SAND2021-2668

SANDIA REPORT
SAND2021-XXXX
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Mechanistic Source Term Considerations for Advanced Non-LWRs, Revision 1

Nathan C. Andrews
Andrews J. Clark
Michael Higgins
Elliott Leonard
Jennifer E. Leute
David L. Luxat
Tina M. Nenoff

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico
87185 and Livermore,
California 94550

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ABSTRACT

This report is a functional review of the radionuclide containment strategies of fluoride-salt-cooled high temperature reactor (FHR), molten salt reactor (MSR) and high temperature gas reactor (HTGR) systems. This analysis serves as a starting point for further, more in-depth analyses geared towards identifying phenomenological gaps that still exist, hindering the creation of a mechanistic source term for these reactor types.

As background information to this review, an overview of how a mechanistic source term is created and used for consequence assessment necessary for licensing is provided. How a mechanistic source term is used within the Licensing Modernization Project (LMP) is also provided. Lastly, the characteristics of non-LWR mechanistic source terms are examined.

This report does not assess the viability of any software system for use with advanced reactor designs, but instead covers system function requirements. Future work within the Nuclear Energy Advanced Modeling and Simulations (NEAMS) program will address such gaps.

This document is an update of SAND 2020-6730. An additional chapter is included as well as edits to original content.

ACKNOWLEDGEMENTS

This effort was completed in collaboration with Argonne National Lab. The authors would like to thank David Grabaskas, Shayan Shahbazi and Bo Feng for their involvement with this effort.

Much thanks are also provided to the nuclear energy community at Sandia National Laboratory for their review of this document, particularly Brad Beeny and Mariah Smith.

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ACRONYMS AND DEFINITIONS

Abbreviation	Definition
AOO	Anticipated Operational Occurrence
BDBE	Beyond Design Basis Event
DBE	Design Basis Event
DOE	Department of Energy
ELAP	Extended Loss of AC Power
FHR	Fluoride-Salt-Cooled High Temperature Reactor
HTGR	High Temperature Gas Reactor
ISLOCA	Interfacing System Loss of Coolant Accident
LMP	Licensing Modernization Project
LTSBO	Long Term Station Blackout
LWR	Light Water Reactor
MSBR	Molten Salt Breeder Reactor
MST	Mechanistic Source Term
MSR	Molten Salt Reactor
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PB-FHR	Pebble Bed Fluoride-Salt-Cooled High Temperature Reactor
PRA	Probabilistic Risk Assessment
RPV	Reactor Pressure Vessel
SOARCA	State-of-the-Art Reactor Consequence Analysis
STCP	Source Term Code Package
STSBO	Short Term Station Blackout
TI-RIPB	Technology-Inclusive, Risk-Informed, Performance-Based
UHS	Ultimate Heat Sink

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1. INTRODUCTION

1.1. Background

This document addresses considerations that must be accounted for in the development of a mechanistic source term to the environment. As understanding of radionuclide release scenarios has evolved over the past several decades, the Nuclear Regulatory Commission (NRC) has enabled a transition from prescribed core melt releases to accident source terms based on a more realistic representation of the consequences of a reactor accident for Light Water Reactors (LWRs). A realistic source term, based on the unique physics and characteristics of the system under accident conditions, is a generally accepted approach to support a risk-informed licensing strategy for specific advanced reactor designs. Such a source term is referred to as a "mechanistic" source term (MST). The concept of an MST was first introduced in SECY-93-092 [1] and was intended to provide the basis for evaluating a representative source term for a broader spectrum of reactor designs, though it has subsequently been applied to development of representative source terms for operating LWRs. An MST accounts for radiological release specific to evaluated accident sequences considering best-estimate phenomenological models and mitigating measures.

TID-14844 served as the original licensing basis for operating LWRs, which was published in 1962 by the Atomic Energy Commission. It requires the assumption of a "substantial meltdown of the core" resulting in the release of 100% of noble gasses, 50% of halogens and 1% of solids to the containment. This is roughly 15% of all activity present in the core. Upon reaching the containment, some attenuation of the activity is allowed based on available systems. This source term is considered to be bounding and non-mechanistic. It is not based on realistic plant response under accident conditions and furthermore does not reflect current experimental evidence. This source term is also not applicable to non-LWRs, as it was developed specifically for LWRs. [2]

The first mechanistic source term developed for regulatory purposes is NUREG-1465, which took into account both experimental insights and knowledge gained from the simulation of accident scenarios with dedicated software. [3] The simulations that were used to support the development of this mechanistic source term were provided by the Source Term Code Package (STCP); a forerunner to modern, integrated severe accident and source term modeling software such as MELCOR. [4] [5] STCP characterized reactor accident phenomena using a series of stand-alone, weakly-coupled codes. The calculation sequence is not able to account for interactions or feedback effects between the phenomena modeled by distinct codes. The NRC periodically updates the *revised* source terms in NUREG-1465 based on new experimental information, changes in the system itself (e.g. high burnup fuel or accident tolerant fuel) and enhancements to system modeling.

With the development of NUREG-1465, the NRC evaluated the continued applicability of TID-14844 to operating reactors. TID-14844 was found to remain appropriate for protecting public health and safety such that operating reactors would not be required to reassess accidents using the revised source terms developed in NUREG-1465. Under some circumstances, however, it was recognized that licensees may identify cost-beneficial licensing actions using the revised, or alternative source terms (ASTs) developed in NUREG-1465. Regulatory Guide 1.183 (RG1.183) which provides guidance for licensees was developed to provide guidance in support of a licensee using an AST in design basis analyses. [6]

The licensing framework for LWRs within the United States has been clearly established; it is under active development for non-LWR advanced reactors. The Licensing Modernization Project (LMP), coordinated by the Nuclear Energy Institute (NEI), is a guideline for a technology-inclusive, risk-informed, performance-based approach for the selection of event scenarios, safety systems and risk-informed treatment of the system. [7] This approach has since gained approval from the NRC in SECY-19-0177. [8] Additional effort is being coordinated by NEI to further establish the licensing framework begun under the LMP. The Technology Inclusive Content of Application Project (TICAP) represents a follow-on component of the non-LWR licensing framework development. It is focused on establishing the guidance for developing content for specific portions of the NRC license application Safety Analysis Report (SAR) for non-LWR designs.

In addition to the non-LWR licensing framework, the supporting technical bases, methods, and tools to develop a license application SAR all represent areas requiring significant development. This is due to the impact of the significantly different physics and chemistry relevant to non-LWRs compared to operating LWRs, and the resulting impact on characterization of radiological source terms necessary to evaluate achievement of NRC safety goals. The NRC has undertaken an initiative under Integrated Action Plan (IAP) Strategy 2 [9] to develop the necessary capabilities for performance-based, risk-informed characterization of non-LWR systems. The NRC's strategy for adopting modeling and simulation tools to perform independent safety assessments as part of non-LWR licensing is presented in the first five volumes of the "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy" [10], [11], [12], [13], [14].

1.2. Regulatory Studies of Note

The development of regulatory guidance on the preparation and use of mechanistic source terms has generally indicated that the overall estimation of nuclear power risk is lower than the original bounding source terms prescribed by TID-14844. Subsequent regulatory studies have developed mechanistic source terms to support a range of evaluations, rulemakings, and other decision-making processes. These studies have followed a broader adoption of risk-informed decision-making by the NRC. This range of studies include probabilistic risk assessments (PRAs) and uncertainty analyses, which serve as guidelines for the development of a mechanistic source term.

Key studies that can be used to inform the application of PRA and uncertainty techniques to advanced reactors include:

- The State-of-the-Art Reactor Consequence Analysis Uncertainty Analyses of Surry, Sequoyah and Peach Bottom [15] [16] [17] provides an example of a probabilistic consequence assessment and uncertainty analysis for three reactor sites and five severe accident scenarios using MELCOR/MACCS. Best practices developed from this study are highlighted in [Add NUREG/CR-7009] to provide guidance on modeling approaches and input parameter selections, which could be applied to PRA applications.
- Vogtle full-scope site Level 3 PRA [18] performed by the NRC evaluates risk from sources of radioactivity beyond the reactor using the SCALE/MELCOR/MACCS code packages, with analysis areas including the spent fuel pool, dry cask storage and integrated site
- In support of this post-Fukushima evaluation of U.S. nuclear power plant safety, the NRC conducted SCALE/MELCOR/MACCS assessments of progression and consequences from loss-of-cooling and loss-of-coolant accidents in an on-site spent fuel pool in NUREG-2161 [19]

- The Performance Based Analysis for BWR Containment Venting under the Containment Protection and Release Reduction (CPRR) rulemaking [20] provides an example on how to use PRA methodology to inform cost/benefit analyses in design and decision making.

1.3. Report Scope

Within this report, a functional review of the radionuclide containment strategies of fluoride-salt-cooled high temperature reactors (FHR), molten salt reactors (MSR) and high temperature gas reactors (HTGR) is performed. This assessment serves as a starting point for further assessments to refine understanding of phenomenological gaps that still exist, and if not adequately addressed, could reduce the quality of a mechanistic source term for the various reactor types and also prevent useful risk-insights to be identified. This report does not examine the mechanistic source term of sodium fast reactors, due to past work in this area [21] [22].

As background information to this review, an overview of how a mechanistic source term is created and used for consequence assessment necessary for licensing is provided. Use of mechanistic source terms within the LMP framework is discussed. Lastly the characteristics of non-LWR mechanistic source terms are examined.

2. MODELING A MECHANISTIC SOURCE TERM AND ASSOCIATED CONSEQUENCE

In order to accurately and realistically capture the type and amount of radionuclides that could be released to the environment during an accident scenario, it is important to develop an appropriate representation of the system and radionuclide inventory. The transport of radionuclides in the system must then be modeled, respecting realistic physical behavior as they move from their initial location (in a fuel matrix or circulating fuel-salt) to the environment. After they are released to the environment, their transport, dispersion, and subsequent deposition must be characterized. When assessing the consequences of radionuclide release two main metrics are used to determine event severity: economic loss and health effects. Within the US, health effects are the basis for licensing a reactor; however, other countries take both into account when making licensing determinations.

2.1. System Description

The starting point for any source term calculation is a realistic representation of the system. The State-of-the-Art Reactor Consequence Analyses (SOARCA) of Peach Bottom [15], Surry [16] and Sequoyah [17] demonstrated that developing and exercising appropriate fidelity representation of systems allows users to capture the full range of realistic behavior during radionuclide release scenarios. These studies captured integral plant accident progression and radionuclide transport within the core region, primary system, secondary system, containment, and reactor/auxiliary enclosures. Accounting for all systems and regions of the plant enables accurate modeling of radionuclide transport within the reactor and plant to determine the ultimate radiological release to the environment.

2.2. Radionuclide Inventory

The total mass of all radionuclides within the system must be calculated and used as an input to the system model. Such calculations are performed with neutronic analysis software such as the SCALE code system. [26] Traditionally, radionuclide inventories are determined for different points in a reactor's operating cycle: beginning of cycle, middle of cycle and end of cycle. These different radionuclide inventories will result in different releases to the environment given the same accident scenario, with end of cycle radionuclide inventories being much higher due to increased irradiation time.

In SFRs and HTGRs with prismatic fuel, representing the location of the radionuclide inventory within a system model is straightforward, because the core is solid and stationary. For MSRs, the calculation of the initial radionuclide inventory has necessitated the development of neutronic analysis software that takes into account the chemical interactions that occur. In MSRs direct fission products can readily be found throughout the entire primary system, as opposed to only neutron activation products and leaked radionuclides from fuel. Pebble bed HTGRs and FHRs also present a unique problem since the pebbles in the core can move individually. Tracking the transport of radionuclides in the core is complicated due to this pebble motion. Understanding the transit history of a pebble through the reactor core is of importance to determining both the fission power and temperature of the pebble. Fission power and temperature have a direct impact on the neutronic behavior of the system and resulting fission product inventory in the pebbles (or potentially released into the reactor primary system).

Characterization of radiological risk also requires the identification of other sources of fission product inventory outside the reactor. For currently operating reactors, this includes the spent fuel pool as well as any fuel stored that has entered on-site dry cask storage. Previous studies have been performed to evaluate risk arising from spent fuel pool accidents. A number of safety issues have arisen in the past, with resulting decision-making required related to minimizing risk associated with high density storage of spent fuel in pools.

In particular, the NRC's post-Fukushima evaluation of consequences from beyond design basis spent fuel pool accidents [27] further supports the conclusion that high density storage of spent fuel in pools reduces risk to the public. This study relied on modeling and simulation, specifically the MELCOR accident progression/source term estimation and MACCS off-site consequence evaluation code systems, to establish the necessary technical bases to support risk-informed decision-making related to the practice of high density spent fuel storage in pools.

Both MELCOR and MACCS have capabilities that promote application to informing risk studies for accidents involving out-of-reactor radiological sources. This has enabled both code systems to be applied to a range of facility safety studies (e.g., DOE nuclear facility safety studies). The code systems represent fundamental analytical capabilities for the NRC in establishing regulatory readiness for regulation of emerging non-LWR fuel cycles [14].

The capabilities necessary to perform safety assessment of non-LWR fuel cycle activities also apply to a range of non-reactor radiological release accidents that can arise in MSR and FHR systems. These concepts contain radionuclides in systems attached to the primary circuit, in particular both reactor types have off-gas systems that remove gases from the circulating salt in the primary circuit. These gases contain radioactive nuclides. MSR systems may contain on-line processing features, which would be an additional concentration of radionuclides away from the core.

2.3. Fuel Modeling

The SCALE code system represents one source of capability for developing radionuclide inventories that accumulate in the fuel during operation. This code system is necessary for the above step in the source term development process. Under accident conditions, the integrity of the fuel matrix and cladding will be challenged by a range of thermal, mechanical, and chemical loads unique to the accident scenarios being considered.

For LWRs, the NRC utilizes the Fuel Analysis under Steady-State and Transients (FAST) code system to calculate the thermal-mechanical response of nuclear fuel under steady-state, anticipated operational occurrences (AOOs), and design basis accidents (DBAs). [28] FAST is used to determined parameters characterizing fuel performance such as temperature, stress/strain, fission gas release, and oxidation. It is used to support a number of regulatory evaluations such as

- Assessment of specified acceptable fuel design limits (SAFDLs) for the UO_2/Zr fuel system specified in NUREG-0800, Chapter 4 [29]
- Assessment of fuel vendor codes and methods
- Determination of initial conditions for DBA analyses
- Perform spent fuel analyses under drying conditions to determine relevant performance metrics such as hoop stress and creep

The NRC assessment of fuel performance for non-LWRs is following a strategy adapting FAST, with a complementary use of the BISON code. [11]

In assessing accident progression and determining radiological source terms, codes such as FAST are typically not directly used by integral plant response codes such as MELCOR. The MELCOR code system is informed by the mechanistic fuel performance modeling in a code system such as FAST, which is typically used to specify best estimate values for fuel failure specified in terms of surrogate conditions such as fuel temperature. This obviated the need for detailed calculations of quantities such as stress/strain in integral plant response assessment codes. Since initial fuel failure typically occurs very early in the events considered by integral plant response codes, the reduction in fidelity associated with this approach typically has a limited impact on the overall evaluation of consequences. Furthermore, this simplified approach also provides a reasonable first assessment of consequences from events terminated with more limited fuel damage and radiological release. Where such events arrested with more limited fuel damage exhibit more substantial releases to challenge one or more safety limits, more detailed assessment tools such as FAST or BISON can be applied to establish a more refined estimate of safety margin for events considered important to risk.

This general approach is fundamental to developing or reviewing a risk-informed safety case for a non-LWR. Codes such as FAST or BISON have or will be incorporating models relevant to refined assessment of safety margins for higher frequency, but lower consequence events categorized as either AOOs or design basis events (DBEs). Consequences from the spectrum of scenarios considered in a risk assessment will, as is the current approach for LWRs, be estimated with code systems such as SCALE, MELCOR and MACCS.

While these integral analyses can provide a lower fidelity estimate of radiological consequence for specific accident scenarios, this estimate is typically appropriate for the broad spectrum of accident scenarios. In situations where derived performance criteria require higher fidelity characterization to ensure adequate safety margin in a design, analysis tools considered in Volume 1 or Volume 2 of the NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy [10] [11] should be applied. This approach does not deviate from the typical approach followed for LWRs. For example, large break loss of coolant accidents (LBLOCAs) are typically assessed using dedicated, higher fidelity methods for specific phenomena, such as during core reflooding, that is appropriate to characterizing safety margin. Such methods are specific to this scenario, while less relevant to the range of other event scenarios defining the overall risk profile for the design.

For systems such as MSRs, where fuel is dissolved into the circulating working fluid transporting heat to a unit harvesting fission energy, the fuel matrix and cladding are not present to perform the traditional function separating radioactive material from the primary heat transport fluid. In these molten fuel systems, retention of fission products in the circulating fluid depends to a large extent on its solubility within the fluid. This requires more detailed evaluation considering, for example, the thermochemical equilibrium state of the fluid mixture as well as processes such as aerosol generation through bubble transport and release.

2.4. Event Selection

In order to develop a source term, appropriate event scenarios need to be selected. These scenarios can either be internally initiated (e.g., a random failure of an SSC) or externally initiated (e.g., a seismic event). For LWRs, a licensing basis of a postulated full core melt following a LBLOCA is used for the regulatory source term. However, additional event scenarios have been rigorously evaluated to ensure that source terms from other potential accident scenarios are evaluated. These include: short-term and long-term station blackout (STSBO, LTSBO), small break loss of coolant accident (SBLOCA), interfacing system loss of coolant accident (ISLOCA), and extended loss of AC power (ELAP). For a non-LWR, the most challenging events would be determined through the application of an LMP analysis. The LMP analysis provides a guideline for a technology-inclusive, risk-informed, performance-based (TI-RIPB) process for selection of licensing basis events (LBEs). This PRA based process represents a significant departure from the prescriptive source term specifications of TID-14844 where bounding source terms used for licensing basis, and looks at the reactor design from a holistic risk-based perspective. Section 3 of this report provides more detail on the LMP.

2.5. System Behavior and Radionuclide Transport

Within a reactor system it is necessary to capture the behavior of functional sub-systems and their generally coupled interactions. Reactor plant systems and their responses to off-normal or accident conditions that need to be captured include but are not limited to:

- Thermal-hydraulic response of the primary reactor heat transport system, the reactor cavity, the containment, and the confinement buildings
- Core uncovering (loss of coolant), fuel heat up, chemical reactions, fuel degradation (loss of rod geometry), and core material melting and relocation
- Heat up of reactor vessel lower head from relocated fuel materials, thermal and mechanical loading, and failure of the vessel lower head, in addition to transfer of core materials to the reactor vessel cavity
- Core-concrete attack and ensuing aerosol generation
- In-vessel and ex-vessel hydrogen production, transport, and combustion
- Fission product release (aerosol and vapor), transport, and deposition
- Behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling
- Impact of engineered safety features on thermal-hydraulic and radionuclide behavior

The phenomena that must be captured in the development of a mechanistic source term are dependent upon the system itself. For an LWR system, a representation of the pertinent phenomena and processes can be seen in Figure 2-1. It is important to note that not all phenomena and processes are considered in every integral plant response simulation performed as part of a risk assessment. This range of phenomena and processes, however, are those that must typically be captured in accident evaluations for a PRA to ensure that the entire risk profile (i.e., the set of accident scenarios determined by the PRA) is captured.

2.6. Modeling of Event Consequence

Following the creation of a mechanistic source term, the consequence of the source term needs to be assessed, both on-site and off-site. Within the United States, this means assessing the dose to workers and the public and associated health consequences. Specifically, to assess achievement of the derived quantitative NRC safety goals [35], it is necessary to evaluate the potential for early fatalities and latent cancer fatalities as a result of an accidental radiological release.

The NEI 18-04 Licensing Modernization Project (LMP) [7] has introduced a consequence-frequency map to evaluate achievement of safety goals across the entire risk profile. The licensing framework developed under the LMP is discussed in further detail in Section 3. The goal of the LMP is to develop a technology-inclusive risk-informed, performance-based licensing framework. As a result, assessment of consequences is a fundamental component in the process established under the LMP.

It is necessary to perform calculations of the behavior of radionuclides on-site and off-site. Modeling of local weather conditions and site characteristics is normally considered in estimating transport and dispersion of radionuclides released on- and off-site.

Important physics for consequence assessment includes, but is not limited to:

- Atmospheric transport and dispersion
- Wet and dry deposition
- Meteorology
- Exposure pathways
- Dose to workers and the public
- Health effects

Beyond the evaluation of achievement of safety goals, radionuclide transport in the environment, and estimation of dose to affected individuals and population, is necessary to support emergency response activities. Models to determine dose to individuals and the population affected by radioactive atmospheric releases from a power plant support decision-making by governments and other local authorities in the event of a radiological release event at a nuclear power plant. Specifically, governments and local authorities utilize these evaluations to guide the emergency response, in particular the planning and implementation of different protective actions necessary to ensure public safety. These types of evaluations also enable a regulatory agency like the NRC to provide guidance to governments and local authorities during such an event.

The Emergency Planning Zone (EPZ) surrounding a nuclear power plant represents the area in which emergency response procedures are preplanned in the event of release scenario, and is typically set at 10 miles within the United States. However, the frequencies and magnitudes of release are expected to be much lower for non-LWRs as compared to their LWR predecessors, causing many reactor developers to explore opportunities to significantly reduce the size of the EPZ from the standard 10 miles. The potential to significantly reduce the EPZ, possibly even to the exclusion area boundary (EAB), has significant benefit to the operational economics of a nuclear power station. Performing a risk-informed analysis to size the EPZ is generally considered essential for non-LWR designs to achieve economic viability.

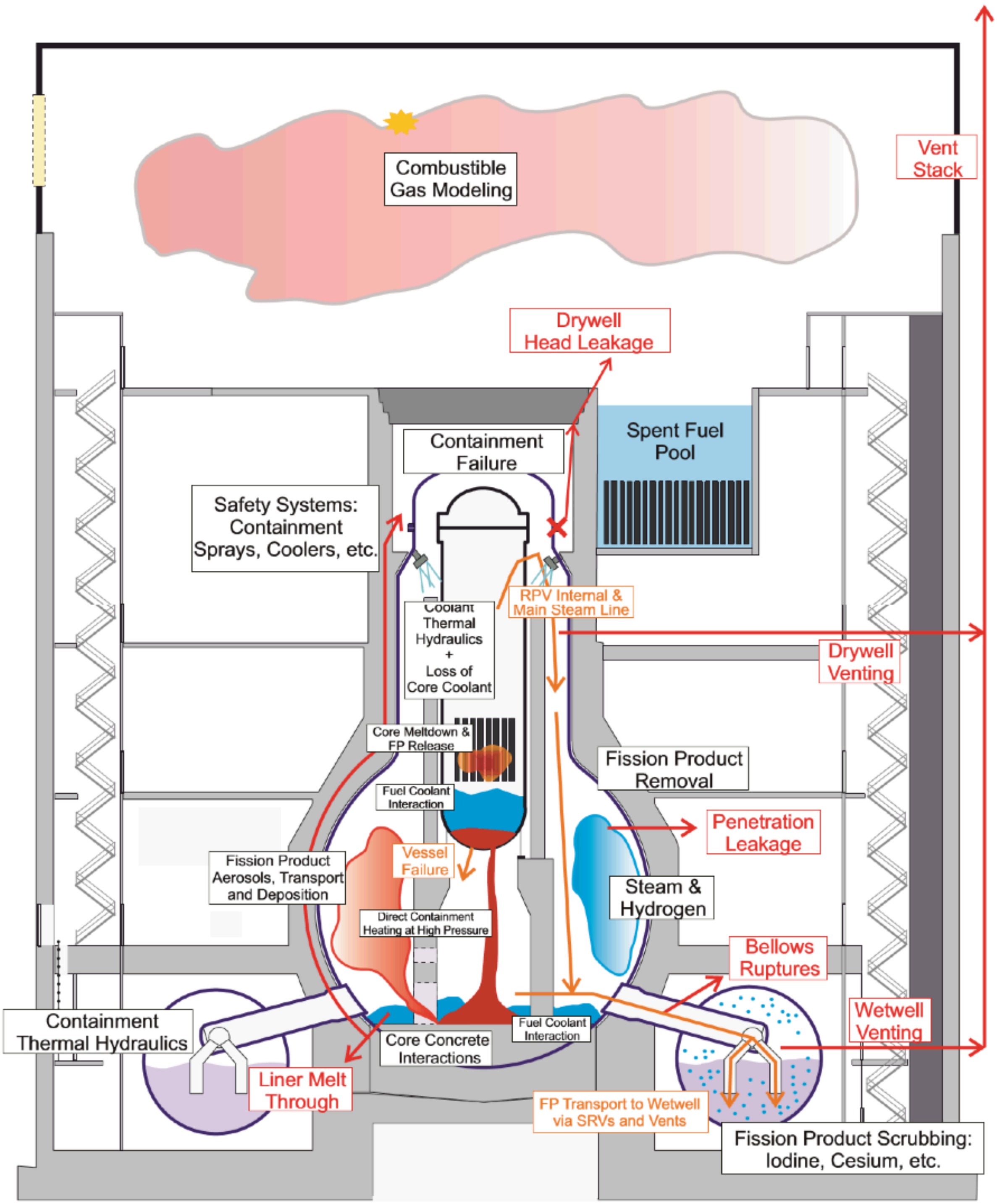

Figure 2-1. Physics addressed within the MELCOR code for an LWR system [36]

2.7. Modeling of Mechanistic Source Term within the NRC Framework

The United States Nuclear Regulatory Commission (NRC) develops estimations of accidental radiological releases (i.e., the radionuclide source term) in the manner shown in Figure 2-2. This approach was originally developed for LWRs, but is broadly applicable to non-LWRs. The primary evolution from the typical approach for LWRs is a broader characterization of an overall nuclear reactor's risk profile. For LWRs, a prescribed design basis source term has been utilized to assess overall performance of a nuclear power plant in achieving safety goals. This source was identified to achieve a reasonably bounding estimate of the range of source terms that could be realized across the spectrum of radiological release scenarios. For non-LWRs utilizing the LMP, a prescribed design basis source term is not required. Rather the performance of the system at preventing or mitigating radiological release to the environment is evaluated across a design's entire risk profile.

Independent of the range of accident radiological release scenarios for which source terms are to be evaluated, the process of performing the analytical evaluations is common. The SCALE code is used to develop radionuclide inventories with sensitivity analysis and uncertainty quantification. These inventories are then used as an input to the severe accident analysis software MELCOR, which models reactor behavior during normal, off-normal and accident conditions. MELCOR computes radionuclide release to the site and the environment as a radionuclide source term.

The consequence of the radionuclide source term is then assessed. This is done through calculations with the RADionuclide Transport, Removal, And Dose (RADTRAD) code for calculations on-site, such as control room dose calculations. [37] RADTRAD uses a combination of (1) realistic system behavior and (2) atmospheric dispersion characteristics to model radionuclides as they move from the primary containment to elsewhere on-site.

For calculations of consequence off-site, such as dose to the public resultant from an event, the MACCS software is primarily used. It accounts for atmospheric transport, dispersion, and deposition of radionuclides. From there it allows the assessment of both health and economic consequences. [38]

The RASCAL software is a response tool used to make recommendations regarding emergency response decisions. During a radiological release event, it is used by the NRC, other federal responders, state governments and others local responders to provide guidance on evacuation decisions and other protective actions. [39]

The combination of these code systems provides regulators with the necessary information to make decisions regarding safety reviews and environmental impact reviews, which are necessary for the licensing of reactors within the US.

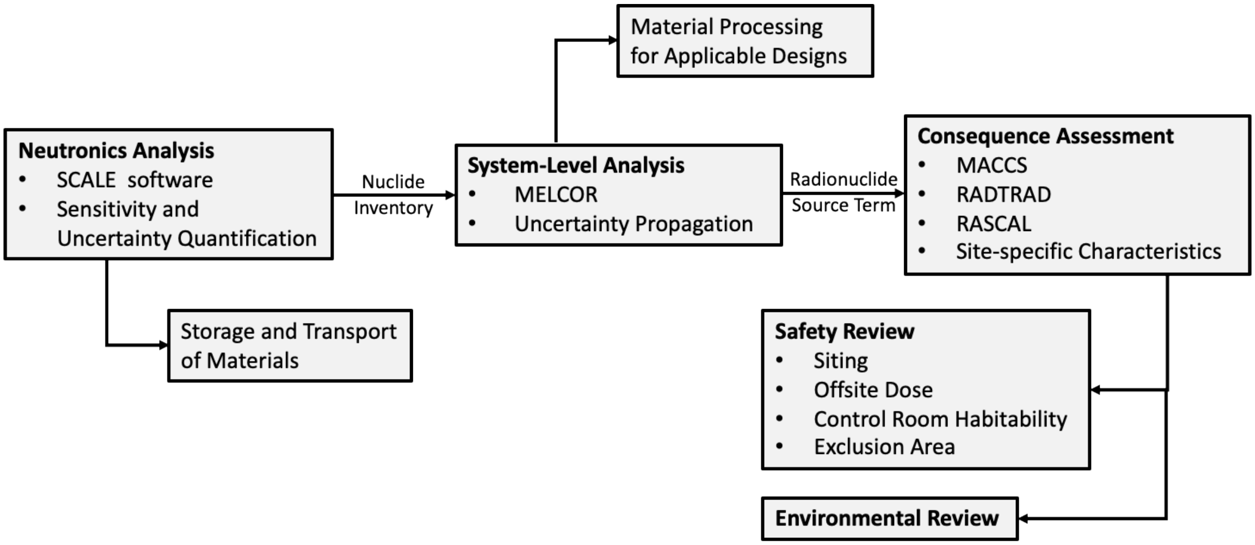

Figure 2-2. Method for the development and use of a mechanistic source term within the NRC framework [36]

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3. RISK-INFORMED PERFORMANCE-BASED TECHNOLOGY INCLUSIVE GUIDANCE FOR NON-LWR LICENSING BASIS DEVELOPMENT

NEI 18-04 provides a guideline for a technology-inclusive, risk-informed, performance-based (TI-RIPB) process for selection of licensing basis events (LBEs), safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-LWRs. [7] NEI 18-04, also referred to as the Licensing Modernization Project (LMP), has gained considerable attention across the industry and received approval by the NRC in SECY-19-0117. [7] [8]

Figure 3-1 provides a high-level depiction of the LMP process from the DOE website:

Figure 3-1. Licensing Modernization Project

The LMP process allows advanced reactor designers to develop a realistic list of events based on their particular reactor design, rather than following prescriptive source terms developed for LWRs, as was the case for previous licensing processes. These events include:

Anticipated Operational Occurrences (AOOs): Anticipated event sequences expected to occur with a frequency of $1x10^{-2}$/plant-year or greater.

Design Basis Events (DBEs): Unexpected event sequences with a frequency of $1x10^{-4}$ to $1x10^{-2}$/plant-year.

Beyond Design Basis Events (BDBEs): Highly unexpected event sequences with a frequency of $5x10^{-7}$ to $1x10^{-4}$/plant-year.

Design Basis Accidents (DBASs): Derived from DBEs in order to set the design criteria and performance objectives for the safety SSCs. DBAs differ from DBEs in that DBAs assume that only safety related SSCs are available to mitigate the event sequence and ensure the dose requirements of 10 CFR 50.34 are met.

Licensing Basis Events (LBEs): Represents the entire collection of event sequences, to include AOOs, DBEs, BDBEs, and DBAs which are considered for the design and licensing basis.

3.1. Selection of Licensing Basis Events

The process to select LBEs is depicted in Figure 3-2, however it is important to note that the steps listed in Figure 3-2 can be conducted in any order and their completion is considered an iterative process [7].

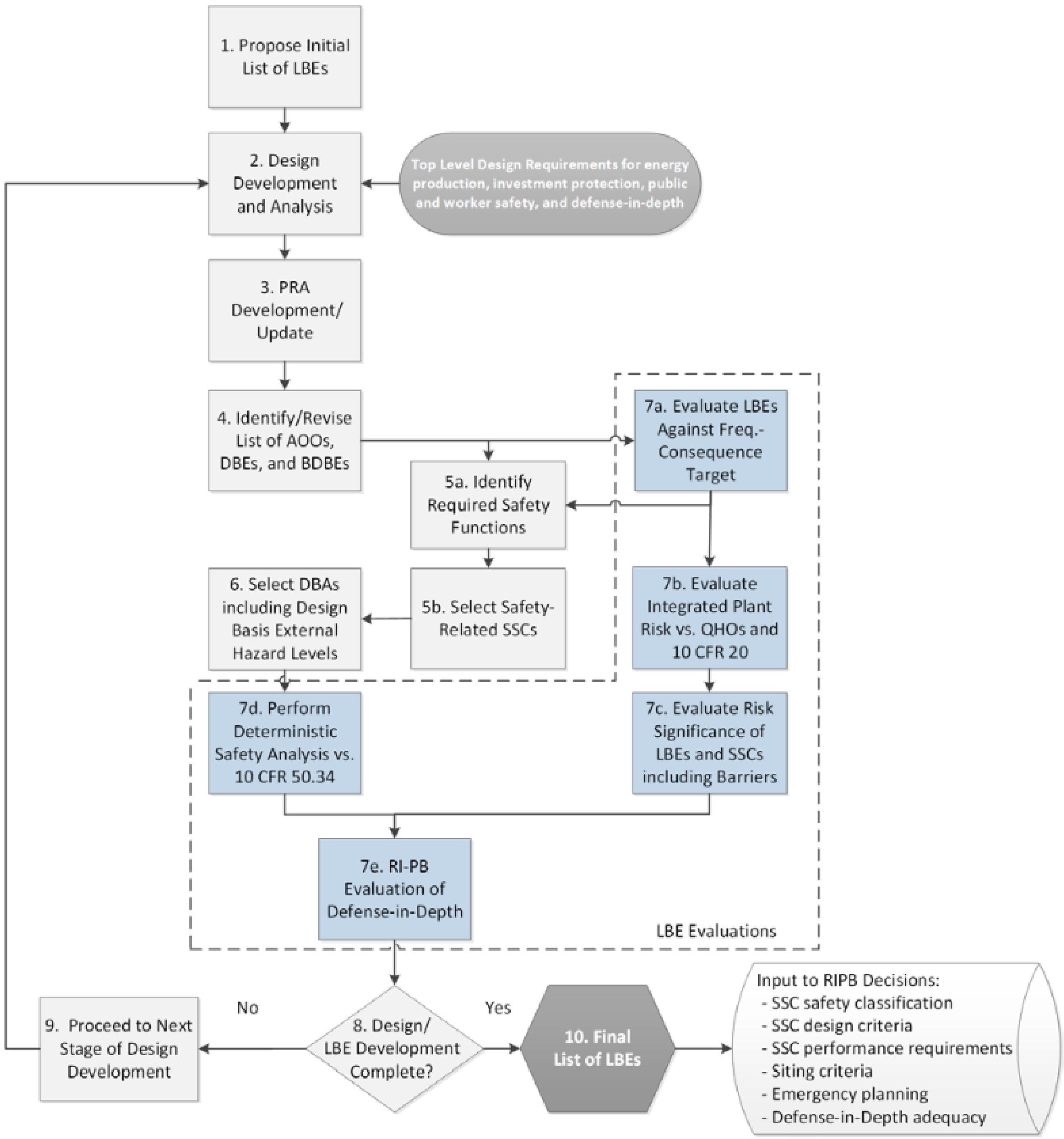

Figure 3-2. LBE Selection Process [7]

The first four steps of the LBE Selection Process are of greatest importance to source term calculations. LBE Step 1, Propose Initial List of LBEs, selects the initial list of LBEs which is comprised of an initiating event, plant response to the initiating event (i.e., sequence of successes and failures of mitigating systems), and a well-defined end state.

LBE Step 2, Design Development and Analysis, includes the definition of safety elements within the system. One approach involves the development of safety function trees. The highest level of the tree identifies the safety objective, which at the highest level and most severe consequences would be containing radionuclides. Flowing downwards from this top level would be the safety functions and systems designed into the plant that are intended to achieve this objective.

LBE Step 3, Probabilistic Risk Assessment (PRA) Development/Update, models the response of each SSC in the plant that performs a function to prevent or mitigate a release of radioactive material. There are many technical elements that make up a PRA model, and some of these technical elements will be absent in the initial design. As the design matures, these technical elements will mature as well. For reactors early in the design phase, source term calculations are desirable. There is a minimum amount of analysis that is needed in order to perform a source term calculation. Some key information that would be required is: [7]

- initiating events,
- event sequence analysis (also referred to as the accident sequence analysis),
- systems analysis (i.e., the causes of failure for each plant system),
- human reliability analysis (if applicable).

Note that there are several elements that factor into the PRA analysis, such as success criteria, data analysis, and prediction of end states. All of the information gathered and evaluated in the PRA is also crucial for source term evaluations. Reactor designs are intending to use LMP and PRA in the design stages, which may also include a source term evaluation. Thus, it is important that as the design and PRA models mature, the source term evaluation models should also mature.

LBE Step 4, Identify/Revise List of AOOs, DBEs, and BDBEs, takes the event sequences being modeled and evaluated in the PRA and groups them into event sequence families, where each event sequence family has a similar initiating event, challenge to plant safety functions, plant response, and end state. [7]

Section 4 of this document describes how a mechanistic source term would be determined for reactors early in the design stage. No attempt is made to calculate source terms for these reactors due to the lack of design detail and computer codes that can evaluate these reactors. [7]

3.1.1. Frequency-Consequence Targets

One of the main outcomes of the LMP process is to plot event sequences (or event sequence families) on a frequency-consequence curve, as shown in Figure 3-3. In this way, reactor designers can evaluate plant risk against frequency-consequence targets designed to assess achievement of safety goals. The y-axis of Figure 3-3 is the frequency of the event sequence. Event sequence frequencies are generally quantified using PRA methods such as event tree and fault tree models. System, structure, and component reliabilities may be used, along with uncertainties, in a probabilistic model that generates event sequence frequencies. PRA methodologies have matured over decades and there is substantial documentation in the open literature to provide a comprehensive calculation of event sequence frequencies. One area, however, that requires additional methodological development is in the area of quantifying passive system reliability across the entire risk profile or a power plant design.

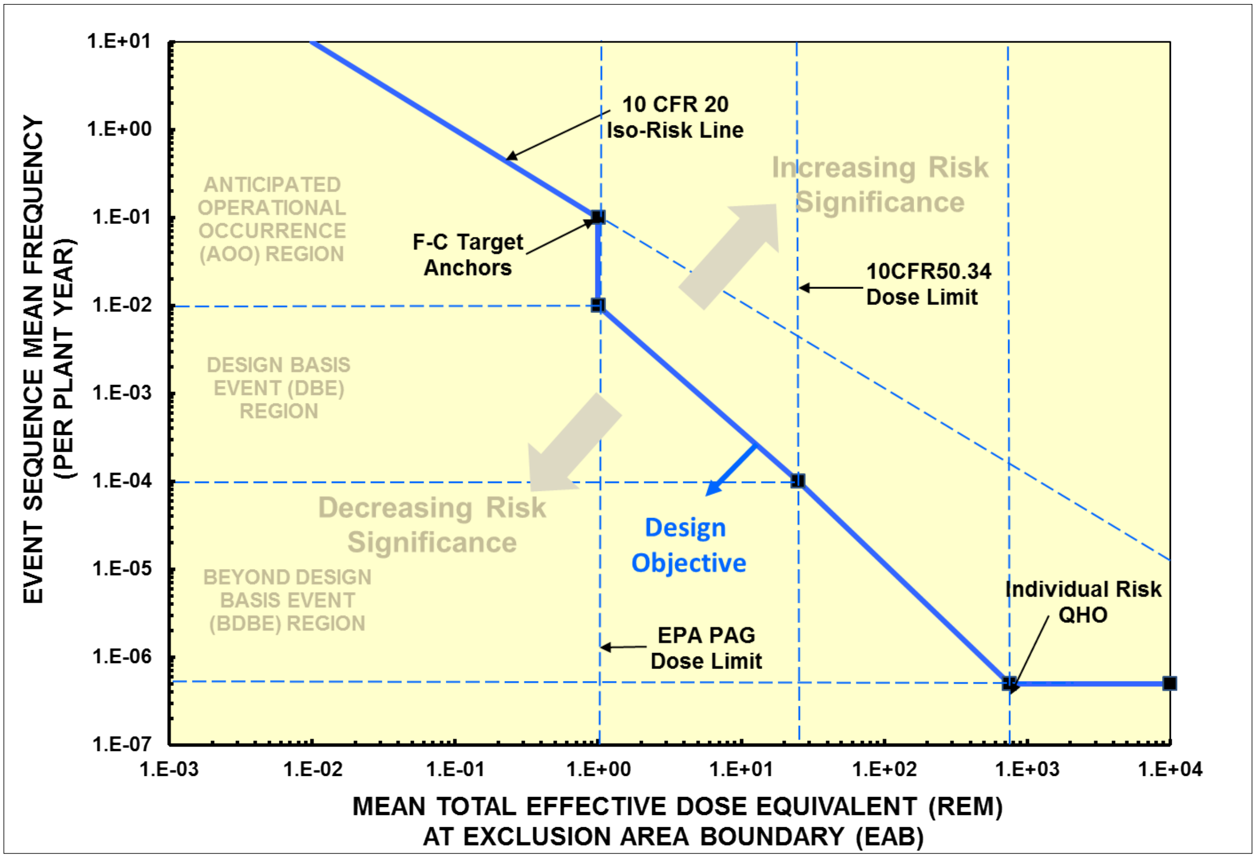

Figure 3-3. LMP frequency-consequence curve to assess event significance and classification [7]

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

After selecting and categorizing LBEs, safety classifications and performance criteria are assigned to reactor structures, systems, and components (SSCs). These SSCs ensure the F-C targets established in Figure 3-3 are met and that any potential releases are within established limits. SSCs are categorized as safety-related (SR), non-safety related with special treatment (NSRST), or non-safety related with no special treatment (NST) depending on their importance to meeting F-C targets in DBAs, DBEs, and BDBEs, or for meeting DID adequacy. The Licensing Modernization Project defines the SSC categories by the following definitions:

Safety-Related: defined as all SSCs that mitigate the consequences of DBEs to be within the LBE F-C Target, and to mitigate DBAs to meet the dose limits of 10 CFR 50.34. Safety related SSCs are also defined as SSCs that prevent the frequency of BDBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target. [7]

Non-Safety-Related with Special Treatment (NSRST): defined as those SSCs that prevent or mitigate any LBE from exceeding the F-C Target, make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs, or required for special treatment in determining DID adequacy. [7]

Non-Safety-Related with No Special Treatment (NST): defined as all other SSCs. [7]

3.3. Evaluation of Defense-In-Depth Adequacy

The concept of defense-in-depth (DID) ensures there are several layers of protection in place to prevent and mitigate potential releases of radiation to the environment. The LMP provides an approach to assessing DID for reactor design, construction, maintenance, and operation. This involves the complementary application of multiple overlapping design features, operating procedures, emergency procedures, and programmatic processes. Ultimately the adequacy of defense in depth is assessed through a risk-informed, integrated decision making process. The framework for evaluating DID through this risk-informed process is depicted in Figure 3-3 below.

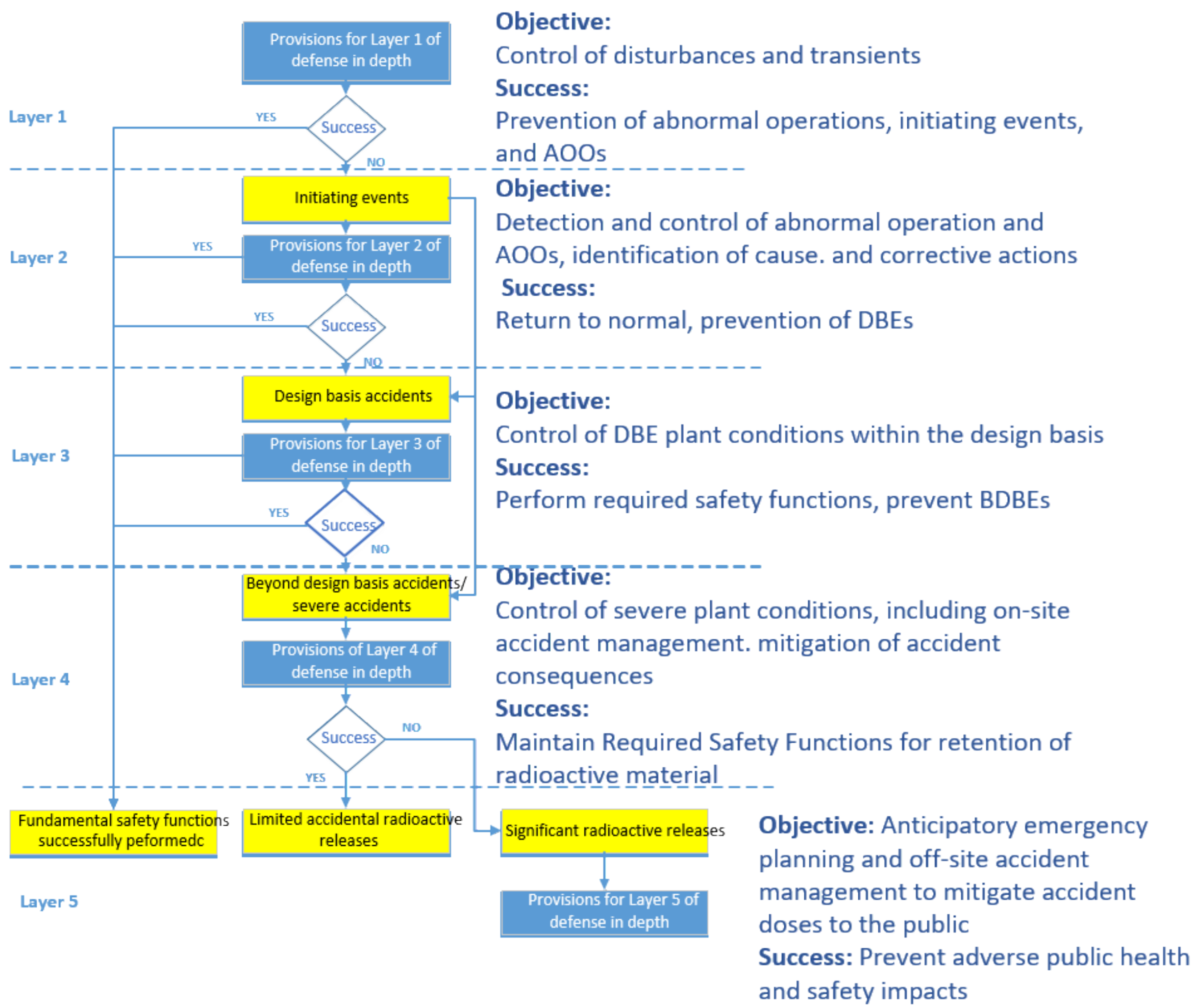

Figure 3-4. LMP Evaluation of Defense-In-Depth

3.4. Role of Source Term Calculations

Source term calculations model the phenomenological processes and radionuclide transport from the event sequences described above. Source term codes model the evolution of an accident transient defined by an event sequence. This is necessary to determine how and when radionuclide release barriers are compromised, and the subsequent transport of radionuclides past one or more of the compromised barriers. Characterizing the release and transport of radionuclides within a plant under accident conditions is complex because of the number of interacting processes and phenomena.

Assessment of LWR source term modeling has generally indicated that the dominant uncertainties affecting quantitative accident progression and source term estimates arise in the interaction of this range of processes and phenomena at reactor scale [40]. Effectively representing these interactions is thus a fundamental necessity for any modern source term code. Systems analysis codes capable of characterizing integral plant response (i.e., integral plant response codes) have evolved as the most effective means of representing the range of interacting processes and phenomena.

The evolution of integral plant response codes, such as EPRI's Modular Accident Analysis Program (MAAP) or MELCOR, have enabled significant advancements in regulatory risk-informed decision-making. A number of activities post-Fukushima demonstrated the effectiveness of MAAP and MELCOR in the risk-informing of the regulatory decision-making process [30] [19]. Modern source term codes are thus fundamental to informing risk-informed decision-making.

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4. CONSIDERATIONS FOR NON-LWR SOURCE TERM DEVELOPMENT

4.1. Key Differences in LWR and Non-LWR Source Term Characteristics

There are significant differences between LWR and non-LWR systems. As with all nuclear power systems, the fundamental safety requirements are common: control the nuclear reaction, remove heat from the fuel system and contain radionuclides. These fundamental safety requirements form a basis upon which to frame a range of technology-inclusive safety requirements and safety assessment capability needs.

Within LWR and non-LWR systems, many of the requirements for modeling the phenomena, processes, or effects relevant to containment of radionuclides are common between the systems. Examples of features important to model include:

- Heat transfer between relevant fluids and heat structures
- Sources of energy generation
- Fission product release from fuel, whether liquid or solid
- Aerosol transport
- Deposition and condensation on structures
- Condensation, evaporation, and agglomeration of radionuclides
- Resuspension and revaporization of radionuclides
- Behavior of containment, including leakage pathways
- Concrete interactions
- Thermochemical interactions
- Radionuclide speciation and decay

A visualization of the similarities is shown in Figure 4-1.

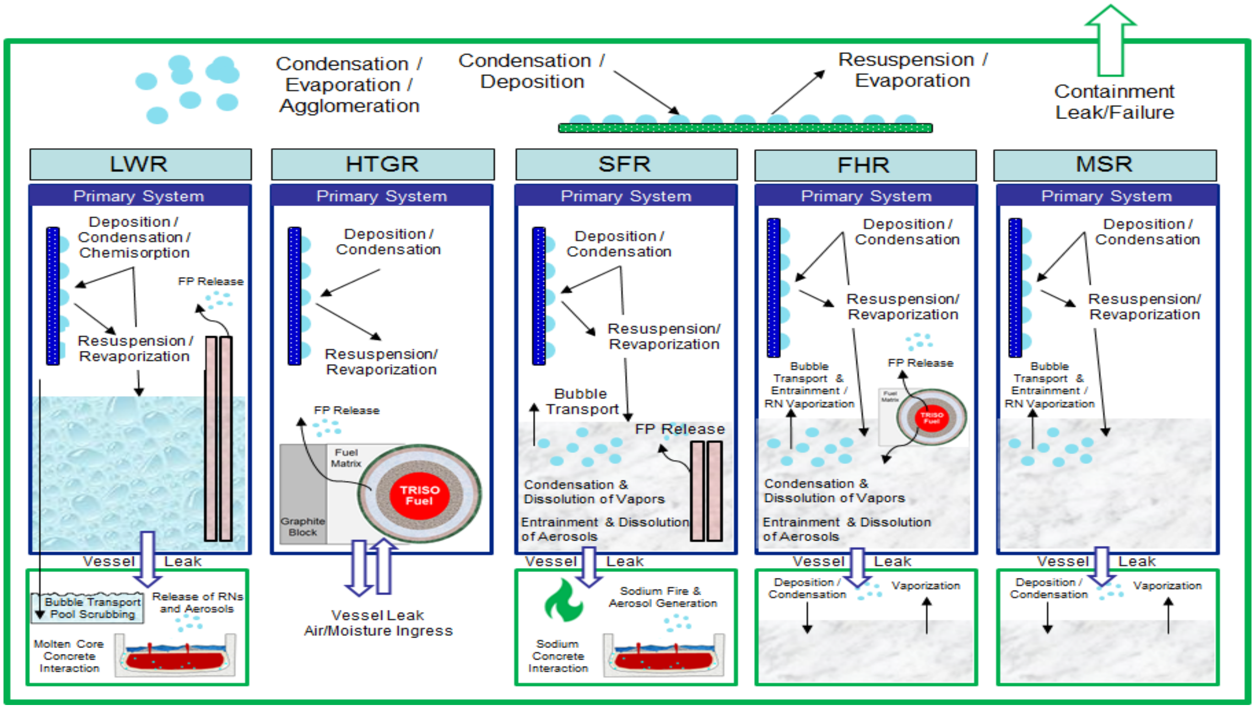

Figure 4-1. Overview of important phenomena to consider within LWRs and different non-LWR reactor types when evaluating source terms. Not all phenomena will be relevant during any given accident scenario, yet they must necessarily be represented within models. [12]

While both LWR and non-LWR systems have much in common when developing a mechanistic source term, there are still differences. These differences may necessitate the development of additional modeling capabilities, or require fundamental research to better characterize processes and phenomena to inform models. The systems themselves will be significantly different with different working fluids, chemical compositions, and system components. Some of the differences will mandate the development of new models, and include:

- Working fluids other than water and air, such as molten salt and sodium
- System components such as pumps, heat exchangers
- Differing fuel forms, to include TRISO fuel
- Fuel-dependent RN speciation and release rates
- Thermochemical interactions between chemical species of structures or fluids under accident conditions

Additional guidance on how to account for non-LWR source terms in a mechanistic manner can be aided by the DOE nuclear facilities safety basis analyses. Assessment of radiological consequences associated with these facilities has utilized a leak path factor analysis to determine what fraction of radionuclides move from a contaminated area to the environment during postulated accidents and other events. Such assessments are routinely performed with integral plant response codes such as MELCOR. These codes have incorporated the necessary models to capture the key processes and phenomena relevant to non-LWR accident progression evaluation and source term estimation. As a result, codes such as MELCOR are directly applicable to characterizing non-reactor radiological release events, such as those that could occur due to malfunctions of an off-gas system. [42]

While each of the reactor design types addressed in this report have unique internal accident initiators, multiple external event initiators and considerations are common amongst all reactor types. Currently the NRC has prescribed licensing basis events for LWRs. However, given the difference between advanced reactors and LWRs, these licensing basis events will generally be different.

4.2. External Events

All advanced reactor types will need to address externally initiated accident scenarios. These scenarios, and the different challenges they present to non-LWR systems may need to be taken into account when performing source term modeling. External events may also compromise passive heat removal systems that are relied upon to cool the system. Such events include but are not limited to:

- Flooding has the potential to introduce water into systems, which are designed to operate without water present. This could be complicated by building reactors below grade or present a unique challenge for less robust confinement structures. The introduction of water into systems could also introduce new accident scenario considerations for non-LWRs. For example, water and sodium are not compatible fluids and their interaction promotes energetic reactions that will fundamentally alter the characterizion of radiological transport.
- Seismic events remain a concern for advanced reactors.
- Projectiles caused by high wind events such as tornados and hurricanes will pose threats to buildings and SSCs contained within these buildings. The performance of confinement structures, that may replace traditional LWR containments as part of a functional containment strategy, may require additional evaluation to assess adequacy of performance under high wind events.

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5. MOLTEN SALT REACTOR FUNCTIONAL ANALYSIS

5.1. System Overview

The molten salt reactor is unique in that in this reactor design fissile elements are not stationary as they are in other designs but rather, they are dissolved into the thermal fluid salt of the primary system. Thus, they circulate through the reactor itself, the primary heat exchanger and primary system pumps. This creates unique fission product inventory management problems. These systems often contain three total loops, the primary loop with the fuel salt, an intermediate loop and a third loop which is used to generate electricity and dump excess to an ultimate heat sink (UHS). The intermediate loop provides an additional barrier to fission product release. A diagram of a generic MSR is provided in Figure 5-1.

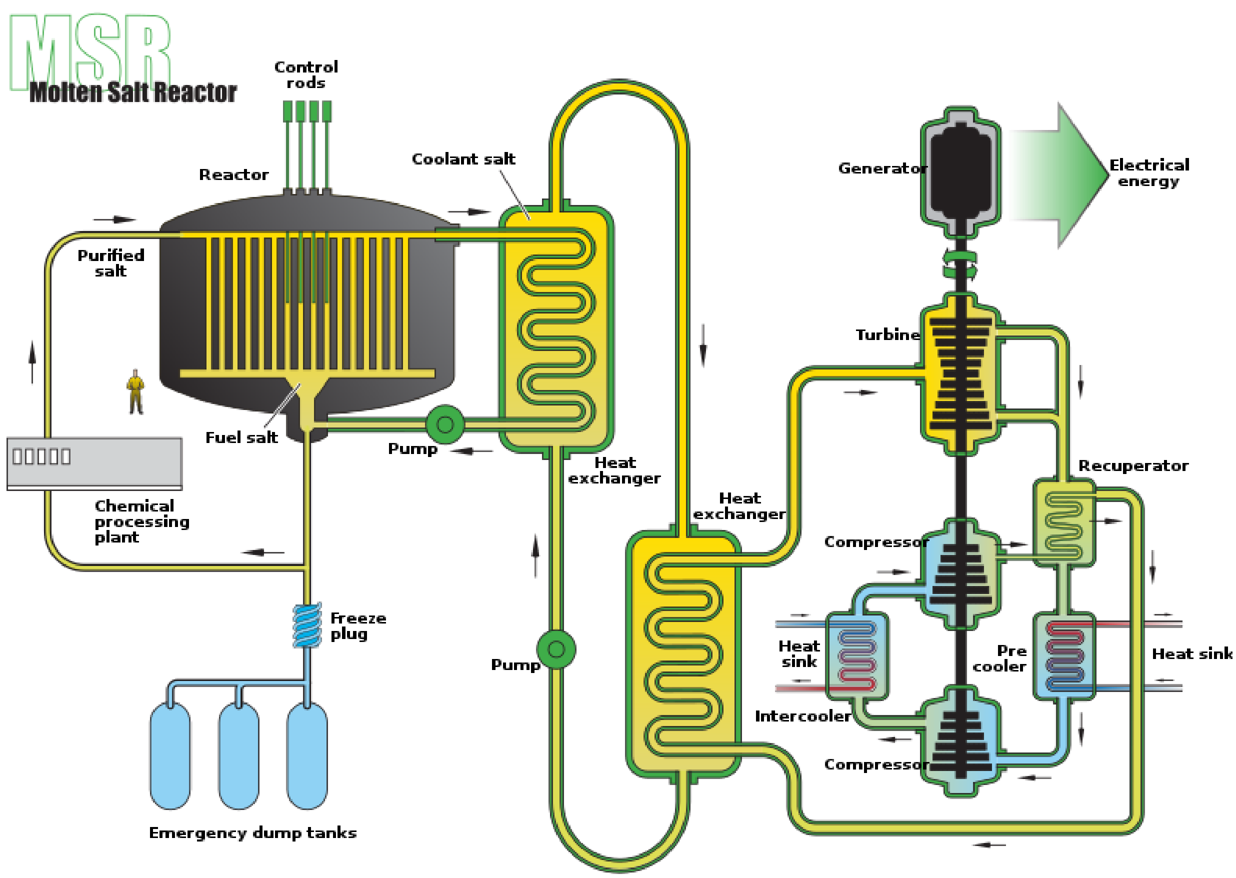

Figure 5-1. Diagram of the Molten Salt Reactor developed as part of the Generation IV International Forum [44]

This analysis presents a functional review of the Molten Salt Breeder Reactor (MSBR). A diagram of this system is shown in Figure 5-2. The MSBR contains two distinct salts in the reactor core, with one salt containing primarily fissile heavy elements and the other containing fertile heavy elements. Such breeder systems are U-238/Pu-239 or Th-232/U-233. In this system both the fuel salt and the blanket salt will contain fission products. As a note, not all proposed MSR designs contain a blanket salt; some designs are optimized for burning spent fuel while others are more focused on a once-through cycle.

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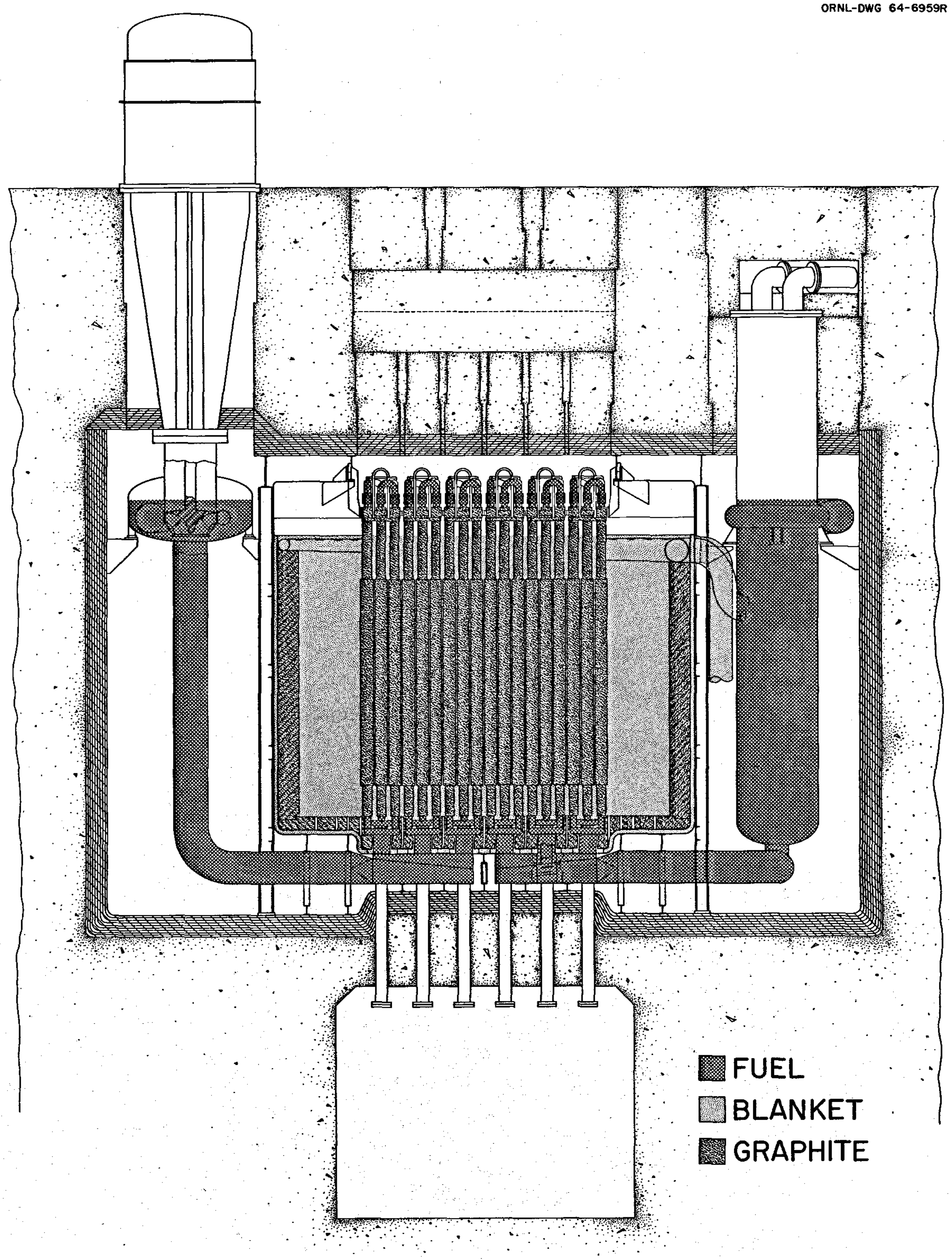

Figure 5-2. Molten Salt Breeder Reactor (MSBR) Diagram [45]

5.2. Functional Tree

The molten salt breeder reactor (MSBR) [45] is described in separate reports so an in-depth study and description of the MSBR is purposely left absent. One unique feature of molten salt reactor technology is the dilution of fuel into the coolant itself. In this way, the fuel is not located in one "central" location, as with LWR reactor cores. In addition to this feature, many breeder reactor designs encompass fuel/blanket salt processing plants on-site to help regulate breeding and delivery of fuel to the primary fuel salt loop. This results in several sources of radioactive material in different locations of the plant, as shown in Figure 5-3. For the sake of demonstration, this analysis will be constrained to just the fuel salt loop. [45]

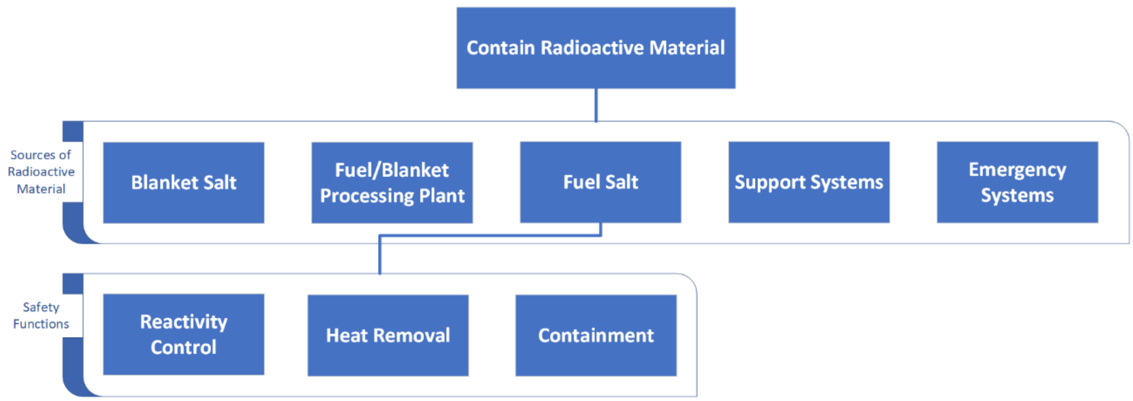

Figure 5-3. Sources of radionuclides within the MSBR

There are three primary safety functions of MSBR: reactivity control, heat removal, and containment. These are shown visually in Figures 5-3 and 5-4. Again, a complete system-by-system description is provided in the list of references.

The importance of these safety function trees is it shows the systems and processes that a source term code will be expected to model. For example, assuming that an event sequence and/or event sequence family properly drains to the emergency drain tank, a source term code will still be needed to predict the event sequence, namely the modeling of reactivity control, heat removal, and containment (assuming that it differs from the primary containment/confinement).

It is expected that as the design matures, each of these safety function trees will grow and include more detailed information. Furthermore, it is expected that fault trees could eventually be developed for each of these safety functions.

It is also important to note that these safety function trees do not encompass the entire range of safety phenomena that would influence event sequence progressions. For example, the fuel salt chemistry is not represented in this figure, but local fuel salt chemistry will influence the source term release and transport mechanisms that are important in a source term code.

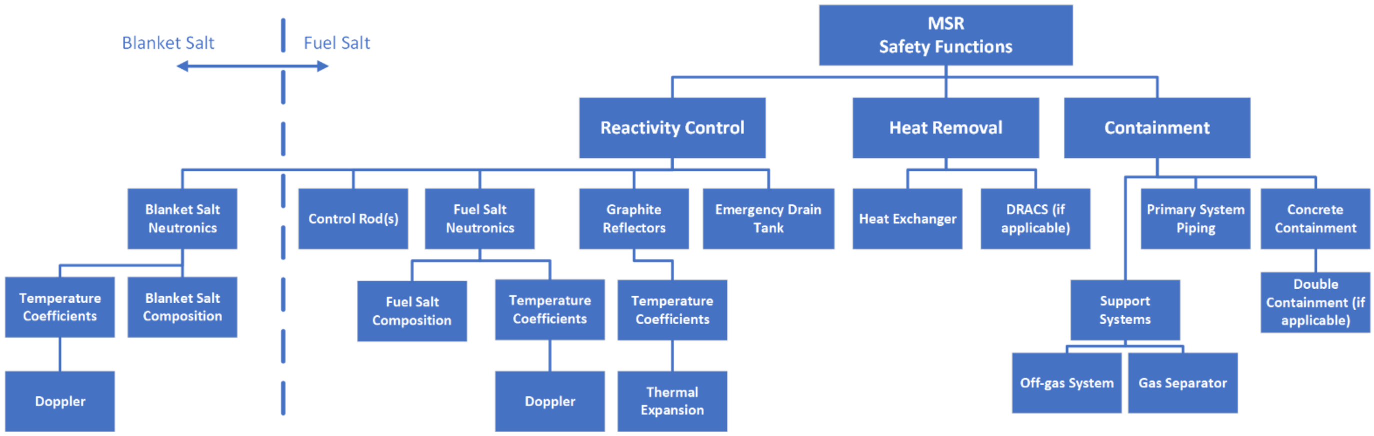

Figure 5-4. Safety function diagram for the MSBR

6. HIGH TEMPERATURE GAS REACTOR FUNCTIONAL ANALYSIS

6.1. System Overview

A diagram of a generic high temperature gas reactor is provided in Figure 6-1. HTGR systems use a gas as a coolant under elevated pressures. This review examines the primary system of a generic HTGR that uses helium as a coolant and graphite as a moderator. The fuel is a compact of spherical tri-structural isotropic (TRISO) either in a pebble or prismatic geometry.

TRISO fuel is a multilayer fuel form that contains a uranium oxycarbide (UCO) fuel kernel surrounded by a porous carbon buffer, silicon carbide and two pyrolytic carbon layers. TRISO fuel can then be compacted into any designed shape, with the most common two fuel matrix designs being pebbles and prismatic fuel elements. A diagram of TRISO fuel can be seen in Figure 6-2.

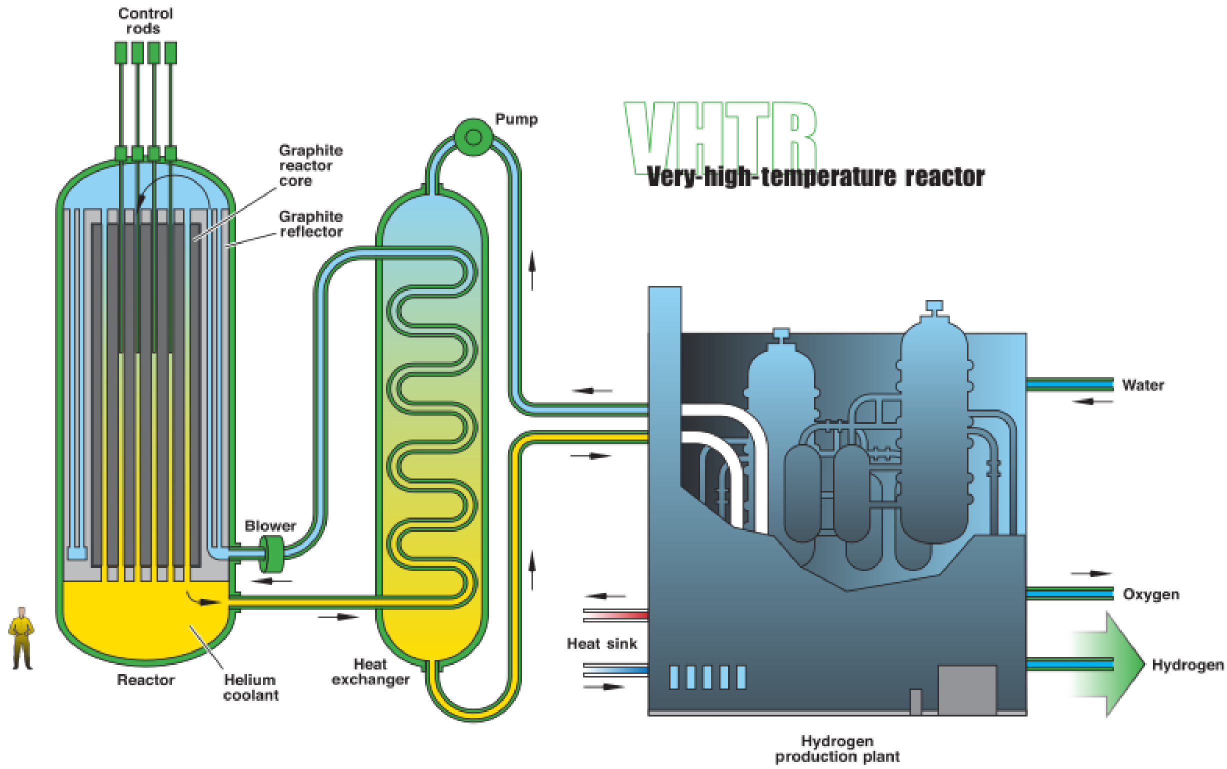

Figure 6-1. Diagram of the Very High Temperature Reactor developed as part of the Generation IV International Forum [44]

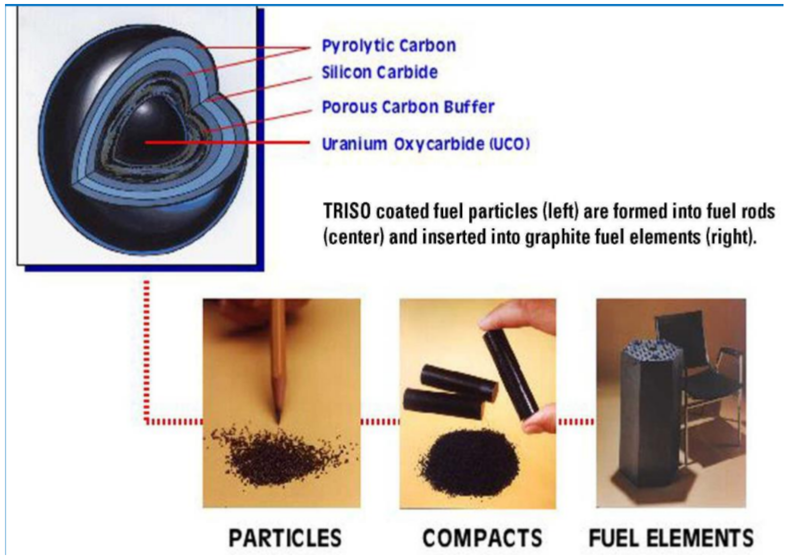

Figure 6-2. TRISO fuel layers and final compact form [46]

6.2. Function Tree

For the High Temperature Gas Reactor (HTGR), the key safety functions were organized into a safety functional diagram (see Figure 6-3). The diagram is organized into three individual branches: reactivity control, heat removal and containment, which comprise the safety theme of the reactor design. Each branch is then further divided into the contributing safety components. A description of each branch is provided below.

Reactivity control for the HTGR design is organized by three key contributors:

- Fuel: the fuel (either pebble or prismatic) consists of several layers including the inner fuel kernel, carbon buffer, pyrolytic carbon, silicon carbide, and pyrolytic carbon outer layer. Temperature increases within the fuel lead to reductions in reactor power due to the negative temperature coefficient.
- Control rods: the control rods include the normal operation reactivity control rods themselves, the control rod drive system, and the channels permitting control rod movement. HTGR benchmarking experiments demonstrate control rod worth up to -2.55$ per rod depending on core positioning [Add IAEA Benchmarking]
- Reactor reserve shutdown system: the reactor reserve shutdown system introduces additional negative reactivity into the core by gravity dropped borated graphite spheres, further increasing the shutdown margin in accident scenarios.

Some pebble bed gas reactor designs also contain "poison pebbles" for additional reactivity control in addition to control rods; these pebbles are not treated explicitly within this analysis. Accident scenarios include localized reactivity excursions (stuck fuel pebble for example) and control rod malfunctions. [47]

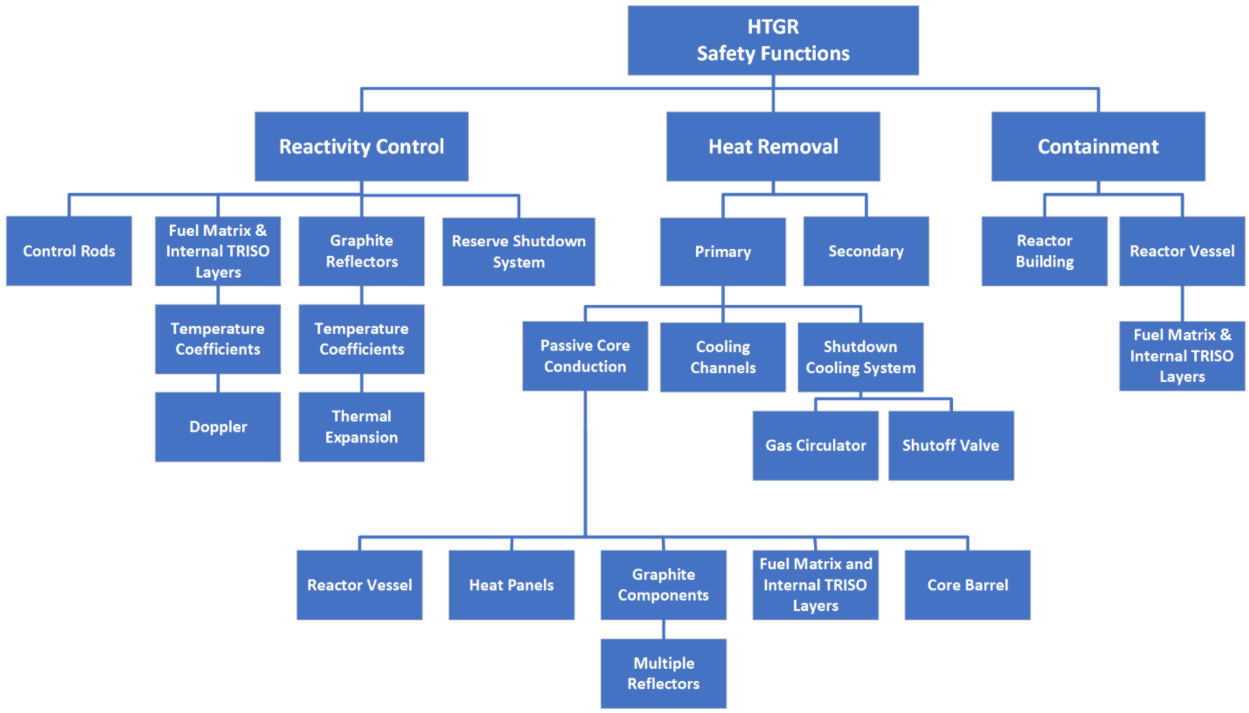

Figure 6-3. Safety function diagram for the HTGR

The heat removal branch is divided between the primary and secondary systems. For the primary system, heat removal mechanisms can be divided into both active and passive heat removal as described below.

Active heat removal mechanisms include convection from the circulation of the coolant gas (i.e., helium in the concept assumed), the flow path of the coolant gas to include the inlet and outlet lines and associated valves, and the cooling channels within the reactor core. The shutdown cooling system (which operates the gas circulator and cooling shutoff valve) also contributes to active heat removal. Passive heat removal mechanisms rely heavily on natural conduction and convection, to include conduction within the fuel (from the fuel kernel out through the multiple layers of the fuel), and natural conduction and convection throughout the pebble bed or graphite matrix, through the multiple reflectors, the core barrel, the reactor pressure vessel (RPV), and finally to the heat panels on the exterior of the reactor vessel.

Several accident scenarios may arise from the potential for structural damage/displacement of multiple elements of the heat removal system. When forced cooling is present, one relies strongly on convective heat removal due to flow of cooling over the core. When forced cooling is lost, the role of conduction and radiation within the core to the periphery increase significantly and become the dominant heat transfer pathways. Anything that would tend to interfere or interrupt those pathways would degrade the overall passive mode of heat removal. Natural disasters (like earthquakes) could cause damage to the gas coolant lines, channels within the core, and/or displacement of the heat panels on the exterior, thereby creating potential impediments to both active and passive heat removal pathways. Temperature excursions could result, with a diminished ability to remove heat from the reactor core. [48]

The secondary system consists of a standard steam generation system that is not unique to HTGRs and is therefore not further described in this functional diagram.

For HTGRs, containment of radioactive material is typically achieved through a multiple-barrier approach. Due to the robust nature of TRISO fuel under even extremely elevated temperature conditions, the fuel matrix and structure serve as a primary barrier. Should fission products escape from TRISO fuel and the fuel pebble or compact, they will distribute throughout the reactor system and deposit on reactor structures. In the event that the structural integrity of the reactor system is lost, fission products will escape into a reactor confinement around the reactor and reactor system. Many vendors are likely to contend that, due to the robust nature of TRISO fuel under accident conditions, a functional containment strategy can be developed that utilizes a less robust reactor confinement structure. This will obviate the need to incorporate a relatively costly containment structure as found on current generation LWRs.

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7. FLUORIDE-SALT-COOLED HIGH TEMPERATURE REACTOR FUNCTIONAL ANALYSIS

7.1. System Overview

A diagram of the reactor vessel of the MK1 Pebble Bed FHR (PB-FHR) is provided in Figure 7-1. This reactor design contains pebble shaped TRISO fuel compacts that are buoyant in a downward flowing fluoride salt. The PB-FHR design reviewed is taken from a joint DOE-University project. For this reactor concept, FLiBe (mixture of lithium fluoride and beryllium fluoride) is the chosen coolant. A schematic of the remainder of the PB-FHR system is shown in Figure 7-2, containing the secondary side and ancillary systems. As can be seen, this system is a hybrid of a pebble bed HTGR and an MSR. A more comprehensive system description can be found in reference materials. [49]

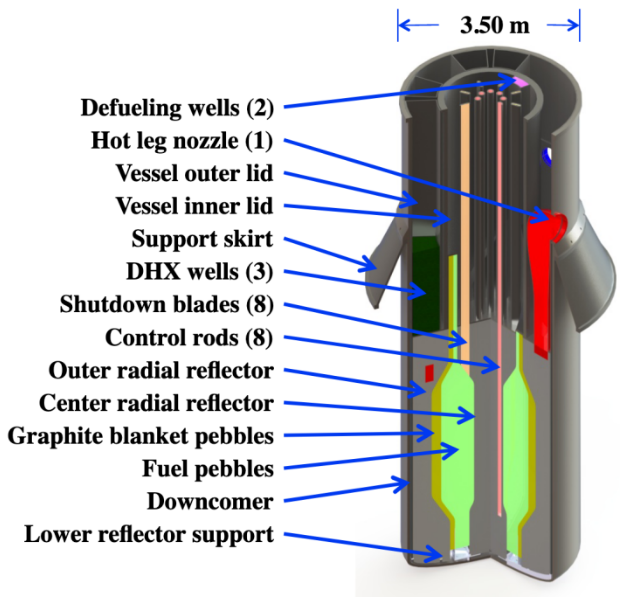

Figure 7-1. Mk1 PB-FHR reactor vessel [49]

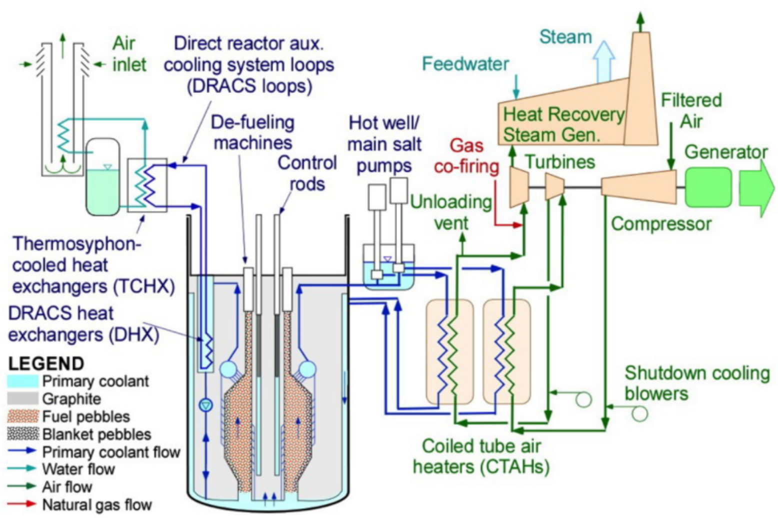

Figure 7-2. Mk1 PB-FHR flow schematic [49]

7.2. Function Tree

There are three primary safety functions of an FHR: reactivity control, heat removal, and containment. A complete description of the FHR system is provided within the list of references. The function tree of the system is shown in Figure 7-3.

Reactivity is controlled within the system through a combination of natural feedback within the fuel and coolant, as well as control rods.

Heat must necessarily be removed from the TRISO fuel particles where it is generated during fission and radioactive decay. For the primary system, heat removal mechanisms can be divided into both active and passive heat removal as described below. Active heat removal mechanisms include cooling from the circulation of the fluoride coolant and the associated flow path. Passive heat removal mechanisms rely heavily on natural conduction and convection, to include conduction within the fuel (from the fuel kernel out throught the multiple layers of the fuel), and natural circulation employed by the Direct Reactor Auxiliary Cooling System (DRACS).

Containment of radionuclides is maintained through the primary system reactor tank, and within the structure of the TRISO fuel kernels. Additionally, the integrity of the reactor must be maintained to ensure that there is no gross transfer of radionuclides to the environment. The off-gas system has a key safety purpose in removing gaseous radionuclides from the primary system coolant. It also serves as a tritium scrubber, removing a significant amount of the isotope from the system, when it is in the form of tritium gas (HT).

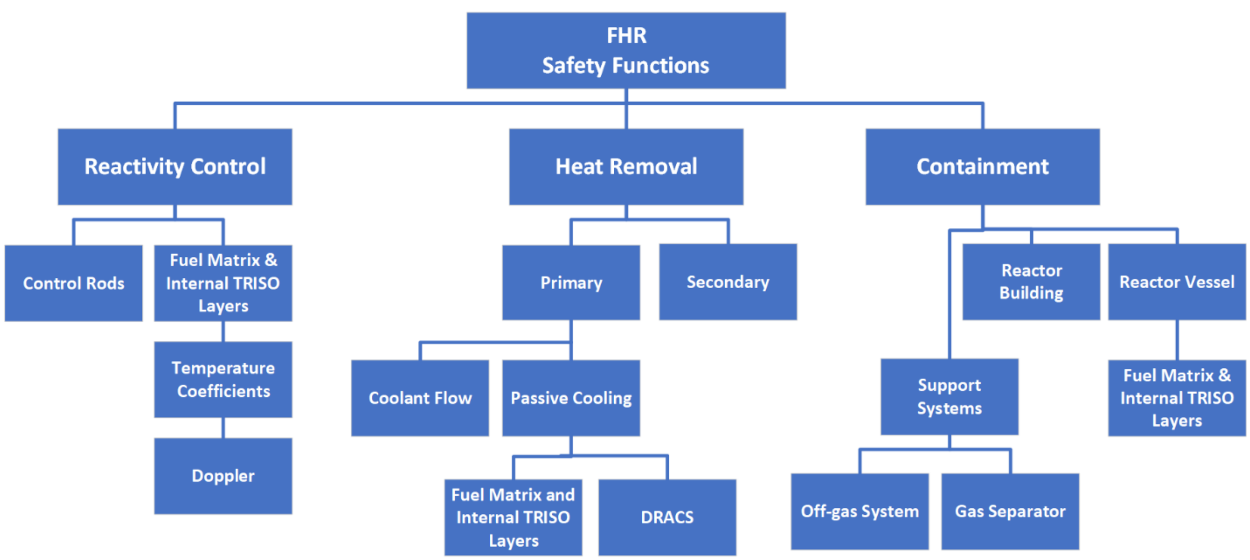

Figure 7-3. Safety function diagram for the FHR

8. PATH FORWARD

This document serves as a starting point for the functional analysis of the source term released from advanced reactors during off-normal events. This document highlights several event scenarios of interest and phenomena/processes necessary to capture in the accident modeling for each event. Future work will continue on this effort in FY21 within the NEAMS program, with contributions from both Sandia National Laboratories and Argonne National Laboratory.

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