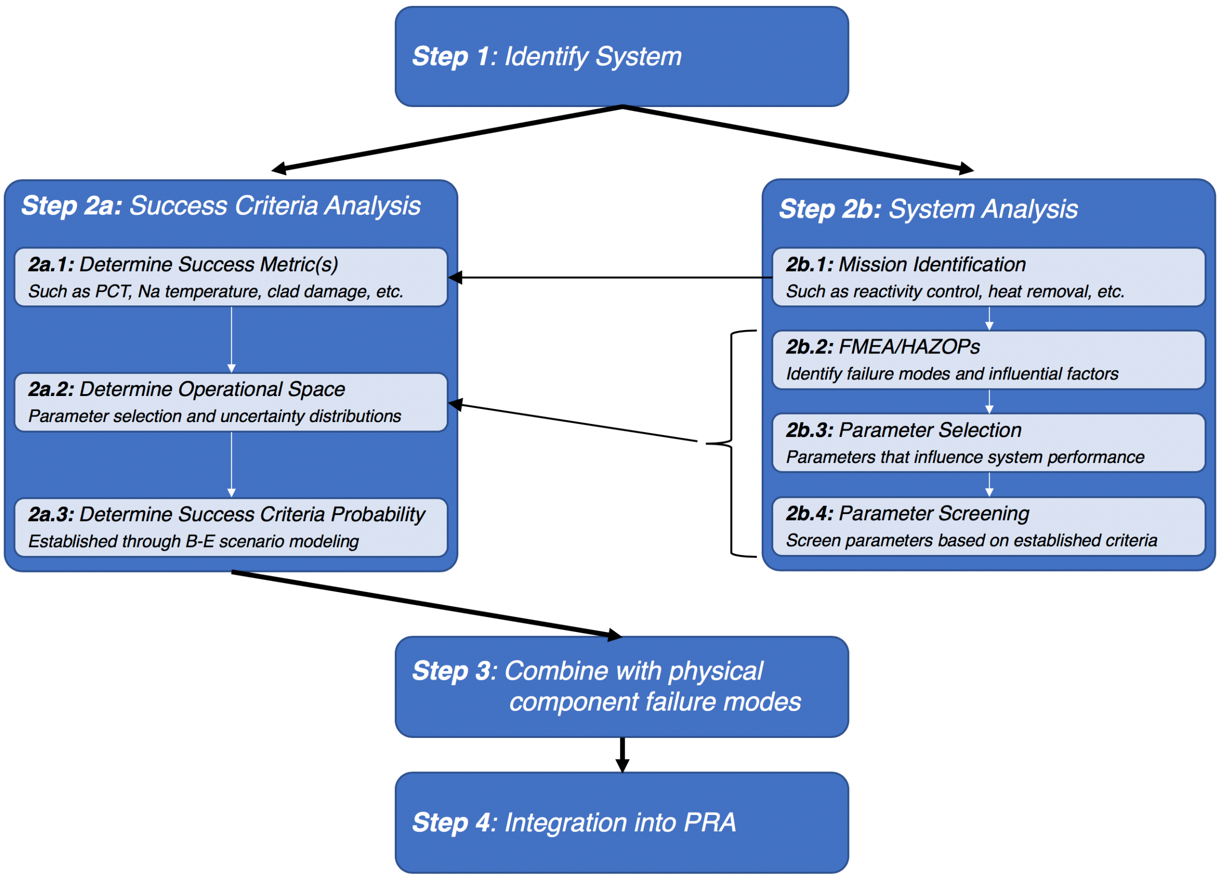


**Figure 3-4: Passive System Reliability and Success Criteria Methodology**

For the VTR conceptual design PRA, insights from the PRISM PRA regarding passive system performance (which utilized the same methodology) were used to provide initial estimates of system reliability and success criteria, with scoping analyses to confirm applicability. Future development phases of the VTR PRA will perform the complete detailed analysis for the VTR design and an automated code linking and uncertainty analysis framework has been developed to allow for repeated analyses in an efficient manner.

## 3.6 Mechanistic Source Term and Radiological Consequence Analysis

For event sequences in the ES analysis that result in the failure of one or more radionuclide barriers, the release of radioactive material from the plant is examined through a mechanistic source term (MST) analysis. In general, the VTR reactor design has five barriers that function to mitigate the release of core radionuclide material to the environment:

1) The metal fuel retains many radionuclides within its matrix.
2) The cladding around the fuel provides a barrier for gaseous fission products (i.e., xenon, krypton) and those present in the bond sodium. Damage to the fuel cladding releases radionuclides to the primary sodium coolant.
3) The sodium coolant acts as a third radionuclide barrier by retaining fission products either by plate-out, chemical solubility or adsorption mechanisms. The radionuclides that are not retained, such as noble gases, will be released to the cover gas space above the sodium hot pool.
4) The reactor vessel is the radionuclide barrier for the primary sodium and radionuclides that are released by the sodium to the cover gas space. As long as the seals around the vessel head penetrations are intact, a small amount of leakage of radionuclides through this barrier into the containment system is modeled based on the nominal leakage rate.
5) The containment system is the final radionuclide barrier. When the containment system is isolated, any radionuclides that travel from the cover gas through the reactor head to the containment system are retained, outside of a small amount of nominal leakage.

Figure 3-5 provides a notational overview of the phenomena that impact the transport and retention of radionuclide material from fuel damage events.

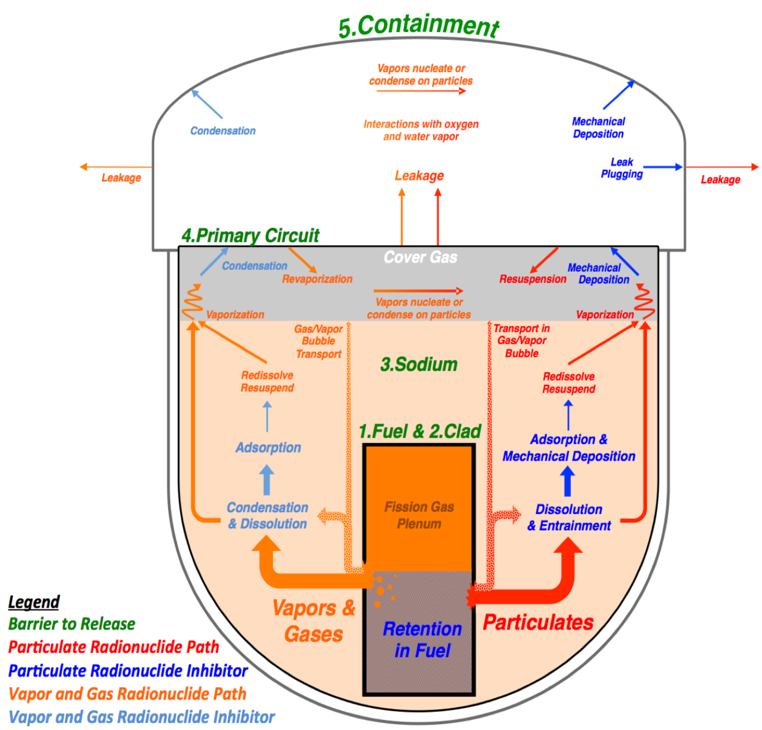


**Figure 3-5: Overview of SFR MST Phenomena**

To facilitate the determination of environmental releases for fuel damage event sequences, the Argonne Simplified Radionuclide Transport (SRT) computer code [23] was utilized, which is capable of modeling radionuclide transport and retention in SFRs. SRT performs a time-dependent analysis of radionuclide transport and retention from damaged fuel elements to the environment. SRT does not perform transient reactor analyses but utilizes the results of system analysis codes, such as SAS4A/SASSYS-1. SRT contains internal models to assess radionuclide migration within the fuel pin, radionuclide release from failed fuel, radionuclide behavior in the sodium pool (including within bubbles), and radionuclide behavior in the cover gas region and containment. For VTR conceptual design PRA, SRT was utilized to perform a preliminary MST of releases from the core, while simplified calculations were used for releases from ex-vessel purification systems.

For the radiological consequence analysis, dispersion parameters are specified by DOE. The internal models within SRT, which conduct X/Q dispersion analyses, were utilized for the assessment of both collocated work and offsite dose analyses using the parameters from DOE.

## 3.7 Scoping Hazard Analyses

Although the VTR conceptual design PRA focused primarily on full-power internal events, scoping analyses were completed for both the seismic and sodium fire hazards, given their importance to system design and safety system strategy.

### 3.7.1 Seismic Analysis

For the scoping seismic analysis, a representative site was selected, and a preliminary seismic hazard curve was developed. The scoping seismic PRA analysis focused on the representative seismically-induced loss-of-offsite-power (LOOP) as an initiator, with the impact on SSC performance assessed through convolution of the SSC fragility and seismic hazard curves. The consequences of the event sequences were evaluated at a high-level, with the conservative release categories assumed for those event sequences with radionuclide release. The analysis aided in the development of the seismic early warning system (seismic SCRAM) of the VTR system, which allows control rod insertion on the detection of the high velocity P-waves before the energetic, but slower S-waves arrive at the plant. In addition, the analysis was beneficial for the preliminary design of SSCs, as it provided insight into generally acceptable levels of seismic robustness for the SSCs.

### 3.7.2 Sodium Fire Analysis

For the scoping sodium fire analysis, a review of potential sources of sodium leaks and fires for the VTR design was performed. The sources of possible sodium leaks included the primary sodium system (reactor/guard vessel, sodium handling), the secondary sodium system, the sodium experimental cartridge loop, and other sodium components (storage vessels, transfer casks, etc.). An event tree approach, based on experience from the EBR-II PRA [15], was adopted to examine the likelihood of sodium leaks and the performance of sodium fire prevention and mitigation systems. While the analysis was preliminary in nature, it provided valuable insight into the different levels of protection needed for the various sources of sodium. Additional detail regarding the experience gained in this process will be discussed in Section 5.2.3.

# 4.0 LMP IMPLEMENTATION DETAILS

The following section details the implementation of the LMP process discussed in Section 2.0 for the VTR conceptual design stage. This includes a step-by-step description of the analyses performed, including SBE identification and categorization, SSC classification, DID analysis, and the IDP process.

## 4.1 SBE Identification and Categorization

As outlined in 2.3, utilizing the LMP process, the VTR PRA is utilized to identify and categorize SBEs based on their frequency of occurrence and guidelines in Table 2-3. There are several steps to translating the event sequences within the PRA to SBEs, as outlined below:

1. **PRA Event Sequences**: Individual event sequences from each event tree in the PRA. Each event sequence includes a frequency of occurrence and an associated consequence (in terms of offsite dose and collocated worker dose)
2. **Event Sequence Families (ESFs):** Groups of event sequences from the PRA with similar characteristics, in terms of initiating event, plant response, and source term[5]. Each ESF includes a frequency of occurrence (the sum of all event sequences within the group) and an associated representative consequence (the bounding consequence of those event sequences within the group). An ESF may contain only a single event sequence.
3. **Safety Basis Events (SBEs):** ESFs with a frequency greater than 1E-6 per year.

Starting with the PRA event sequences, ESFs are formed by collecting those event sequences with similar characteristics. However, some ESFs consist of only a single event sequence. The ESFs represent the complete set of scenarios that are considered for SBE identification and categorization. Those ESFs with a 95$^{th}$ frequency value above 1E-6 are categorized as SBEs. The SBEs are then further categorized into the Anticipated, Unlikely, and Extremely Unlikely categories based on the 95$^{th}$ percentile frequency value and the guidelines in Table 2-3.

In total, the VTR conceptual design PRA contained close to 4,000 individual event sequences. These were grouped into approximately 3,000 ESFs, with most ESFs containing only a single event sequence. A frequency truncation limit of 1E-15 per year was utilized to exclude those ESFs with a frequency far below the threshold for consideration. The truncation process results in approximately 350 retained ESFs for SBE analysis.

The frequency of each ESF was compared to the SBE guidance in Table 2-3 for categorization as SBEs. Of the ~350 retained ESFs, ~40 were above the 1E-6 per year threshold and categorized as SBEs. The ESFs below the threshold were considered tracked events and were retained for further exploration during the importance analyses, DID assessment, and cliff-edge effect analysis discussed in subsequent sections. It is important to note that this assessment does not include the event sequences from the scoping hazards analyses, which were evaluated separately for preliminary insights. It is expected that the number of considered ESFs will grow in

[5] The formal definition of ESFs from the ASME/ANS Non-LWR PRA standard was modified between the 2013 trial use version and 2021 publication, based partially on experience gained during the development of the VTR conceptual design PRA. The revised definition now provides additional flexibility to develop appropriate ESFs based on similar characteristics [21].

future phases of VTR PRA development as additional hazards are included in the formal analysis[6].

Next, the identified SBEs were compared to the consequence target of the F-C curves for their respective category. For the VTR conceptual design PRA, all SBEs satisfied both the offsite and collocated work consequence targets by a significant margin. This is not unexpected for several reasons. The metal-fuel, pool-type design of SFR includes a high-level of passive and inherent safety. However, the risk-informed design process also utilizes the guidance provided by the F-C curves to inform design decisions and maintain a significant level of margin to the consequence target. The latter factor will be discussed in further detail in Section 5.2.

## 4.2 Importance Analysis

A central aspect of the LMP process is the utilization of risk-insights to inform the classification of SSCs. As described in Section 2.3, the VTR project established a series of criteria to aid in the determination of both SC and SS SSCs. The actual implementation of this process consists of a series of importance analyses that explore the role and impact of specific safety functions and SSCs on plant risk. In total, three types of importance studies were performed: function-specific risk achievement studies, success path studies, and safety significant studies. The risk achievement study explores the risk importance of a single function, while the two other studies develop risk importance for a set of functions.

### 4.2.1 Risk Achievement

Risk achievement studies are conducted to provide insight into those functions or SSCs that are being relied upon to ensure the safe operation of the reactor and therefore satisfy the criteria of the F-C curve. The results provide one of the bases for SSC classification by demonstrating which SSCs fulfill the SC classification criterion in Table 4-1.

**Table 4-1: Risk Achievement SSC Classification Criteria**

| SSC Classification | Criteria | |
|---|---|---|
| | **Type** | **Description** |
| Safety Class (SC) | Risk-Informed | • Offsite F-C Curve:<br>o For SBEs greater in frequency than 10-6/yr, an SSC is SC if its removal causes the SBE to violate the F-C curve, when considering a one-by-one removal of SSCs, crediting all remaining SSCs, regardless of safety classification, at appropriate reliability levels (and with appropriate accounting of common cause failure). |

The risk achievement analysis is performed by simulating that major VTR functions (such as the reactor protection system, RVACS, etc.) are unavailable in the PRA event sequences. This is completed within the PRA model by setting the basic events and/or gates to a failed condition. The PRA model is then requantified to examine the impact of system unavailability. If removal of the function/SSC results in ESFs[7] violating the offsite F-C curve, then the analysis indicates that

[6] Although new ESFs may be created due to the inclusion of hazards, as they will represent new initiators, the existing ESFs likely capture many analogous challenges to plant systems.

[7] All ESFs are considered in this analysis (not only SBEs), as the unavailability of a function/SSC may cause an ESF that was previously located in the *beyond extremely unlikely* region to increase in frequency above the 1E-6 per year threshold.

the function/SSC may warrant SC classification. Only the offsite F-C curve was utilized for this analysis.

The risk achievement results were presented in several ways. First, summary results can be provided in a format similar to Table 4-2, which highlights what SSCs prevent an exceedance of the offsite F-C curve for specific SBEs. Second, the results of individual risk achievement studies can be plotted on the offsite F-C curve directly to provide insight into the margin to the F-C curve, or the degree of curve exceedance. The results of the risk achievement study are insightful in a number of ways. For example, through this analysis the specific ESFs driving SC classification are identified. This information is valuable for determining the design and functional requirements for the SSC and avoiding blanket requirements for all SC SSCs.

**Table 4-2: Example Risk Achievement Summary Results**

| ESF | SSC_1 | SSC_2 | SSC_3 | … |
|---|---|---|---|---|
| ESF_1 | ✕ | ✓ | ✓ | . |
| ESF_2 | ✕ | ✓ | ✓ | . |
| ESF_3 | ✓ | ✕ | ✓ | . |
| … | . | . | . | . |

✕-Consequence target cannot be met without SSC
✓- Consequence target can be met without SSC

### 4.2.2 Success Path

Success path studies utilize the PRA event trees to determine the combination of functions/SSCs necessary to maintain *anticipated* and *unlikely* SBEs below the 25 rem offsite dose criteria. This analysis is performed to address the SSC classification criteria in Table 4-3. Although DBAs are only derived from SBEs in the *unlikely* category, all SBEs above 1E-4 per year were examined for completeness. This analysis is another step in the process for determining the set of SC SSCs for the plant.

**Table 4-3: Success Path SSC Classification Criteria**

| SSC Classification | Criteria | |
|---|---|---|
| | **Type** | **Description** |
| Safety Class (SC) | Risk-Informed | • Offsite F-C Curve[1]:<br>○ For DBAs derived from SBEs in the "Unlikely" category (<10-2 to >10-4/yr), an SSC is SC if it is necessary for the DBA to satisfy the 25 rem consequence limit when utilizing deterministic, prescriptive analysis of the event sequence and crediting *only* SC SSCs. |

To perform the success path analyses, differing sets of functions/SSCs were selected, with initial selection based on the insights of the risk achievement studies. For each set, the 25 SBEs in the *anticipated* and *unlikely* categories were re-evaluated against the 25 rem offsite dose criteria by setting all other functions/SSCs to the failed state. For the VTR conceptual design studies, five different sets of SSCs were evaluated through this process. Of the five sets, only two successfully prevented an exceedance of the 25 rem offsite dose criteria for all 25 selected SBEs. The results of the success path analysis can be summarized in a table such as Table 4-4.

**Table 4-4: Example Success path Summary Results**

| SBE[1] | Set-1 | Set-2 | Set-3 | … |
|---|---|---|---|---|
| SBE_1 | × | ✓ | ✓ | . |
| SBE_2 | × | ✓ | ✓ | . |
| SBE_3 | ✓ | × | ✓ | . |
| … | . | . | . | . |

×- Consequence target cannot be met with only designated SSC Set
✓- Consequence target can be met with only designated SSC Set
[1] Only contains SBEs in the *anticipated* and *unlikely* categories

### 4.2.3 Safety Significance

As highlighted in Section 2.4, there are several risk informed criteria regarding the designation of SS SSCs. Specifically, there are two criteria, presented in Table 4-5, that involve a demonstration of adherence to both the offsite and collocated worker F-C curves when only crediting SC and SS SSCs. The analyses to support these criteria are similar to the success path assessments of the previous subsection.

**Table 4-5: Safety Significance SSC Classification Criteria**

| SSC Classification | Criteria | |
|---|---|---|
| | **Type** | **Description** |
| Safety Significant (SS) | Risk-Informed | • Offsite F-C Curve:<br>○ For SBEs in the “Extremely Unlikely” region (<10-4 to >10-6/yr), an SSC is SS if its removal causes the SBE to violate the F-C curve when considering *only* SC and SS SSCs appropriate for the SBE.<br>• Collocated Worker F-C Curve:<br>○ For SBEs greater in frequency than 10-6/yr, an SSC is SS if its removal causes the SBE to violate the F-C curve when considering *only* SC and SS SSCs appropriate for the event sequence. |

For the VTR conceptual design studies, the two relevant criteria were examined through separate analyses. First, the SBEs in the *extremely unlikely* category, were re-evaluated utilizing only the SC SSCs from the success path studies and several potential combinations of SS SSCs. As the success path studies resulted in two sets of acceptable SC SSCs, the safety significance analysis examined both possibilities. For each of the two sets of acceptable SC SSCs, different combinations of potential SS SSCs were identified based on preliminary analyses. It’s important to note that the sets of potential SS SSCs were different for the two sets of SC SSCs under investigation, as they included alternative methods of fulfilling certain RSFs. The results of these analyses identified a small set of SS SSCs for each of the two SC SSC sets.

Following the offsite dose safety significance analysis, the collocated worker curve assessment was performed. For this analysis, all SBEs were considered, not just those in the *extremely unlikely* region. To aid in the process, the risk achievement studies were repeated for the collocated worker F-C curve to provide initial insight into what additional SSCs may be needed to satisfy the collocated worker F-C curve. The results of this study did not discover any SSCs that were not already identified in the previous sets of SC and SS SSCs. A separate success path analysis was also performed to confirm that the identified set of SC SSCs would also be capable of satisfying the 100 rem collocated worker dose limit for *anticipated* and *unlikely* SBEs. Although this is not a formal criterion for classification, it provided additional confidence in the selection of

SC SSCs, as they were adequate for preventing an exceedance of the 100 rem collocated worker dose.

Lastly, all SBEs were re-evaluated utilizing only the proposed sets of SS and SC SSCs and compared to the collocated worker F-C curve. For each of the two SC SSC sets, two sets of two additional SS SSCs were identified that would satisfy the collocated F-C curve criteria.

### 4.3 Integrated Risk, Risk Significance, and Cliff-Edge Effects

As highlighted in Section 2.6, the VTR safety basis considers three integrated or cumulative risk metrics. While these are not formal authorization criteria, they provide useful guidance regarding plant safety and also contribute to the determination of risk significance and therefore the classification of SSCs, as presented in Table 4-6.

**Table 4-6: Risk Significance SSC Classification Criteria**

| SSC Classification | Criteria | |
|---|---|---|
| | **Type** | **Description** |
| Safety Significant (SS) | Risk-Informed | • SSC performs risk significant function, where risk significant is defined as:<br>○ If the SSC makes a significant contribution (>1% of the limit value) to the cumulative risk metrics. |

The first cumulative risk metric, derived from 10 CFR 20, examines the total frequency of exceeding 0.1 rem offsite dose across all ESFs and whether it exceeds 1 per plant year. This metric is calculated by summing the frequency of all ESFs that result in an offsite dose greater than or equal to 0.1 rem. For the CSDR, the events most likely to impact this metric (e.g., fuel and waste handling events) were generally evaluated deterministically and not yet included in the PRA. However, future phases of the VTR PRA will explicitly evaluate these events. The two other cumulative risk metrics are derived from the QHOs (and analogous DOE guidance) and are calculated by summing the risk of early fatality and latent cancer across all ESFs.

The next step involved a determination of risk significant SSCs according to those SSCs that contribute>1% of the limit value to the cumulative risk metrics. As the VTR conceptual design met all three cumulative risk metrics by a considerable margin, a qualitative examination demonstrated that no additional SSCs would be considered risk significant based on this criterion. In addition, no risk significant SBEs were identified based on the guidance in Section 2.5.

The cliff-edge effects analysis examined those high consequence ESFs at frequencies between 1E-6 and 1E-8 per year. The only ESFs in this area were preliminary ESFs developed as part of the scoping seismic assessment. As expected, these ESFs were identified for further analysis as part of the detailed external hazard assessment in future development phases and the use of less conservative analysis approaches and assumptions will likely change their positioning on the F-C curves.

### 4.4 Defense-in-Depth Adequacy Analysis

DID assessments in support of the VTR CSDR took several forms. Historically, extensive effort has been expended in the development of metal-fuel pool-type SFRs to provide a high-level of reliability and robustness to potential plant transients. These features are not discussed here, as overviews are available elsewhere [24, 25]. The focus here is on those additional analyses conducted as part of the VTR safety basis to evaluate DID adequacy. The guidance from [26], presented in 2.7, provided the basis for both preliminary qualitative and quantitative assessments of plant capability and programmatic DID.

In terms of overall guidelines, the VTR conceptual design analysis demonstrated that the VTR design satisfied the F-C curves with significant margin. In addition, PRA analyses and reviews from the IDP (discussed later) confirmed that the design did not rely solely on a single feature or system to satisfy the five layers of DID. Similar constraints are in place regarding the overreliance on single features in both the *unlikely* and *extremely unlikely* regions. These objectives could be met due to the multiple passive avenues available to complete fundamental safety functions, such as heat removal through the secondary loop in passive mode or passively through the RVACS. Additional analyses centered on examining the frequency of demands on SC equipment and identifying key areas of uncertainty.

While plant capability DID is fairly straightforward to assess given the quantitative and qualitative criteria in Table 2-5, programmatic DID is more abstract and involves considerations of margin and various types of uncertainty (parameter, modeling, etc.). At the conceptual design phase, only a basic review of programmatic DID was performed by the IDP, focused on major areas of uncertainty that could impact key decisions regarding SBE categorization and SSC classification. In some cases, the IDP made conservative selections regarding SSC classification or preliminary SSC design requirements to address the level of uncertainty, with a goal of further refinement in the next development phase.

## 4.5 Integrated Decision Panel (IDP)

As highlighted in Section 2.8, the IDP is a critical element of the RIPB decision-making process. In preparation for activities during the conceptual design phase, an IDP procedure document was prepared that outlines the purpose, responsibilities, membership, process, training, and records of the IDP. IDP membership included team members from the following areas, in addition to a scribe:

- Safety Analysis
- Design Engineering
- System Engineering
- Risk Management (i.e., Probabilistic Risk Assessment (PRA))
- Operations and Maintenance
- Nuclear Safety

The IDP procedures also provided guidance for the key roles of the IDP, including the review of SBEs, the classification of SSCs, and the evaluation of DID adequacy.

Regarding SBE identification, categorization, and evaluation against the F-C curves, the IDP members have the following considerations:

- Is the selection of IEs and event sequences reflected in the SBEs sufficiently complete?
- Are the uncertainties in the estimation of SBE frequency, plant response to events, mechanistic source terms, and dose well characterized?
- Are there sources of uncertainty not adequately addressed?
- Have hazards or potential IEs not characterized within the PRA been supported by hazard evaluation and appropriate bounding events identified?

- Have all risk-significant SBEs been identified and properly grouped into bounding event families?
- Has the PRA evaluation provided an adequate assessment of "cliff edge effects?"
- Have protective measures to manage the risks of multi-reactor module and multi-radiological source event sequences been adequately defined?

If the evaluation identifies unacceptable answers to any of these questions, additional SBEs may be identified or compensatory actions considered, depending on the risk significance of the SBE. The above SBE review includes a PRA/LMP evaluation of cliff edge effects.

For SSC classification, the following questions should be considered as part of evaluating and documenting SSC classification information:

- Is the technical basis for identifying the RSFs adequate?
- Is the selection of the SC SSCs to perform the RSFs appropriate?
- Have protective measures to manage the risks of all risk-significant SBEs been identified, especially those with relatively high consequences?
- Have protective measures to manage the risks for all risk-significant common-cause IEs such as support system faults, internal plant hazards such as fires and floods, and external hazards been identified?
- Is the risk benefit of all assigned protective measures well characterized, e.g., via sensitivity analyses?

The IDP assesses DID adequacy from both a plant capabilities and a programmatic perspective. Unlike the plant capabilities for DID that can be described in physical terms and are amenable to quantitative evaluation, the programmatic DID adequacy should be established using engineering judgment by determining what package of DID attributes are sufficient to meet the objectives below:

- Assuring that adequate margins exist between the assessed LBE risks relative to the F-C Target including quantified uncertainties
- Assuring that adequate margins exist between the assessed total plant risks relative to the cumulative risk targets
- Assuring that appropriate targets for SSC reliability and performance capability are reflected in design and operational programs for each LBE
- Providing adequate assurance that the risk, reliability, and performance targets will be met and maintained throughout the life of the plant with adequate consideration of sources of significant uncertainties.

During the programmatic review, the PRA/LMP results should provide sensitivity analysis to demonstrate how the LMP results would change given assumptions or uncertainties are revised.

Margin adequacy includes a determination that the appropriate codes were applied to safety-SSCs (included in SS safety categories) and that the most demanding parameters for that component, conservatively estimated, have been used for the design point. For SSCs classified as SS, the special treatment is identified and reviewed by the IDP.

The IDP was convened multiple times during the VTR conceptual design stage. The first IDP was performed following the initial completion of the VTR conceptual design PRA and initial performance of the LMP process. This step was important for establishing the preliminary set of SBEs and the initial classification of SSCs. It also identified high priority topics for further analysis and design consideration. These points are discussed further in Section 5.3.6. A secondary IDP was held in preparation for the drafting of the CSDR and addressed design changes and new analyses performed since the first meeting.

# 5.0 CONCEPTUAL DESIGN EXPERIENCE

Valuable experience was gained utilizing the LMP approach in support of the CSDR and post-CSDR analyses. This section provides an overview of key insights from the process, including details associated with the technical implementation of the approach, RIPB design considerations, and also lessons learned regarding both the application and management of the process.

## 5.1 Technical Implementation

The following section provide additional detail regarding the technical implementation of the RIPB LMP approach, such as the tools utilized to develop the PRA and the quantitative analyses. Also, insights gained during the technical implementation are summarized.

### 5.1.1 VTR PRA Development

The VTR PRA model was developed using the EPRI Integrated Risk Technology suite of tools that include the computer codes CAFTA Version 6.0b [27] and PRAQuant Version 5.2 [28], and FORTE. These computer codes have been demonstrated throughout the industry to produce appropriate results. No method specific limitations have been identified with regard to the software tools or the methodology implemented to quantify the model.

Using CAFTA, the VTR PRA model was developed by importing all model event trees, system fault trees, initiating events, passive system reliability results, and associated basic event databases. These PRA inputs were previously discussed in Section 3.0 and combining all this information within CAFTA allows for the usage of PRAQuant, where a top logic fault tree was created out of the VTR PRA model. All system fault trees were merged with the top logic fault tree. The model was then quantified to arrive at event sequence frequencies for the various release categories discussed in Section 3.0. The derived event sequences and associated frequency and consequence were then utilized for the LMP assessments outlined in Section 4.0. Due to the nature of the RIPB LMP process and the evolving conceptual design, PRA quantification was performed iteratively, in coordination with design studies and the LMP sensitivity analyses.

### 5.1.2 PRA Analysis Experience

One of the major changes of the VTR conceptual design PRA quantification process compared to traditional PRA analyses was that all event sequences, including the OK sequences (*i.e.*, those with no radionuclide release), were quantified. This information is necessary for completing the RIPB LMP analyses, such as the identification of SBEs, and results in an expansion of the number of event sequences tracked throughout the process (part of the ~4,000 individual event sequences highlighted in Section 4.1).

As outlined in Section 4.0, implementing the RIPB LMP approach requires conducting a series of sensitivity analyses that explore the importance of particular event sequences, functions, and SSCs. While the utilized PRA analysis tool, CAFTA, is capable of performing sensitivity analyses, in many cases additional analysis was necessary to obtain the results required for the LMP assessments. For this purpose, custom scripts were developed in the R programming language that utilized the output from CAFTA and performed additional analyses, such as the comparisons against the F-C curves or the identification of subsets of SSCs for SC and SS designation. The development of the scripts and performance of the analyses required additional resources and time.

In addition to the challenges associated with the capabilities of existing PRA tools to conduct LMP analyses, questions also arose regarding the typical approaches to PRA development. For construction of the VTR conceptual design PRA, the small event tree/large fault tree method was utilized. In this approach, top events in the event trees are kept to a minimum, while the majority of detail regarding function/system performance is contained within the fault tree. However, this structure presents a possible issue when performing LMP sensitivity/importance analyses and SBE identification. If multiple functions or SSCs are contained within a single fault tree and not represented as separate top events, the event sequence analysis may have difficulties capturing the individual impact (the discussion of reactivity control systems in Section 5.2.1 is one potential example). While workarounds to address this problem and capture fault tree information are possible, they do require additional human effort.

## 5.2 Risk-Informed Design Experience

As noted in previous sections, the RIPB LMP approach was utilized for both design and authorization decisions. The use of RIPB information during the reactor design process to inform design decisions is a balance between design maturity and flexibility in the design. For the VTR conceptual design phase, there is increased flexibility in the design but also a reduced level of detail within the PRA models. Therefore, the RIPB design considerations during the conceptual design stage primarily focused on high-level system functionality and reliability.

### 5.2.1 Reactivity Control Systems

Given their importance in fulfilling one of the RSFs, reactivity control systems were an initial focus of VTR conceptual design activities. The VTR design includes both active and passive approaches for reactivity control. The metal fuel core is designed to achieve inherent negative reactivity feedback with increasing core temperatures. However, the active reactivity control systems are relied upon for normal operation and are also credited as a SC means of reactivity control during transient event sequences.

The initial VTR design included both a reactor protection system (RPS) and an independent, diverse protection system (DPS), as an approach to achieve very high reliability of active reactivity control. Analyses of this system configuration demonstrated a high level of protection against failures to provide reactivity control. However, as the design of the systems matured, the complexity of the arrangement between the two systems and the convolution of their interaction resulted in increasingly complicated plant operations and procedures. Such complexity could potentially introduce unforeseen operational scenarios and was generally in conflict with the goal to simplify plant design and operations.

The main benefit of the DPS to active reactivity control reliability was its contribution as a diverse pathway for control rod insertion, which reduced the potential impact of common cause failure (CCF). In an attempt to simplify plant design and operation, a design study examined the elimination of the DPS with a modification to the RPS to increase the diversity within the system. By changing the logic of the RPS and diversifying the selection of components, the desired redundancy and diversity could be achieved solely within the RPS. The results of the risk studies comparing the options were an important contributor to the design decision, which resulted in the elimination of the DPS and a substantial simplification to the reactivity control strategy for the reactor.

### 5.2.2 Electromagnetic Pump Performance

The VTR design utilizes electromagnetic (EM) pumps for both the primary and secondary sodium loops. EM pumps have considerable benefits in terms of system simplicity, given a lack of moving parts and reduced system penetrations. However, EM pumps do not have the inherent inertia of centrifugal pumps, which allows centrifugal pumps to continue to provide a reduced level of flow following a loss of power. Therefore, coastdown mechanisms, such as flywheels or batteries, are typically included with EM pump systems to ensure a gradual decrease in pump flow to facilitate transition to natural circulation during a loss-of-pump-power event. The VTR conceptual design phase included several studies regarding the design of the coastdown mechanisms for the primary EM pumps, due to the impacts on plant cost, layout, and SSC classification.

To aid in the design studies, the VTR PRA and associated safety analyses examined the repercussions of primary EM pump coastdown failure for a variety of transient event sequences. The goals were to establish which scenarios required successful performance of the coastdown mechanisms and the associated requirements of the system. The VTR PRA provided insight regarding those transient scenarios that should be considered in the analysis and those that could be excluded given a sufficiently low frequency of occurrence. The results of the analysis also contributed in a complementary fashion, as the requirements associated with the necessary number of operational coastdown mechanisms also guided the reliability requirements for the system. The latter had consequences regarding the necessary level of independence between the coastdown mechanisms. The risk-informed insights developed the basis for the SC classification of the EM pump coastdown mechanisms. Lastly, the analyses also informed the design parameters of the coastdown mechanisms themselves (such as runtime, power, etc. [29]).

### 5.2.3 Sodium System Design

As highlighted in Section 3.7, VTR PRA activities in the conceptual design stage included a preliminary assessment of the hazard associated with sodium leaks and fires. The central goal of the analysis was to inform the requirements associated with the sodium fire prevention and mitigation system. Scoping analyses were performed regarding the potential repercussions associated with sodium fires in different areas of the plant, including their impact on SC or SS SSCs. The results of the scoping analyses were utilized in several ways. First, when possible, the plant design was modified to minimize the consequences of sodium leaks/fires by separating equipment and ensuring adequate barriers are in place. Second, when a potential impact on a SC or SS SSC could not be eliminated, the PRA analyses informed the reliability requirements associated with the sodium fire prevention and mitigation system.

One location where the latter situation occurred is within the head access area, which contains piping for the secondary sodium system along with the control rod drive mechanisms and interfacing walls with RVACS. To ensure the functionality of the control rods and RVACS in the event of a sodium leak, various protections were put into place. First, double-walled piping was selected for the sodium piping runs within the head access area to provide an additional barrier to release. In addition, leak monitoring is also placed between the piping layers to allow for early detection of leaks. Similarly, the piping is sloped in a fashion that results in any leaked sodium moving away from the reactor head and towards a drain tank located outside the head access area. Additional protections were also put in place to ensure that sodium leaks could not impact RVACS, such as providing seals around the reactor vessel to avoid sodium drainage into RVACS and adding partitioning walls to separate RVACS air ducts from the head access area volume.

For other areas that contained sodium piping, the use of piping leak-jackets or only single-walled piping was deemed sufficient due to minimal impact on SC or SS SSC functionality. These analyses also intersected with requirements regarding investment protection, where the consequences of sodium fires may not be tolerable due to their impact on asset preservation rather than plant safety. While this is not a formal consideration within the LMP process, as it is not directly a regulatory matter, similar risk-informed analyses can be performed but with a different end goal.

### 5.2.4 Sodium-to-Air Heat Exchanger Operation

A unique aspect of VTR is the utilization of sodium-to-air heat exchangers (SAHXs) for the rejection of reactor power to the environment during normal full-power operation and shutdown. During normal operation, the SAHXs operate in an active mode, where large fans blow air across the sodium tubes within the SAHXs and the sodium flow is circulated in the secondary loop by EM pumps. However, during loss of power scenarios, the secondary sodium system operates in a passive fashion, where natural circulation of air across the sodium tubes removes heat and sodium circulation within the secondary circuit is also dependent on natural circulation. As the secondary system can operate passively, there is no need to provide emergency electrical power to the secondary EM pumps or SAHX blowers during loss of power scenarios, reducing the need for diesel generators.

The ability of the SAHXs to passively remove heat from the reactor system during loss-of-offsite-power events plays a critical role in the VTR safety basis by reducing the likely demands on the RVACS, which is the SC heat removal pathway. Reducing demands on SC SSCs is one of the key tenants of the LMP DID strategy (see Table 2-5). Therefore, it is important to properly credit passive functionality of the secondary loop and SAHXs in the plant safety analyses, however, there is complexity to such analyses. The VTR secondary system includes ten SAHXs (five on each secondary loop), which is beneficial to the VTR safety basis but also potentially complicates system operation.

Due to the number of SAHXs in the secondary system and the ability to operate passively, there is a possibility of overcooling the sodium during plant shutdowns. To prevent such scenarios, dampers on the SAHXs can be used to limit airflow and reduce heat removal. Within the PRA, the operation of SAHXs is a balance between the prevention of overcooling and the preservation of the SAHX heat removal pathway, if required. PRA assessments examined the options for SAHX system operation, such as the automatic closure of system dampers on reactor SCRAM, and the potential impact on system availability. These analyses included an examination of human reliability and potential errors in operation. The result was a system operational procedure that ensures that the functional ability to remove sufficient heat through the SAHXs is preserved until a time when it would no longer be necessary, even with additional failures within the plant. This approach is in contrast to a procedure that would first disable the system through damper closure, then attempt to re-establish heat removal through control room actions, if the need were to arise.

## 5.3 Implementation Lessons Learned

As is the case whenever a new method or approach is implemented, new challenges arise, and lessons are learned. This section summarizes some of the key lessons learned during the implementation of the LMP approach for the preparation of the VTR CSDR, including the types of analyses conducted, addressing external events, properly crediting SSC reliability, and the management of a complex system of analyses and interfaces.

### 5.3.1 Establishing Analysis Purpose

A central challenge in the use of risk-informed approaches early in the design process is properly establishing the goals and expectations of the specific PRA analysis. As PRA was only widely adopted after the construction of most U.S. nuclear power plants, it has historically been utilized as a tool for assessing the risk associated with existing facilities. In this traditional role, the systems and components are established and significant effort is expended to accurately capture their behavior and reliability within the PRA. This is quite different than a PRA performed at the conceptual design stage, where many systems have yet to undergo detailed design. Some systems may only be conceptual and at the functional level, with many options available for achieving the selected safety function.

In the context of PRA analyses performed at the conceptual design stage, there may be different goals motivating the analysis, as outlined below:

- **Realistic Assessment of Plant Performance**: SSC reliability is estimated using available data and methods, with a goal to derive a realistic assessment of plant risk. Examples include PRA analyses supporting the categorization of SBEs.
- **Derivation of SSC Reliability Requirements**: Utilizing the goal value of associated risk metrics (like the F-C curve), the assessment works backwards to derive the SSC reliability that is necessary for achieving the risk criteria. The analysis supports SSC design and requirements specification.

As highlighted, a PRA analysis completed early in the conceptual design stage could be performed in pursuit of different goals, either an attempt at accurately assessing the true frequency of event sequences by capturing the probable reliability of the conceptual system or components, or an evaluation to establish the necessary reliability requirements of an SSC based on risk criteria. In the former, efforts are taken to properly represent the reliability of the system/component using available data and methods. In the latter, the goal values of the output risk metrics (or other criteria) are utilized to work backward and determine the system reliability necessary for the satisfaction of the criteria.

Although both analysis approaches are conceptually straightforward, implementation can be difficult if the goals of the analysis are not clearly established. For example, a single PRA analysis might include aspects of both categories, where the design requirements of a single SSC are being derived but in conjunction with realistic estimates of reliability for other SSCs in the event sequence. As highlighted above, many of the PRA analyses conducted as part of the VTR conceptual design stage focused on the establishment of system reliability requirements. The contrast in methods to establish SSC reliability (data estimates versus derivation based on a risk goal) often led to confusion and could impact the implementation of PRA insights. Therefore, it is imperative that the purpose of the analysis and methods utilized to establish SSC reliability are effectively communicated with transparency between teams.

### 5.3.2 External Events Considerations

While the VTR conceptual design PRA consisted of only a scoping seismic hazard analysis, the consideration of external hazards is an important factor for advanced reactor design and authorization/licensing. This is primarily due to the ability of certain external hazards to defeat multiple layers of DID simultaneously. In comparison to the historical treatment of external hazards in reactor licensing, which generally utilized prescriptive and conservative approaches, the RIPB approach of the LMP method provides a realistic, detailed view of plant performance. These insights are valuable for informing the plant strategy for specific hazards, the design of SSCs, and also for the evaluation of DID adequacy. However, assessing the plant response to very low frequency external hazards can be challenging, given uncertainties in both the hazard itself and the behavior of SSCs under extreme loads. Insights on these matters gained during the VTR conceptual design phase have partially motivated separate DOE:NE efforts to further explore the treatment of low frequency external events under a RIPB framework, and ref [30] contains additional detail on the topic, including the challenges and opportunities associated with the approach. This topic will be explored further in the next phase of VTR PRA development, which includes detailed probabilistic analyses of external hazards.

### 5.3.3 Addressing Common Cause Failure

Advanced reactors, such as VTR, have generally reduced the frequency of internal event initiators through the use of inherent or passive safety systems and redundancy and diversity of active systems. Therefore, radionuclide release is typically only possible in scenarios with multiple system failures. As independent system failures as extremely unlikely to occur, CCF is often the dominant failure pathway for active systems but can be one of the most difficult to assess accurately, especially for novel system designs. The benefits of increased redundancy on system reliability decrease with each additional level. Diversity aids in the reduction of CCF mechanisms, as outlined in Section 5.2.1 with the RPS, but it can be difficult to specify diversity requirements for advanced reactor systems. Available suppliers of equipment may be unknown (if any exist), and other typical CCF pathways, such as shared installers, maintenance programs, etc., are at a level of detail not yet specified for the systems. For other systems, the operational experience may not be sufficient to properly understand potential CCF mechanisms. This often results in the use of likely conservative assumptions regarding CCF probability or the implementation of alternative system designs to achieve the desired levels of reliability.

### 5.3.4 Properly Crediting System Redundancy

Related to the CCF topic is the approach for properly crediting system redundancy and its impact on SSC classification, with an example presented here regarding the SAHXs. As highlighted in Section 5.2.4, for VTR, heat removal through the secondary system operating in passive mode is important to reducing the number of demands on RVACS and generally improves the safety of the plant by providing another highly reliable means of heat removal. These findings were supported by the VTR PRA analysis and the LMP sensitivity studies, which resulted in a SS classification for heat removal through the secondary system. However, the heat removal capabilities of the secondary system, even in passive mode, are generally much greater than what is required to ensure safe operation during loss of power scenarios. Therefore, only a subset of the ten SAHXs is necessary to achieve the associated success criteria. Classifying all ten SAHXs as SS and imposing the related design and operation requirements is likely unnecessarily burdensome. In contrast, requiring SS requirements for only a subset of SAHXs is also challenging, as that could unintentionally limit the SAHXs credited in associated safety analyses, creating an unrealistically conservative view of system reliability and availability, and results in an essentially arbitrary selection of SAHXs for increased oversight. For the CSDR, identifying the importance of secondary system heat was adequate, with an optimal solution to SAHX classification strategy pending future discussions with the authorization body in the next development phase.

### 5.3.5 Communication of Results

A question that presented itself multiple times through the VTR conceptual design phase was how to properly and efficiently capture and communicate the results of the VTR PRA and LMP analyses. As outlined in Section 4.0, detailed and sophisticated analyses are performed to inform key decisions regarding the design and safety basis. However, the results and reasoning for them must be communicated to multiple parties, such as the design and operations teams within the project and also the authorization body. The quantity of information produced is a concern, along with presenting it to those who may not have a background in reactor safety and risk assessment.

Even within the team conducting the PRA and LMP analysis, it was necessary to develop techniques to visually communicate detailed insights quickly by highlighting the key insights.

Several key lessons were learned through this process. First, it is necessary to translate typical PRA nomenclature to a format that is intuitively accessible. For example, reference to specific ESFs typically included the identifier from the PRA model, which is difficult to interpret for those not familiar with the underlying model. Therefore, it was necessary to provide a description of each ESF in terms of initiator and plant response (success/fail of SSCs) to properly capture the information needed. Similarly, for each SSC requirement derived from the LMP analysis, it was important to highlight the reasoning and the supporting analyses. This was especially critical for those SSCs that had a safety classification due to multiple functions, such as the reactor vessel, which is SC due to its role as a primary system boundary but also part of the heat removal pathway for RVACS.

Lastly, there is an additional challenge regarding the appropriate manner of distilling information for inclusion in authorization documents, such as the CSDR. Such documents should only contain the key findings of the analyses, but it can be difficult to present such information without including many of the supporting details. Insights captured as part of this process were communicated to the TICAP as part of a tabletop exercise conducted by the VTR team [31].

#### 5.3.6 IDP Experience and Insights

As described in Section 4.5, multiple IDPs were conducted as part of conceptual design phase activities. The IDP process proved to be valuable for a number of reasons. First, as highlighted in Section 4.0, the RIPB analyses conducted as part of the LMP provide guidance regarding plant design and safety basis decisions, but there were many pathways available to achieve the overall goals. For example, different sets of SC and SS SSCs could be established, each of which satisfies the SSC classification criteria and F-C curves. What these analyses do not directly capture is the potential impact on factors such as plant layout, system cost, maintainability, etc. The input from the IDP members is necessary for the selection of the optimal strategy when accounting for all aspects, including safety and operation.

The IDP process was also critical for addressing uncertainty, which is especially challenging at the conceptual design phase. There is appreciable uncertainty in many areas of the VTR conceptual design PRA given the numerous design options under consideration and the maturity level of the SSC designs. Similarly, the safety analysis results, which contribute to the PRA in the determination of success criteria and event sequence outcome, also contain uncertainties due to preliminary modeling efforts and the associated assumptions. Therefore, the RIPB insights from the PRA must be assessed in the context of these uncertainties. While sensitivity studies and uncertainty analyses assist in this process, not all uncertainties can be quantitatively evaluated at the conceptual design stage. The IDP assessment was necessary to address these factors and to select system designs and SSC classification that account for the associated uncertainty. As outlined in Table 2-4, the IDP is only able to increase the classification of SSCs and may not dismiss RIPB insights to reduce SSC requirements.

#### 5.3.7 PRA Management and Tools

Section 5.1 highlighted some of the technical challenges that were encountered while developing the VTR conceptual design PRA and conducting the associated LMP analyses. First, current PRA tools are not designed to efficiently perform the sensitivity and importance analyses required for the LMP process. Therefore, the development of additional programs/scripts was necessary, which expanded the number of items under the quality assurance program given the importance of the results to the VTR safety basis. In addition, the number of tracked PRA models grew substantially as different design options were developed and associated analyses were performed. As the LMP analyses involved significant human effort, given the lack of purpose-built tools, the possibility for errors in both analysis and documentation increased.

The VTR project has the additional challenge of being a multi-organizational collaboration spanning the national laboratory system and industry. Substantial effort is necessary to complete the PRA and LMP analyses, with contributions for each organization. However, the traditional approach to conducting PRA assessments locally (in both organization and computing systems) hampers the ability to contribute and subsequently share results in an efficient fashion. This issue also has potential repercussions regarding data security as information is routinely transferred between organizations.

Lastly, it is important to establish a direct connection between the safety basis and design teams, as the LMP analyses contribute to the design requirements of SSCs. Therefore, the SSC designers must fully understand the requirements and their reasoning. As described in Section 5.3.5, this is partially a communication issue but also a technical management hurdle. An interfacing system that expedites the transfer of information between the teams (in a straightforward and intuitive fashion) can greatly reduce the potential for misinterpretation and the need for redesign.

Based on the experience from the conceptual design phase, the VTR PRA is adapting new management tools for subsequent development phases. This includes the use of a central repository, which tracks all details associated with model development, such as PRA models, PRA notebooks, success criteria analyses, source term analyses, and all associated documentation. The repository is accessible to all contributing organizations and has automated controls to satisfy necessary quality assurance procedures. In addition, it ensures a proper level of data security. The version control capabilities of the repository aid in the tracking of multiple analysis pathways, with separate development branches for specific authorization documents.

# 6.0 CONCLUSIONS

The VTR project is utilizing a RIPB approach for reactor design and authorization, which leverages risk insights to aid in key decision-making. The RIPB approach is based on recent progress by the NRC-endorsed LMP. In addition to informing design decisions, the RIPB process guides SBE identification and categorization, SSC classification, and the evaluation of the adequacy of DID to support VTR authorization by the DOE.

As part of initial reactor design efforts, a VTR conceptual design PRA was developed to support the RIPB process. The VTR conceptual design PRA focus was at-power internal events, with scoping analyses for seismic and sodium fire hazards. Both DOE and industry standards were utilized to guide its development, in addition to past SFR PRA experience. Following the RIPB process, a series of sensitivity and importance analyses were conducted to provide guidance to the VTR authorization safety basis. In parallel, multiple design studies were performed utilizing the PRA to compare alternative design options.

Preliminary results from the RIPB approach and the VTR conceptual design PRA were utilized to support the development of the VTR CSDR. The initial identification and categorization of SBEs, SSC classification, and DID evaluation were contained within the CSDR, which was submitted to DOE in 2019 as part of the CD-1 submittal package. Following review, DOE approved the CSDR in April 2020 and the CD-1 package in late 2020.

The experience gained through the application and implementation of the RIPB process in support of the VTR CSDR has proved valuable in several ways. Throughout the process, initial insights and lessons learned were published and presented at conferences to aid the advanced reactor industry [32-35]. In addition, the VTR safety basis team participated in a TICAP tabletop exercise [31] to provide insights to the TICAP that have been incorporated into their draft guidance [36]. Insights gained during preliminary seismic analyses have motivated subsequent DOE:NE activities regarding the regulatory treatment of low frequency external events, a topic that has also been highlighted by the advanced reactor industry. Most importantly, the submittal and approval of the VTR CSDR represent a major milestone in the use of RIPB methods in support of advanced reactor licensing.

# ACRONYMS, DEFINITIONS, AND SYMBOLS

## 6.1 Acronyms

| Acronym | Explanation |
|---|---|
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| AOO | Anticipated Operational Occurrence |
| ARCAP | Advanced Reactor Content of Application Project |
| ASME | American Society of Mechanical Engineers |
| BDBE | Beyond Design Basis Event |
| CCF | Common Cause Failure |
| CD | Critical Decision |
| CSDR | Conceptual Safety Design Report |
| DBE | Design Basis Event |
| DID | Defense-In-Depth |
| DPS | Diverse Protection System |
| EM | Electromagnetic |
| ES | Event Sequence |
| ESF | Event Sequence Family |
| FSAR | Final Safety Analysis Report |
| IDP | Integrated Decision Panel |
| IE | Initiating Event |
| IPT | Integrated Project Team |
| ISLOCAs | Interfacing Systems Loss of Coolant Accidents |
| JCNRM | Joint Committee on Nuclear Risk Management |
| LBE | Licensing Basis Event |
| LMP | Licensing Modernization Project |
| LOOP | Loss of Offsite Power |
| LWR | Light Water Reactor |
| MST | Mechanistic Source Term |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NS | Non-Safety |
| NSRST | Non-Safety Related with Special Treatment |
| NST | Non-Safety Related with No Special Treatment |
| PCT | Peak Clad Temperature |
| PRA | Probabilistic Risk Assessment |
| PSAR | Preliminary Safety Analysis Report |
| QHO | Quantitative Health Objective |
| RIPB | Risk-Informed Performance-Based |

| Acronym | Explanation |
|---|---|
| RPS | Reactor Protection System |
| SAHXs | Sodium-to-Air Heat Exchangers |
| SBE | Safety Basis Event |
| SC | Safety Class |
| SDIT | Safety Design Integration Team |
| SDS | Safety Design Strategy |
| SR | Safety Related |
| SRT | Argonne Simplified Radionuclide Transport Code |
| SS | Safety Significant |
| SSCs | Structures, Systems, and Components |
| TED | Total Effective Dose |
| TICAP | Technology Inclusive Content of Application Project |
| DOE | US Department of Energy |
| VTR | Versatile Test Reactor |

## 6.2 Definitions

| Term | Definition |
|---|---|
| <<Term>> | <<Definition>> |

## 6.3 Symbols

| Symbol | Definition |
|---|---|
| <<Symbol>> | <<Definition>> |

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