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Risk-Informed, Performance-Based, Technology-Inclusive Regulatory Infrastructure

June 2020

Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities

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Changing the World's Energy Future

Idaho National Laboratory

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EXECUTIVE SUMMARY

This report summarizes a risk-informed, performance-based, and technology-inclusive approach to determine source terms for dose-related assessments at advanced nuclear facilities to support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495) [1] and the NRC's response to the Nuclear Energy Innovation and Modernization Act (NEIMA) Public Law No: 115-439, of January 2019 [2]. This approach uses a graded process that allows both the non-mechanistic source terms calculation methods, which adopt conservative approaches and assumptions based on known physical and chemical principles, and, more importantly, the mechanistic source term calculation methods, which consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of licensing basis events to be used for the design and licensing of advanced nuclear technologies.

The source terms developed with this graded approach and radionuclide inventories elsewhere in the facility that are determined during source term analysis can be used to address licensing issues to support the application processes of 10 CFR Part 50 for a construction permit and operating license or 10 CFR Part 52 for a Combined Operating License (COL), Standard Design Certification, Early Site Permit, Standard Design Approval or Manufacturing License. They can also be used for other purposes, including equipment environmental qualification, control room habitability analyses, and assessments of severe accident risks in environmental impact statements.

There are many advanced reactor concepts being developed, including the high-temperature gas-cooled reactor, sodium-cooled fast reactor, lead-cooled fast reactor, molten-salt reactor, and microreactor. The graded approach presented in this report for source terms determination is, to the extent possible, generic to any of these reactor designs and to future reactor designs.

This report provides information on the review of the regulatory foundation for the use of conservative bounding source terms as well as event-specific mechanistic source terms for advanced nuclear reactor designs.

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ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AOO	anticipated operational occurrence
AST	alternative source term
BDBE	beyond design basis event
CD	core damage
CDF	core damage frequency
DBA	design basis accident
DBE	design basis event
DBEHL	design basis external hazard level
DID	defense-in-depth
DOE	Department of Energy
EAB	exclusion area boundary
EPA	Environmental Protection Agency
EPZ	emergency planning zone
F-C	frequency - consequence target chart
FMEA	failure modes and effects analysis
HTGR	high-temperature gas-cooled reactor
INL	Idaho National Laboratory
LBE	licensing basis event
LMP	licensing modernization project
LPZ	low population zone
LWR	light-water reactor
MACCS	MELCOR Accident Consequence Code System
MCA	maximum credible accident
MHTGR	modular high-temperature gas-cooled reactor
MST	mechanistic source terms
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NGNP	next generation nuclear power plant
NRC	US Nuclear Regulatory Commission
ORIGEN	Oak Ridge Isotope GENeration
PAG	protective action guide

PDS	plant damage state
PIRT	phenomena identification and ranking table
POS	plant operating state
PRA	probabilistic risk assessment
QHO	quantitative health objective
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
SCALE	Standardized Computer Analyses for Licensing Evaluation
SHA	system hazard analysis
SMR	small modular reactor
SRIR	site radionuclides inventories at risk
SSC	structures, systems, and components
STPA	system-theoretic process analysis
TEDE	total effective dose equivalent
UR	undesirable release
URF	undesirable release frequency

1. OVERVIEW

1.1 Purpose

The primary purpose of this report is to describe a risk-informed, performance-based, technology-inclusive determination of source terms for dose-related assessments for advanced nuclear reactor facilities to support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495) [1] and the NRC's response to the Nuclear Energy Innovation and Modernization Act (NEIMA) Public Law No: 115-439, of January 2019 [2].

The regulations in 10 CFR Part 20 [3] establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the U.S. Nuclear Regulatory Commission (NRC), which is associated with the assessment of plant conditions and forecast, and actual or projected radiological assessments.

The radiological accident consequences analysis for reactor siting is described in 10 CFR 50.34(a)(1), which establishes regulatory dose criteria at the reactor's exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ) [4]. Guidance on radiological source terms and consequence analysis is derived from this regulation for satisfying regulatory requirements and Commission Policy, as related to limiting the effects on public health and safety and other societal consequences in the event of accidents. Other current NRC regulations associated with source terms include 10 CFR50.49(e)(4), which applies to environmental qualification of electrical equipment based on the most severe design basis accidents (DBA), and control room habitability requirements in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, which specifies habitability dose criteria in the control room under accident conditions for current light-water reactors (LWRs) and may also be considered for advanced reactors.

The variety of advanced nuclear reactor technologies and designs has led to an increased use of radiological consequences as acceptance criteria for decisions related to design and licensing. Examples include the sizing of emergency planning zones (EPZ) based on estimated offsite consequences and safety classification of structures, systems, and components based on their role in preventing or mitigating offsite consequences. In an October 19, 2018 letter from the Advisory Committee on Reactor Safeguards (ACRS) to the Commission, a comment related to draft regulatory guide DG-1350, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," [5] on performance-based EPZ stated that it was "important for the staff to provide guidance on how source terms should be developed." This is because, without additional source terms development guidance to technologies other than those that are LWR-centered, the staff would need to review design and licensing information on a case-by-case basis, which is contrary to the Commission goal of reducing regulatory uncertainty for other nuclear technologies. The ACRS letter further noted that "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors" (SECY-16-0012) [6] stated that the staff "have been in pre-application discussions with small modular reactor (SMR) designers, and the methods proposed by potential applicants appear to generally build on currently approved methods." Additionally, in a March 19, 2019 letter addressing a review of draft regulatory guide DG-1353 (finalized as RG 1.233), "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," [7] the ACRS stated that "guidance for developing mechanistic source terms should be expanded."

NEIMA directed the NRC to:

> develop and implement, where appropriate, strategies for the increased use of risk-informed, performance-based licensing evaluation techniques and guidance for commercial advanced nuclear reactors within the existing regulatory framework, including evaluation techniques and guidance for the resolution of source terms policy issues described in SECY–93–092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," [8] and SECY–15–077, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies," [9] and identified during the course of reviews by the Commission of commercial advanced nuclear reactor licensing [pre-applications or] applications.

NEIMA specifically identified mechanistic source terms (MST) as one of the issues for which regulatory guidance should be prepared by January 2021. The scope of this document is focused on developing a risk-informed, performance-based, and technology-inclusive methodology for the determination of the source terms up to release to the environment for advanced reactors. Developing methodologies for dose determination, such as transport in the environment, exposure pathways, dose factors, and human health impacts and shielding, is outside the scope of this document.

1.2 Background

The use of postulated accidental release of radioactive materials and consequent radiological doses has long been deeply embedded in the regulatory policy and practices in the licensing and siting of nuclear reactors and protection of public health. However, large uncertainties exist in the analysis of the details of the timing and type of accident that could occur and the related amount of radioactive material that could be released in the event of an accident. Non-mechanistic methods, using conservative approaches and assumptions based on known physical and chemical principles, have been traditionally used for LWRs to yield conservative dose estimates to demonstrate compliance with regulatory requirements. As stated in "Policy Issues Related to Licensing Non-Light Water Reactor Designs" (SECY-03-0047) [10], "current light-water-reactors (LWRs) use site-specific parameters (e.g., exclusion area boundary) and a deterministic predetermined source term into containment to analyze the effectiveness of the containment and site suitability for licensing purposes." The LWR non-mechanistic source terms were first described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," [11] which was published by the United States Atomic Energy Commission in 1962. TID-14844 specified a non-mechanistic approach in the calculation of the amount of fission product inventory release to the containment atmosphere (i.e., "in-containment accident source term" or "source term") to calculate the radiological doses of the "maximum credible accident (MCA)" resulting from substantial core meltdown as a bounding fission product release in an LWR. The LWRs currently operating in the U.S. were licensed originally based on "in-containment source terms" specified in Regulatory Guide (RG)-1.3 [12] and RG-1.4 [13], with the specifications derived from TID-14844. The MCA is postulated as a nuclear accident that would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of the facility. For example, for the operating light-water reactors, the MCA has been frequently postulated as the complete loss of coolant upon the complete rupture of a major pipe (large-break loss-of-coolant accident). Conservative assumptions are used to compensate for uncertainties in the source term calculations for the purpose of calculating offsite doses in accordance with 10 CFR Part 100, "Reactor Site Criteria" [14]. For example, according to TID-14844, 100% of the core inventory of noble gases and 50% of the iodine (half of which are assumed to deposit on containment interior surfaces very rapidly) are assumed available for release to the atmosphere with a constant leakage rate of 0.1% per day. Using this approach would result in exposure doses probably many times higher than what would actually be expected, even if the postulated MCA should occur.

Since the publication of TID-14844, substantial additional information on fission product releases has been developed, in terms of the timing, nuclide types, quantities, and chemical form, based on significant severe accident research. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," [15] which specifies a revised source term methodology to formulate an alternative to the postulated source terms used in the past. This revised source term was more physically based to provide more realistic estimates of the source terms release into containment, given a severe core-melt accident. NUREG-1465 presents representative accident source terms for LWRs (one for pressurized-water reactors and a similar one for boiling-water reactors) and is applicable to the operating LWRs as well as future LWRs. These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. Information on the gap and in-vessel release phases from NUREG-1465 were adapted into the regulatory practices of NRC in 2000 through RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" [16]. RG 1.183 provides guidance on an acceptable alternative source term (AST) for design basis radiological consequences analyses, such as those addressed in Chapter 15 of typical LWR final safety analysis reports. In addition to providing acceptable inputs and assumptions for an AST based on NUREG-1465 [15], RG 1.183 [16] also described the attributes of an acceptable accident source term for licensees that wished to develop their own alternative. An AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and that has been approved for use under 10 CFR 50.67, "Accident source term." The alternative source term is not based upon a single accident scenario but instead must represent a spectrum of credible severe accident events.

Although initially used only for siting evaluations, the source term has been used in other design basis applications. As discussed in SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," dated December 19, 1994, [17] the staff uses reactor accident source terms such as given in TID-14844 [11] and the later issued RG-1.183 [16] not only for assessing potential doses to the public following an accident but also in areas such as:

(1) equipment qualification under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,"
(2) control room habitability,
(3) engineered safety features,
(4) atmosphere cleanup systems,
(5) primary containment leak rate,
(6) containment isolation timing,
(7) post-accident sampling, and
(8) shielding and vital area access.

Analogous to the LWRs, quantitative determination of the radioactive materials that could potentially escape from an advanced reactor during normal operation or as a result of an accident and ultimately be released to the environment plays a critical role in the facility's design and NRC's requirements to protect public health against radiation hazards. For advanced reactors, as described in the HTGR Mechanistic Source Terms white paper (INL/EXT-10-17997 [18]), the phrase "source terms" refers to the quantities, timing and other characteristics of radionuclides released from the facility to the environment. It is noted that for LWRs, the phrase "source terms" refers to the magnitude and mix of radionuclides released from the fuel to the containment atmosphere, expressed as fractions of the fission product inventory in the fuel as well as their physical and chemical form, and the timing of their release. The advanced reactors have significant design differences relative to the existing LWRs, specifically with regard to materials, coolant, reflectors, and potential applications. Examples of coolant-based advanced reactor designs include sodium-cooled fast reactors, lead-cooled fast reactors, high-temperature gas-cooled reactors (HTGR), and molten-salt reactors. These designs propose using different barriers to the release of radionuclides, which resulted in the need to change to a technology-inclusive reference release location (i.e., environment vs. containment) in the definition of source term for non-LWRs as compared to that used for LWRs. For additional information on functional containment in lieu of leak-tight containment structures, see SECY-18-0096 [19], approved by the Commission staff requirements memorandum dated December 4, 2018 (SRM-SECY-18-0096 [20]).

Advanced reactors may be designed with various power output levels and fall into three categories—large reactors, SMRs, and microreactors. Although not explicitly defined in the regulation, large reactors are generally designed to operate at thermal power levels greater than 1,000 MWt, SMRs up to 1,000 MWt, and microreactors up to 50 MWt. Advanced reactors are designed with inherent or passive safety features to remove decay heat in an effort to enhance the safety for the plant workers and the public. Advanced reactors may be modularly constructed, and, specifically, the SMRs' small size allows them to be deployed in areas with smaller energy needs, their small size allows for more site flexibility and additional reactor units can be incorporated into the design as needed and clustered to create a multimodule, large capacity power plant.

Microreactors, on the other hand, are designed to be factory manufactured and transported. These reactors are referred to as special purpose reactors with the ability to provide heat and power to remote communities and industrial users. These reactors are designed to be self-regulating and not rely on physical systems to ensure the safe shutdown and removal of decay heat.

Because most advanced reactors are expected to operate at a lower power level, the amount of radioactive material released to the air during normal operations and under accident conditions may be reduced, compared to large LWRs. For example, a reduction in source terms allows the LPZs, EPZs, and the distances required to meet dose-consequence regulatory criteria to be adjusted to better fit the facility size. As SECY-16-0012 [6] stated:

> These reduced source terms could form the basis for an applicant request to establish emergency planning zones that are smaller than what is currently required by Title 10 of the Code of Federal Regulations (10 CFR) 50.47(c)(2). In addition, the reduced source terms could result in smaller exclusion areas and LPZs as defined in 10 CFR 100.3, as determined in accordance with the safety assessment and dose criteria in 10 CFR 50.34(a)(1). Any NRC-approved reduction in the size of the LPZ could, in turn, allow such a reactor to be sited in closer proximity to a large population center as compared to large LWRs, as provided under 10 CFR 100.21(b). Any proposed site would also need to be consistent with other NRC requirements including 10 CFR 100.21(h), which limits, in qualitative terms, how close to the large population center a site can be.

Significant progress has been made through the years in understanding reactor accident behavior for LWRs, including fission product release and transport. This increased technical understanding results in more detailed mechanistically-based assessments of source terms, or mechanistic source terms, to estimate the release and behavior of these fission products, which may be applicable to advanced reactors. However, recent NRC activities related to advanced reactors (e.g., functional containment performance criteria (SECY-18-0096 [19]), scalable EPZ sizes (SECY-18-0103 [21]), possible changes to security requirements (SECY-18-0076 [49]), and the licensing basis considerations of RG 1.233 [7]) recognize the limitations of existing LWR-related guidance, which requires a return to first principles such as fundamental safety functions supporting the retention of radionuclides. Toward that end, NEI 18-04 [22], "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development", presents a process for the licensing of advanced non-LWRs developed by the industry-led Licensing Modernization Project (LMP). In that document, a modern, technology-inclusive, risk-informed, and performance-based process is defined for the 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. The LMP process uses a set of frequency-consequence criteria (F-C target), as shown in Figure 1-1, to select LBEs and classify SSCs. As described in NEI 18-04, the risk-informed licensing basis uses a F-C target curve to describe dose criteria as a function of event scenario frequency.

In June of 2020, NRC issued RG 1.233 [7] and endorsed NEI 18-04 as "one acceptable method for non-LWR designers to use when carrying out these activities and preparing their applications." Mechanistic source terms play a critical role in evaluating the consequences of LBEs, which are in turn considered in establishing the safety classification and performance criteria for SSCs, and assessing DID for the design and related programmatic controls. The mechanistic source terms are used to estimate the radiological consequences within the analyses of event sequences as described in NEI 18-04 to compare to the F-C target curve in the selection and evaluation of LBEs. RG 1.233 describes the relationship as follows:

> Although NEI 18-04 does not address the topic in detail, the development of mechanistic source terms for designs and specific event families is another element of an integrated, risk-informed, performance-based approach to designing and licensing non-LWRs. The NRC staff expects applications or related reports to describe the mechanistic source terms, including the retention of radionuclides by barriers and the transport of radionuclides for all barriers and pathways to the environs. Where applicable, a facility may have multiple mechanistic source terms and specific event sequences to address various systems that contain significant inventories of radioactive material.

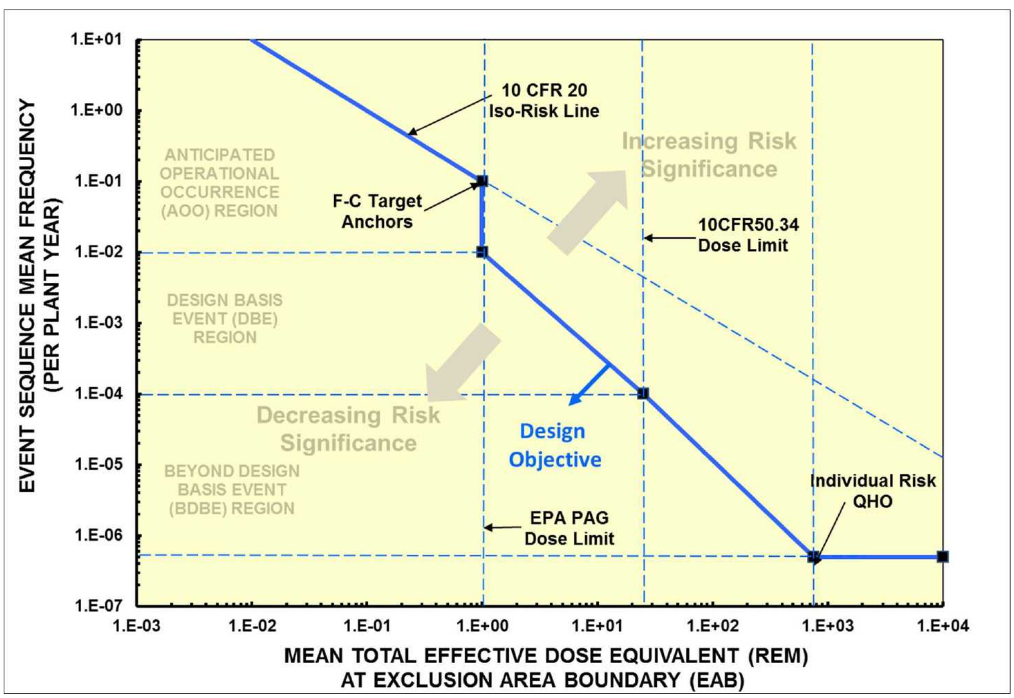


Figure 1-1 F-C target curve (NEI 18-04 [22]).

SECY-03-0047 [10] defines "mechanistic source term is the result of an analysis of fission product release resulting from the design-specific accident scenarios and accident progression being evaluated. It is developed using best-estimate phenomenological models with uncertainty quantification of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs." The use of a mechanistic analysis includes accounting for fission product retention and removal processes, as illustrated in Figure 1-2 for one non-LWR concept, and can substantially attenuate the magnitude of the release as compared to a more non-mechanistic approach.

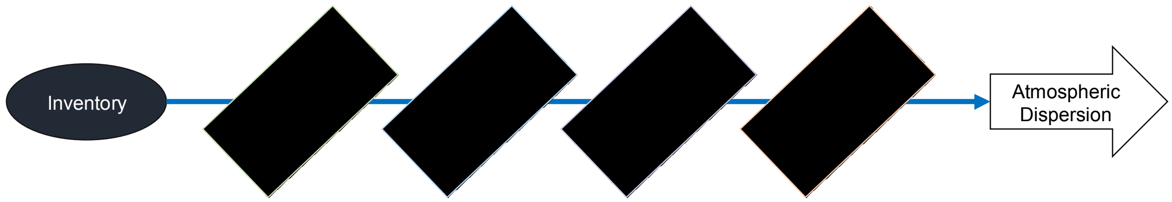


Figure 1-2 Illustration of radionuclides retention and removal process for one non-LWR concept (reproduced from SAND2020-0402 [23]).

The mechanistic source term, for the non-LWR concept illustrated in Figure 1-2, can be correlated using the following multifactor formula:

(1)

where:

is the total release to the environment of radionuclide *over the* entire release duration time (t)

is the initial fission product inventory at the time of the reactor accident for radionuclide

is the fraction of release of radionuclide from fuel system boundaries to the fuel matrix

is the fraction of release of radionuclide from fuel matrix to primary system

is the fraction of release of radionuclide from primary system to leak path *is the fraction of release of radionuclide from* leak path to the environment

Equation (1) shows that all the factors that determine how much of the inventory is released across a given barrier and thus persists to the source term are accounted for in the calculation of source terms. Each factor is, in turn, a function of its initial design characteristics (e.g., materials), operating conditions (e.g., burnup, aging), and transient/accident conditions (e.g., time, temperatures, pressures, chemistry).

SECY-03-0047 [10] states that the mechanistic source terms should be allowed and defines a scenario-specific mechanistic source term that is based upon the characteristics of the fuel and plant to determine the magnitude, timing, and nature of fission product release from the core. “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, Volumes 1 and 2” (NUREG-1860 [24]) further defines the conditions under which design-specific and scenario-specific mechanistic source terms can be used in licensing. These conditions include:

- Having sufficient experimental data to confirm the source term (e.g., quantity and form of radionuclides, timing of release); and
- Accounting for uncertainties in the source term determination (e.g., use 95% confidence level).

Using an MST approach requires the availability of adequate tools and analysis methods with sufficient models and supporting scientific data that simulate the physical and chemical processes that describe the radionuclide inventories and the time-dependent radionuclide transport mechanisms to predict the radiological release for dose calculations. The other important facet in using MST is the development of the scenarios to be analyzed, with which the risk-informed and performance-based approach will be adopted. The risk-informed and performance-based approach integrates probabilistic risk assessment (PRA) methods and MST methodologies into a unified approach aimed at assessing the performance of a particular advanced reactor design to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty.

A "risk-informed" approach considers risk insights together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. As stated in RG 1.233 [7], “NEI 18-04 describes an expanded role for PRA for non-LWRs beyond current 10 CFR Part 52 requirements or Commission policy for potential applications under 10 CFR Part 50.” PRAs are used to estimate risk by predicting what could go wrong, the likelihood of occurrence, and the severity of the consequences. PRAs also ensure that “significant insights are not obscured by artificially biased results derived from the application of uneven conservatisms.” The risk-informed approach facilitates the integration of safety, security and preparedness (defense-in-depth) by having risk as a common measure with which to compare and assess the impact of each on the others. As such, the risk-informed approach provides the means to implement a unified concept for protecting public health and safety, the environment and the common defense, and security. It also helps ensure coherence among design, construction, maintenance, operation, security, and inspection.

A “performance-based” approach described in “Strategic Plan, Volume 3” (NUREG-1614, Vol. 3 [25]) focuses on desired, measurable outcomes as the primary basis for regulatory decision-making rather than prescriptive processes, techniques, or procedures. It leads to defined results without specific direction regarding how to attain these results. Performance-based regulatory actions focus on identifying performance measures that ensure an adequate safety margin and offer incentives for licensees to improve safety without formal regulatory intervention by the NRC. The main attributes for a performance-based approach described in NUREG-1614 are: (1) measurable, calculable, or objectively observable parameters that exist or can be developed to monitor performance, (2) objective criteria that exist or can be developed to assess performance, (3) licensees have the flexibility to determine how to meet the established performance criteria in ways that encourage and reward improved outcomes, and (4) a framework that exists or can be developed in which the failure to meet a performance criterion, while undesirable, will not in and of itself constitute or result in an immediate safety concern. Performance-based regulation focuses on effectiveness and efficiency of the decision-making process.

Combining risk-informed and performance-based approaches together yields a comprehensive approach, considering risk insights, engineering analysis and judgment including the principle of DID and the incorporation of safety margins, and performance history. This approach [26] enables the decision-making process to (1) focus attention on the most important activities, (2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes, and (5) focus on the results as the primary basis for regulatory decision-making. Using a risk-informed and performance-based approach allows important scenarios to be identified in the source term evaluation.

2. OBJECTIVE

The objective of this report is to describe a risk-informed, performance-based, technology-inclusive approach to determine source terms for dose-related assessments at non-LWR nuclear facilities. The developed approach uses a graded and iterative process, which allows both the non-mechanistic and more detailed mechanistic methods to be used in performing source term calculations. The non-mechanistic approach uses conservative models and assumptions based on known physical and chemical principles, and mechanistic source term calculation methods consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of LBEs.

This report supports the NRC staff and the nuclear industry by providing a general description on determining source terms, including mechanistic source terms for facilitating discussions among stakeholders. The approach outlined in Section 3 is applicable to advanced nuclear technologies, such as future non-LWRs, SMRs, microreactors, and may be useful for nonpower production or utilization facilities.

It is noted that advanced reactor applicants are not required to use an MST or the process laid out in a LMP. Applicants may choose to develop a source term for an MCA using mechanistic, deterministic, or a combination of methods. This document is formulated to support these methods.

Although the information in this document is focused on development of an MST for accident assessments to determine offsite dose consequences, the determination of radiological source terms for other licensing assessments has similar features. For example, the determination of the equilibrium coolant radionuclide inventory for assessment of the radiological waste system design would include similar initial steps, such as determination of the core inventory and release to coolant during normal operations. Similarly, the development of non-mechanistic source terms may use some similar steps but with a conservative bias for bounding information.

3. TECHNOLOGY-INCLUSIVE RADIOLOGICAL SOURCE TERMS METHODOLOGY

The end goal of the development of radiological source terms is to use the developed source terms to evaluate the safety and siting of the facility; evaluate radioactive material release mitigation systems, structures, and components; evaluate radiation protection design; or evaluate the environmental qualification of certain equipment to prove that resultant doses are within regulatory criteria.

The focus of this report is on developing mechanistic source term techniques for evaluating offsite radiological consequences, which could be used to make decisions related to matters such as plant design features, siting, and emergency planning zone sizes. Many methodology components are used within the process to determine the source terms. In some cases, a non-mechanistic methodology can be used, and a bounding case can be made for meeting the dose criteria without further use of mechanistic components. Figure 3-1 [18] illustrates a general list of components feeding the pathways to compare to radiological regulatory criteria. The source terms are the key to the bounding calculations and the radiological dose determination; therefore, development of the source terms is not complete until final acceptable radiological doses are determined for the design.

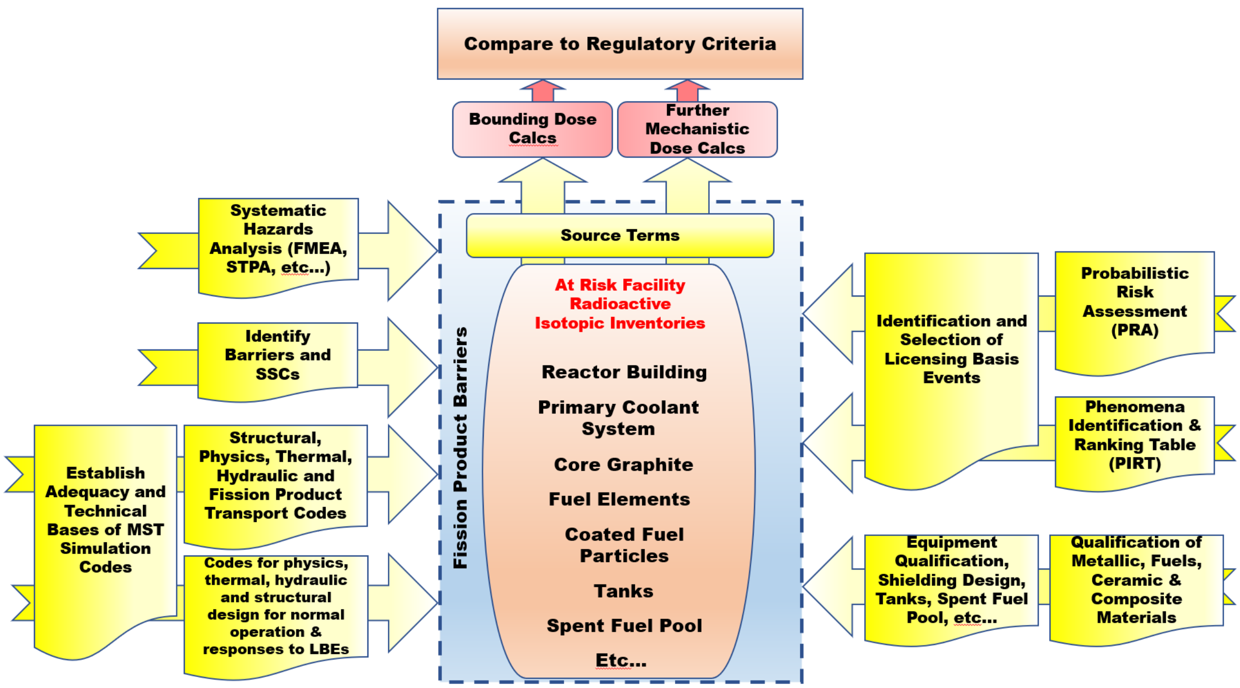


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).

Several factors need to be considered in the source term determination for non-LWR technologies. As these are defined and characterized, the influence of each on the calculated dose is established. This influence permits developing a target for each element in the source term calculation to meet the safety goals of the facility design. The development of these targets and the degree to which each element of the source term calculation must be characterized are addressed in the following iterative steps and Figure 3-2 and discussed in the subsequent sections in more details:

Step 1: Identify Regulatory Requirements

Identify the Site & EAB/LPZ radiological consequence regulatory criteria that ensure the health and safety of the public and protect the environment.

Step 2: Identify Reference Facility Design

Select the reference facility design and identify facility system failure modes and safety SSCs of these systems, or needed for these systems, during all foreseeable operating modes. Use a system hazard analysis (SHA) such as Failure Mode and Effects Analysis (FMEA) or System-Theoretic Process Analysis (STPA) as necessary.

Step 3: Define Initial Radionuclide Inventories

Determine equilibrium radionuclide inventories (or appropriate values if equilibrium conditions are not achieved for a particular plant design) in all plant systems (e.g., fuel, barrier 1, barrier 2, etc.) during normal steady-state operation.

Step 4. Perform Bounding Calculations

These bounding calculations are performed to determine the dose consequences of the releasing radionuclide inventories identified by the previous step for the "maximum credible accident." Demonstrate compliance with the established regulatory criteria.

a. If compliance is demonstrated with margins to the F-C targets or other performance measures, prepare the documentation and submit to the NRC for approval, and the process related to assessing offsite consequences may end. If the use of a conservative source term is not able to support the evaluations of design features and offsite consequences, proceed to the next step. Note that margins to F-C targets and assumptions related to SSCs serving to prevent or mitigate events may contribute to other design and licensing decisions such as SSC classification.

Step 5. Conduct SHA and Perform Simplified Calculations

Conduct a SHA (FMEA, STPA, or equivalent) to identify potential SSC failure modes that lead to radioactive releases, as well as to identify a spectrum of postulated LBEs. As described in NEI 18-04, these assessments also contribute to probabilistic risk assessments that are expected to support the design and licensing of advanced reactors.

Develop realistic assessment of the barriers being relied upon for evaluated design basis event (DBE) sequences and resultant inventory release fractions across barriers (Equation 1) based on this analysis. Consider the behavior of the barriers and determine dose consequence by using simplified methods.

If the dose calculations show compliance with established regulatory criteria and the transient and barrier-specific release fractions can be justified to the NRC, the process ends. Otherwise, consider performing more detailed dose calculations using NRC-approved codes and actual site meteorological data. If the calculated dose meets regulatory criteria with margin, prepare the documentation and submit to the NRC for approval, and the process ends. Using siting as an example, if the calculated dose exceeds 10 CFR 50.34(a)(1)(ii)(D) dose criteria, proceed to the next step. Developers may also define performance measures (e.g., lower dose goal than criteria given in regulation) based on design goals such as desiring more flexible siting options or a scalable EPZ.

Step 6. Consider Risk-informed System Design Changes

Consider a system redesign to include additional SSCs as identified by hazard analysis, which will either return to Step 3 or proceed to Step 7.

Step 7. Select Initial List of LBEs and Conduct PIRT

Carry out activities as described in NEI 18-04 to select an initial list of LBEs and to conduct PIRT (Phenomena Identification and Ranking Table) to identify important phenomena for LBEs.

Step 8. Establish Adequacy of MST Simulation Tools

Establish adequacy of MST simulation tools and develop testing programs if needed:

a. Identify and characterize factors and parameters (e.g., temperatures, pressures) affecting radionuclide generation and transport during possible event sequences for the subject reactor technology or nuclear facility.

b. As needed to support meeting the regulatory criteria, identify how well each factor is currently characterized to validate its target in establishing the source term and, where the current characterization is deficient, define the gaps between what is needed and what is known.

c. If needed, develop and complete analytic and testing programs to fill those gaps.

Step 9. Develop and Update PRA Model

Develop and update PRA models for the subject reactor or nuclear facility, which could receive input from Step 12.

Step 10. Identify or Revise the List of LBEs

Use the risk information obtained through the performance of all prior steps to identify or revise the list of LBEs.

Step 11. Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

Analyze and include external events unique to the site of the facility which can cause LBEs.

Step 12. Perform Source Term Modeling and Simulation for LBEs

Perform source term and dose modeling and simulation for the selected LBEs.

Step 13. Review LBEs List for Adequacy of Regulatory Acceptance

Develop a final list of LBEs. If the final list is not complete, go back to Step 6.

Step 14. Document Completion of Source Term Development

Prepare documentation for source term calculations and submit to the NRC for approval.

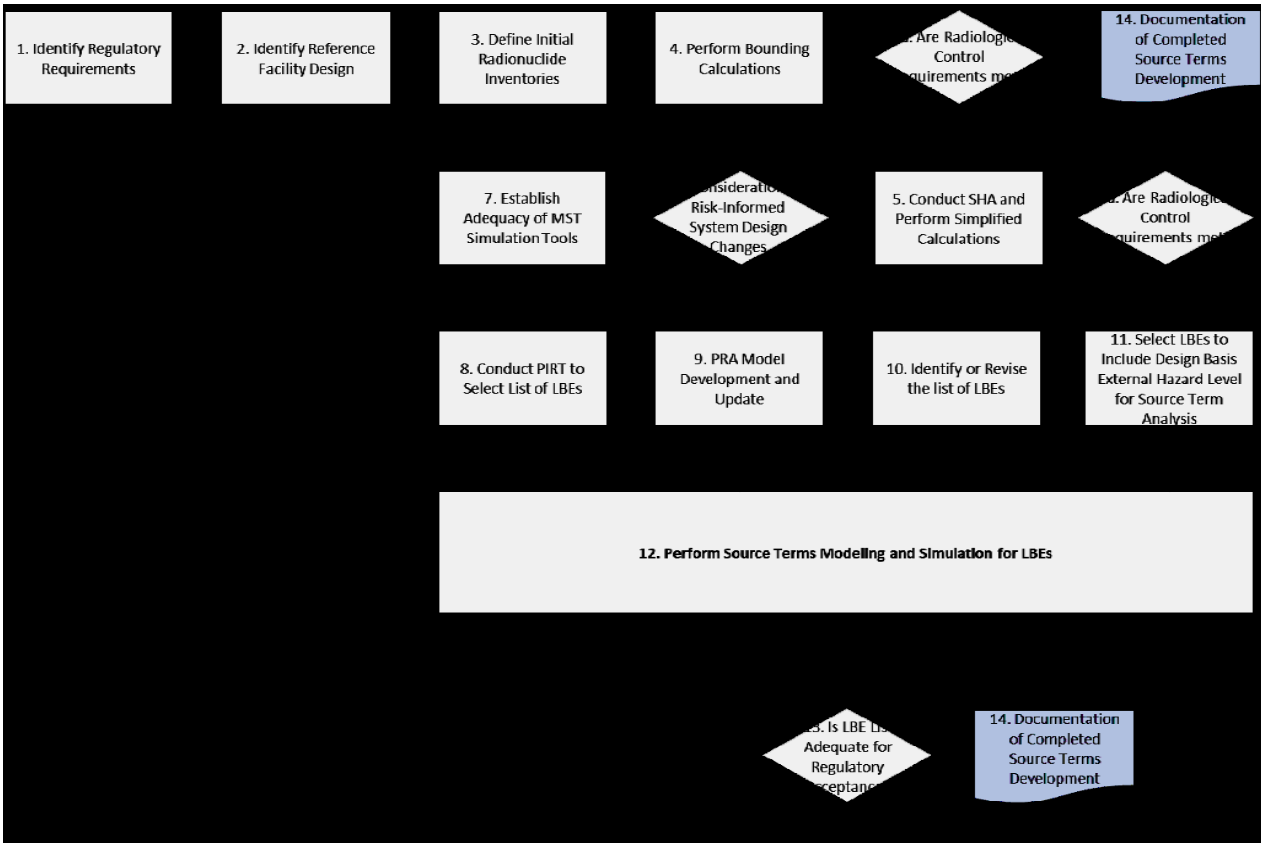


Figure 3-2 Technology-inclusive source terms determination methodology.

When referring to Figure 3-2, there are several pathway loops that can lead to completion of source terms development (Step 14). The first three pathways use a non-mechanistic or simplified mechanistic approach. One is to use initial bounding calculations from Step 4 to meet radiological control requirements. This is intended for facilities that have a small enough initial inventory of source terms to meet radiological control requirements upon a full release of the initial inventory. The second pathway can use the SHA performed in Step 5 to identify barriers and a maximum fractional release to perform a simplified mechanistic bounding analysis that would again meet radiological control requirements. A third pathway, which is still not a full MST approach, is to use the loop of redesign (Step 6) after failing Step 5a and then following through to Step 4a or Step 5a to its conclusion while meeting the radiological control requirements. If these pathways are not sufficient, a complete MST approach is desirable. Steps 6 through 13 are consistent with the MST process defined in NEI 18-04 for selecting and evaluating LBEs. The only exception is the addition of Step 8 to establish the adequacy of MST simulation tools. This step is necessary to ensure the MST simulation tools have acceptable level of pedigree in terms verification, validation, and uncertainty quantification.

3.1 Identify Regulatory Requirements That Require Radiological Source Term Information

Top-level radionuclide control requirements will be established for advanced nuclear facilities using existing regulatory requirements and design goals established by developers. The objective of setting the top-level radionuclide control requirements is to limit the calculated dose under all LBEs so that regulatory requirements for the protection of the health and safety of the plant workers, the public, and the environment are met. Limits on radionuclide release from the reactor building that are consistent with these top-level radionuclide control requirements are needed to establish the target values for all of the barriers to radionuclide release and ultimately to establish allowable in-service fuel failure and as-manufactured fuel quality requirements. The key top-level radionuclide control requirements expected to be imposed for the advanced nuclear reactors or nuclear facilities are listed in Table 3-1. The top-level radionuclide control requirements are based on established regulatory practice, e.g. NRC regulations in 10 CFR 20 [3], 10 CFR 30 [27], 10 CFR 50 [4], 10 CFR 52 [28], 40 CFR 190 [29] and EPA (Environmental Protection Agency) protective action guides (PAGs) [30]. It is noted that 10 CFR 20 limits the radiation doses from licensed operation to individual members of the public. Although not technically applicable to non-LWR designs, 10 CFR 50 Appendix I identifies design objectives for release from LWRs during normal operation to be as low as reasonably achievable. Both of these regulations are concerned with the cumulative dose acquired annually, rather than during a single event. Section 50.34 requires an applicant for a license for a power reactor permit or license to demonstrate that doses at the EAB and the outer boundary of the LPZ from hypothetical accidents (i.e., per event) will meet specified criteria. Part 100 refers to the same dose criteria in 10 CFR 50.34 for determining site suitability. The development of source terms for purposes other than determining an offsite dose may have additional or different regulatory requirements. For example, the environmental qualification of equipment is done per the requirements of 10 CFR 50.49, which does not have specific regulatory dose criteria.

Table 3-1 Top-Level Regulatory Requirements

Top-Level Regulatory Requirements		**Comment**
1	10 CFR 30, Schedule C	Emergency plan
2	10 CFR 50.34(a)(1)(ii)(D) TEDE ≤ 25 rem at EAB over worst two-hour dose period TEDE ≤ 25 rem at outer edge of low population zone (LPZ) for the duration of the passage of the plume	Facility siting Offsite dose criteria
3	10 CFR 50, Appendix I, LWR Design Objectives for Radionuclides in Plant Effluents, dose to individual in unrestricted area: Whole Body Dose ≤ 5 mrem/yr Dose to any organ ≤ 15 mrem/yr	Plant effluents
4	10 CFR 20 Subpart C Occupational Dose Limits: Total effective dose equivalent (TEDE) <5 rem/yr Organ Dose ≤ 50 rem(/yr)	Standards for occupational protection
5	10 CFR 20 Subpart D Public Dose Limits: Annual TEDE ≤ 0.1 rem Hourly External Dose ≤ 0.002 rem	Standards for public protection

6	40 CFR 190 Subpart B Environmental Standards for the Uranium Fuel Cycle, (LWRs), normal operations, annual dose equivalent: Whole Body ≤ 25 mrem Thyroid Dose ≤ 75 mrem Organ Dose ≤ 25 mrem	Standards for fuel cycle
7	10 CFR 52.47 Offsite Dose Criteria for LBEs, standard design certification: TEDE ≤ 25 rem for 2 hours at the EAB TEDE ≤ 25 rem for duration of passage of plume at the LPZ boundary	Offsite dose criteria*
8	EPA PAGs for Radioactive Release for Public Sheltering & Evacuation (EPA 2017): TEDE over four days ≤ 1 rem Thyroid Dose ≤ 5 rem	Public shelter & evacuation
9	NRC Safety Goal Policy Statement (NRC 1986)	Safety goal

* It is noted that the same offsite dose criteria for LBEs can also be found in 10 CFR52.17 for early site permit, 10 CFR52.79 for combined license, 10 CFR 52.137 for standard design approval, and 10 CFR 52.157 for manufacturing license.

3.2 Identify Reference Facility Design

This step is important because focusing on the specifics of the advanced nuclear design provides the interconnection of all systems with the methodic analysis for the determination of source terms. The subject reference nuclear reactor and facility design is established by the developer when ready for evaluation. The design parameters and features, such as nuclear fuel, reactor core, heat transport systems, and engineered safety features within barrier 1; systems and engineered safety features within barrier 2; etc., are identified (see Figure 1.1). The facility operating modes such as online refueling or shutdown refueling, normal operations, events such as anticipated operational occurrences (AOOs), DBEs and beyond design basis events (BDBEs), and the DBAs are described. The definitions of AOOs, DBEs, BDBEs and DBAs are consistent with those found in NEI 18-04 [22].

3.3 Define Initial Radionuclide Inventories

The initial inventories (in Equation 1) of the radionuclides important for the calculations of offsite

consequences at accident initiation are calculated using NRC accepted computer codes (e.g., the SCALE ORIGEN module for isotope generation and depletion) and methods. The initial inventories calculated are considered as the Site Radionuclide Inventories at Risk (SRIR) for release, and they represent some maximum quantity of radionuclides present or reasonably anticipated for the process or structure being analyzed. Different SRIRs may be assigned for different accidents as it is only necessary to define material in those discrete physical locations that are exposed to a given stress. The initial calculation of radionuclide inventories should include the radionuclides in fuel, and system information and depletion methods are subsequently used to calculate inventories resulting from all radionuclides residing in all systems barriers (i.e., Figure 1-2: barrier 1, barrier 2, etc.) due to an activation and leakage of the initial core inventory. For the generation of fission products in fuel, assumptions on fuel, core design, and management (e.g., operating cycle length, burnup limits, etc.) and the type of inventory (e.g., equilibrium nominal end of life) should be described. The use of conservative modeling assumptions or treatment of uncertainties in the initial inventories should be described. Initial radionuclide inventories are given by isotope either as total activity (for solid fuel) or activity concentration (in fluid).

3.4 Perform Bounding Calculations to Estimate Consequence of Site Radionuclides Inventory at Risk for Release

A bounding analysis employs assumptions that are meant to produce the worst-case consequence resulting from a “maximum credible accident” for a given facility or system of that facility. It is also a starting point analysis for a facility to illustrate the potential, or lack thereof, level of radioactive hazard associated with a facility. A possible resource for such an analysis is 10 CFR 30. Schedule C of 10 CFR 30 contains a list of release fractions and maximum release limits of various isotopes that would avoid the need for public evacuation plan. The release fractions of Schedule C are meant to be the worst-case release for facilities that handle or produce radioactive byproduct material. These release fractions are the result of accident analyses, operation experience, or known physics limitations, for example note the “Nuclear Fuel Cycle Accident Analysis Handbook” (NUREG-1320 [31]) or “Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities” (DOE-HDBK-3010-94 [32]). In addition, the dose calculation employs the assumption that annual averaged meteorological weather data is not available and therefore conservative meteorological weather conditions are assumed of Pasquill-Gifford Type “F” plume stability for a wind velocity of 1 m/s, see the “Technical Basis for Regulatory Guide for 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"” (NUREG/CR-2260 [33]).

Proceeding with a bounding analysis for the given facility requires that, after the initial radionuclides inventories at risk are determined, bounding calculations that estimate the consequence for release are performed by calculating the product of the release fraction listed in 10 CFR 30 Schedule C for a particular radioactive isotope times the inventory at risk. If this product is equal to or below the release limit for that isotope as listed in 10 CFR 30 Schedule C, an emergency plan is not needed for responding to a release of radioactive material for facilities applicable to 10 CFR 30. 10 CFR 30 also contains a formula for multiple isotope releases, which is the sum of the ratios of actual release to the release limit. If this sum is less than or equal to one, an emergency plan is not needed for responding to a release of radioactive material. The information in 10 CFR 30 Schedule C is based on showing that the consequences of the release would be less than one rem TEDE offsite. Similar analyses must be performed for comparison to other radiological criteria listed in Table 3-1.

If compliance has been demonstrated, prepare the documentation, including a description of methods, assumptions, and consideration of uncertainty, and submit to the NRC for approval, and the source term determination portion of the design and licensing process ends here, provided that the release fractions used can be justified as applicable to your facility and that the calculated margins to radiological limits have been achieved by the facilities design. Otherwise, proceed to the next step.

3.5 Conduct SHA to Identify Potential Failure Modes and Determine Dose Consequence Using Simplified Methods

In this step, a SHA equivalent to a FMEA [34] or a STPA [35] is conducted to identify potential failure modes that could lead to source terms. The intent is to utilize SHA to identify all release paths described in Figure 1-2 and Equation 1. This information has a two-fold purpose: one is used to take credit for SSCs beyond those credited in the bounding calculations performed in Section 3.4 while providing a simplified source term, and the second purpose is to identify SSCs and barrier penetration pathways for further steps in the deterministic or mechanistic process. The use of a SHA or similar technique is consistent with the discussions in NEI 18-04 [22] on developing a technically sound understanding of the potential failure modes of the reactor concept, how the plant would respond to such failure modes, and how protective strategies can be incorporated into formulating the safety design approach. The incorporation of safety analysis methods appropriate to early stages of design, such as FMEA and process hazard analysis, provide early stage evaluations that are systematic, reproducible, and as complete as the current stage of design permits and support the development of the PRA (see Step 3.9).

A SHA will identify the SSCs and barrier penetration pathways and to some extent the effects of failure in preparation of PRA, PIRT, and modeling analyses.

SHA processes gather system experts and documentation to answer questions about the design pertaining to barriers to radioisotope inventory transport during normal and off-normal operations. Questions answered include:

- What is the failure mode?
- What are the interactions that occur due to the event?
    - What do the interactions cause?
- How likely is a failure to happen?
- What is the effect of the failure on the system?
- What is the outcome in transport of radioisotope inventory release fractions?

The following factors should be considered for the SHA derived release fractions: the SSC damage ratios (fraction of the materials at risk actually impacted by the accident generated conditions), leak path factors (fraction of the radionuclides in the aerosol transported through some confinement barrier that are deposited in a filtration mechanism), airborne release fraction (or airborne release rate for continuous release) (airborne release rate is a coefficient used to estimate the amount of a radioactive material suspended in air as an aerosol and thus available for transport due to a physical stress from a specific accident), and respirable fraction (fraction of airborne radionuclides as particles that can be taken up through air inhaled by the human respiratory system. Particulate releases from LWRs are commonly assumed to include particles with 10-µm Aerodynamic Equivalent Diameter or less).

If SHA can identify and quantify the effectiveness of SSCs and barriers to radioisotope release, it can be used to define release fractions for a spectrum of postulated DBEs. Once the bounding release fractions for the at-risk radionuclides inventory have been determined, a mechanistic source term analysis can be performed using simplified methods. These simplified methods are described in Simplified Approach for Scoping Assessment of Non-LWR Source Terms (SAND2020-0402 [23]). Subsequently, the resulting dose consequence of these source terms can be estimated by using other NRC accepted computer codes and methods.

If the dose consequence analyses demonstrate compliance with radiological criteria listed in Table 3-1, an argument can be made that the source terms do not need to be developed further. The process then moves to the documentation phase, which should include a description of methods, assumptions, and consideration of uncertainty. Otherwise, the process proceeds to using the SHA information attained to complete the subsequent steps.

3.6 Consideration of Risk-Informed System Redesign

As pointed out in NEI 18-04 [22], the design development is performed in phases and often includes a preconceptual, conceptual, preliminary, and final design phase and may include iterations within phases. The subsequent steps may be repeated for each design phase or iteration until the list of LBEs becomes stable and is finalized. If the system as designed is not adequate to meet the radiological safety control requirements of a bounding or mechanistic case, consider a system redesign to include strengthened barriers and/or SSCs as identified by SHA, PIRT, or PRA. During the earlier phases prior to the final design phase, using simplified source term methods (e.g., SAND2020-0402 [23]) to evaluate the release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance can efficiently identify the design features that are most important to mitigate different classes of accident scenarios. The mechanistic source term methodology described in the subsequent steps play a more important role in the evaluation of the mitigation strategies during the final design phase.

System redesign, using a risk-informed approach as shown in "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (RG-1.200 [36]), can direct efforts towards the greatest benefit for meeting radiological regulatory criteria. PRAs used in risk-informed redesign activities may vary in scope and level of detail within each phase. The PRA needs to be maintained and upgraded, where necessary, to ensure it represents the actual state of the design phase.

3.7 Select Initial List of LBEs and Conduct PIRT

As noted in regulatory guide RG 1.233 [7], established methods for addressing radiological source terms for LWRs have limited applicability to non-LWR designs, and mechanistic source term analysis may be used to estimate radiological consequences for such designs. Toward that end, it is necessary to select the initial list of LBEs to develop the basic elements of the safety analysis including mechanistic source term analysis during design development. The initial list of LBEs is to be selected using a deterministic approach based on engineering judgment. This approach has been used for licensing operating LWRs and involves no use of PRA information and insights. NEI 18-04 [22] has a detailed description on how to select the initial list of LBEs.

The MST methodology for the evaluation of the initial list of LBEs will need to meet the three provisions outlined in Section 3.8 from SECY-93-092 [8]. SECY-93-092 further outlines that "The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design." The PIRT process can be used to ensure that these conditions are met. The PIRT process is a systematic way of identifying safety-relevant and safety-significant phenomena and ranking the importance and knowledge level associated with these phenomena for the LBEs. This ranking is ideal for advanced reactors in the conceptual design phase and for assessing through a source terms PIRT whether the transport of fission products can be adequately modeled based on present knowledge levels, as required by the above MST provisions.

The PIRT process consists of nine steps:

1. Identify issues
2. Identify specific objectives
3. Define hardware and scenarios
4. Define evaluation criteria
5. Identify current knowledge base
6. Identify phenomena
7. Develop importance ranking
8. Define knowledge level
9. Develop documentation

During the PIRT process, a comprehensive list of phenomena relevant to safety for potential hardware failure models and accident scenarios is developed by a panel of experts. After that, the importance of the phenomena is ranked either high, medium, or low relative to certain evaluation criteria. The process has previously been applied to understand radionuclide transport in certain advanced reactor systems, for example see "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)" (NUREG/CR-6944 [37]), and is generalized in a technology-inclusive way in what follows. An example outcome of the process, applicable to mechanistic source term analysis, is given in Table 3-2.

Table 3-2 PIRT - Identify Issues.

#	Phenomenon	Importance	Rationale	Knowledge Level	Rationale	Model Status
1	Transport phenomenon A	High	Primary barrier for radionuclide transport	Low	Lack of, or uncertain, experimental data	Major need
2	Transport phenomenon B	Medium	Minor barrier for radionuclide transport	Medium	Some experimental data available	Minor need
3	Transport phenomenon C	Low	No credit taken for barrier C in source term analysis	High	Well characterized experimentall y	Adequate

Table 3-2 also includes a column titled “model status,” which may be used as a part of the process to assess the adequacy of models generally or certain codes in particular to perform mechanistic source term calculations for a given advanced reactor type. Here the status is classified as a “Major need,” “Minor need,” or “Adequate.” A status of “Adequate” would refer to models that are well verified and widely accepted, or that such models have been implemented, verified, and validated in the computer code in question. A status of “Minor need” indicates models that might be improved if informed by some additional experimental data, or such models that need minor modification within a code or are straightforward to implement. A “Major need” indicates models that are speculative in nature, not well informed by experimental data, or highly uncertain, or code implementations that lack such a model entirely in addition to its verification and validation.

To the extent that each phenomenon listed in the table corresponds to transport across a barrier, each is associated with a release fraction across that barrier, as in Equation (1); conservatism in a given transport step (as in the third example in Table 3-2) would correspond to a release fraction of one for that step.

3.8 Establish Adequacy of Mechanistic Source Term (MST) Simulation Tools and Develop Analytic and Testing Programs

The adequacy of the mechanistic source term simulation tools will be assessed in this step to take specific account of the unique features of each reactor type. The use of design-specific and event- or scenario-specific mechanistic source terms can be justified by having sufficient experimental data to confirm the source term (e.g., quantity and form of radionuclides, timing of release) and accounting for uncertainties in the source term determination (e.g., use 95% confidence level). The assessment of the computer codes involves verification, validation, and uncertainty quantification. The factors affecting radionuclide generation and transport for the subject reactor technology or nuclear facility will be identified and characterized. As needed to support meeting the regulatory criteria, identify how well each factor is currently characterized to validate its target in establishing the source term and, where the current characterization is deficient, define the gaps between what is needed and what is known. If needed, develop analytic and testing programs to fill those gaps and determine appropriate programmatic controls (e.g., inspections and surveillances) that may be needed during plant operations. The adequacy of the MST simulation tools can be established according to the provisions specified in “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements” (SECY-93-092 [8]), which states that source terms should be based upon mechanistic analysis provided that:

- The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through research, development, and testing programs to provide adequate confidence in the mechanistic approach.
- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.
- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.

Since it may take a long time to complete the testing programs, this step will proceed in parallel with the evolution of the design of an advanced reactor. The completed analytic and testing programs for the source terms would have filled the technical gaps identified between what is needed and what is known. The radionuclide generation and transport phenomena are more fully characterized and understood. The MST computer codes will be updated and validated with the newly acquired data and knowledge.

One important outcome from the completion of the analytic and testing programs is the identification, evaluation, and management of uncertainties. Uncertainties need to be addressed in the calculation of both frequencies and consequences of the event sequences. Since the sequences include rare events and event combinations postulated to occur in complex systems for which there may be limited experience, the consideration of uncertainties is a vital part of understanding and determining the extent of the risk. A range of uncertainties needs to be considered and quantified in the MST calculations, including parameter uncertainty associated with the basic data and model uncertainty associated with analytical physical models and success criteria in the PRA, driven by modeling choices and by the state of knowledge about the new designs and the interactions of human operators and maintenance personnel with these systems. Sensitivity studies should be considered as an important means for examining the impacts of modeling uncertainties. All identified and quantified uncertainties (aleatory and epistemic) should be included in the MST calculations.

NEI 18-04 [22] describes the consideration of uncertainties, including from the MST, in several places, including as follows:

> The PRA's quantification of both frequencies and consequences should address uncertainties, especially those associated with the potential occurrence of rare events. The quantification of frequencies and consequences of event sequences, and the associated quantification of uncertainties, provides an objective means of comparing the likelihood and consequence of different scenarios against the F-C Target....

3.9 PRA Model Development and Update

When using the approach described in NEI 18-04 [22], PRA should be performed to model LBEs in a probabilistic manner. PRA standards, such as ASME/ANS-RA-S, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", ASME/ANS RA-S-1.2, "Severe Accident Progression and Radiological Release (Level 2) PRA Methodology to Support Nuclear Installation Applications", and ASME/ANS RA-S-1.3, "Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications," detail the processes for developing a design-specific PRA. Also, consider the use of the Non-LWR PRA Standard that is currently in development. The ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) issued "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants", ASME/ANS RA-S-1.4-2013, for trial use in 2013. In "Non-Light Water Reactor Implementation Action Plan," SECY 19-0009: Enclosure 1 [38], it is noted that the use of this trial standard by national and international organizations and feedback to the JCNRM will lead to a final draft. The PRA is not just the event tree/fault tree logic model. The PRA consists of a group of analyses which informs the logic model, which in turn informs the consequence modeling.

PRA is iterative with modeling and simulation. PRA both informs the modeling software of the potential LBE sequences and is in turn informed by the outcome of performance tools that validate and/or modify the PRA sequences discussed below. Any design change can affect the PRA, and the PRA should be used to represent the current state of the design in a probabilistic manner for risk-informed decisions.

PRA consists of two over-arching types of analyses, "static" PRA and "dynamic" PRA. Static PRA is solely based on the probability of events occurring in sequences to determine an outcome. Dynamic PRA utilizes the simulation of both probabilistic information and physics-based information.

Static PRA is used for many design and regulatory decisions. Static PRA starts with the probability of basic events occurring based on published or developed performance data. These consist of the frequency of an initiating event, such as loss of offsite power, failures of a component to perform its intended function on demand or over a period of operational time, or failures of operators to perform a specific task within an allotted time. The basic events are placed in logic trees called fault trees for each safety system. Event trees are started by an initiating event and then questioning the safety system fault trees to determine what the likelihood of a specific outcome from an initiating event is. A specific path through the logic trees to an end state provides a probability of the outcome and is called a sequence. For LWRs, all sequences that lead to an end state of core damage (CD) are gathered to calculate the core damage frequency (CDF), which is used in regulatory decisions. A PRA can extend beyond the first level of CD to describe the physical state of the plant and be used to determine the radiological consequences through dose-consequence software programs, such as MACCS. Further information can be gained from static PRAs by utilizing importance measures to determine the most important components in the system to prevent CD and radiological release. Action can be taken to improve the CDF or the state of the plant if a CD were to occur by addressing the highly important components through improvement in design such as increasing system redundancy. The CD and CDF are not descriptive of all technologies, where the "core" can be a very diverse term. By using the definition that CD allows radionuclide inventory to penetrate the first barrier of fuel cladding, a technology-inclusive way of describing the undesirable outcomes of CD and CDF is undesirable release (UR) and undesirable release frequency (URF) of radionuclide inventory from the defined barrier.

Dynamic PRA utilizes physics-based and probability-based modeling to determine the outcome of an initiating event through one sequence. While static PRA is required for regulatory decisions, including licensing, dynamic PRA is a powerful tool in determining the validity of sequence end states. Dynamic PRA can be performed through the use of physics-based performance tools and simulation. The validation of the outcome of sequences through the event trees is one function of dynamic PRA.

PRA is developed in three levels, as is outlined in the ASME/ANS RA-S series standards. It is recommended to use the most recent edition of "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants" (currently ASME/ANS RA-S-1.4-2013) where there is any conflict between the LWR and non-LWR standard or issues related to the use of offsite consequences in decision-making versus surrogate criteria such as CDF.

The first level of a PRA models the events that cause damage to the inner-most barriers containing the fuel. Traditionally, this has been called core damage; however, the first barriers to containment of the fuel in some designs can differ from what is commonly thought of as a "core." In molten-salt reactors for instance, the fuel is contained in piping, and the "core" might be considered the fuel and piping combination. In other designs, TRISO spheres provide the first barrier within the fuel design itself, but the "core" can be considered the first containment barrier outside of the collection or matrix of TRISO pellets. For consistency, we will refer to the fuel and the first containment barrier as the core and to the first barrier breach as core damage.

The second level of a PRA models the physical state of the facility once a CD event has occurred. This logically turns on and off safety systems based on the event and informs the further capabilities of barriers, leading to consequence modeling.

The third level of a PRA models the consequence, or dose, for evaluation of EAB/LPZ radiological limits and/or the F-C target. This is a level where results can be listed as end states within the PRA ET/FT model, but it is determined by a consequence dose calculation program that utilizes radionuclide transport and dosimetry algorithms, such as MACCS. Level three PRA is informed by the source terms released from the final barrier to the atmosphere. This source term release is determined by performance tools, such as accident progression and source term programs like MELCOR.

The Non-LWR PRA Standard discusses many applications outside of the LWR PRA standard. The Non-LWR PRA Standard's scope also covers many areas outside of those found in other standards and should be used if there are any conflicts between standards. The scope of the Non-LWR PRA Standard (from ASME/ANS RA-S-1.4-2013):

a) Different sources of radioactive material both within and outside the reactor core but within the boundaries of the plant whose risks are to be determined in the PRA scope selected by the user. The technical requirements in this trial-use version of the standard are limited to sources of radioactive material within the reactor coolant system (RCS) pressure boundary (RCPB) (*and just within the RCS for a pool reactor*). Technical requirements for other sources of radioactive material such as the spent fuel system are deferred to future editions (*of the Non-LWR PRA Standard*).

b) Different plant operating states (POSs) including various levels of power operation and shutdown modes.

c) Initiating events caused by internal hazards, such as internal events, internal fires, and internal floods, and external hazards such as seismic events, high winds, and external flooding.

d) Different event sequence end states, including core or plant damage states (PDSs), and release categories that are sufficient to characterize mechanistic source terms, including releases from event sequences involving two or more reactor units or modules for PRAs on multireactor or multiunit plants.

e) Evaluation of different risk metrics including the frequencies of modeled core and PDSs, release categories, risks of off-site radiological exposures and health effects, and the integrated risk of the multiunit plant if that is within the selected PRA scope. The risk metrics supported by this standard are established metrics used in existing light water reactor (LWR) Level 3 PRAs such as frequency of radiological consequences (e.g., dose, health effects) that are inherently technology neutral. Surrogate risk metrics used in LWR PRAs such as core damage frequency and large early release frequency are not used as they may not be applicable to non-LWR PRAs.

f) Quantification of the event sequence frequencies, mechanistic source terms, off-site radiological consequences, risk metrics, and associated uncertainties, and using this information in a manner consistent with the scope and applications PRA.

The use of PRA in the development of a design determines the metrics of the current design (event sequence frequencies, iterative development of mechanistic source terms, offsite radiological consequences, risk metrics, and associated uncertainties) from the source terms that are released and provides a platform for quantifying the effects of modifications on the design for comparison to prior metrics.

3.10 Identify or Revise the List of LBEs

The plant licensing basis is, to a large extent, dependent upon risk information. The risk information obtained from the updated PRA models needs to be fed back into the licensing analysis to ensure that the plant licensing basis remains valid. This would entail updating the list of LBEs initially selected in Step 3.7 with the risk insights obtained from Step 3.9. When the updated risk information indicates that a change in the plant licensing basis is warranted, the appropriate changes will be made to update the list of LBEs.

The selection of accidents to be considered in the identification of source terms plays a lead role in the use of mechanistic source terms, because it defines the specific scenarios and associated release mechanisms used to assess such source terms. In Section 3.7, a methodology for the identification of an initial list of LBEs for non-LWR technology has been presented; those scenarios might include:

- Anticipated Operational Occurrences (AOOs)
- Design Basis Events (DBEs)
- Beyond Design Basis Events (BDBEs)
- Design Basis Accidents (DBAs)

To come up with a robust and inclusive list of LBEs for any advanced reactor technology, a systematic approach is required. LBEs are defined as the events derived from the reactor technology and plant design of interest that are used to derive design-specific performance requirements for structures, systems, and components and are generally inferred from the licensing process. Considering that the selection of such events needs to be performed, potentially, for new technologies, a combination of deterministic and probabilistic methods should be used for both the identification and consequence assessment of such events.

The selection process needs to be considered as an integral part of the overall design process and, consequently, it must be "re-iterated" since its selection (and outcomes) informs the design requirements of safety-related and non-safety-related systems and components. Once an initial set of LBEs is identified, the design can be refined to reduce the likelihood or associated risk of a specific LBE.

The process can be exemplified in multiple stages:

1. A deterministic approach is used to select an initial event set providing a starting point for the assessment of the source terms.
2. The LBEs are updated every time the design and analysis evolve.
3. A review of the LBEs is performed at the end of the design phase to evaluate conservatisms in the selected events.

3.11 Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

External events are chosen deterministically on a basis consistent with that used for LWRs (SECY-19-0117: Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors) [39]. A set of design basis external hazard levels (DBEHLs) will be selected to form an important part of the design and licensing basis. This will determine the design basis seismic events and other external events that the safety-related SSCs will be required to withstand. When supported by available methods, data, design, site information, and supporting guides and standards, these DBEHLs will be informed by a probabilistic external hazards analysis and will be included in the PRA after the design features that are incorporated to withstand these hazards are defined. Other external hazards not supported by a probabilistic hazard analysis will be covered by DBEHLs that are determined using traditional deterministic methods.

3.12 Perform LBEs Source Term Modeling and Simulation

As previously mentioned, the selection of the LBEs to include in the source term calculations is an iterative process that needs to be repeated in any stage of the design (or when substantial changes to the design are made).

The source term assessment needs to characterize the generation, release, transport, and retention of fission product and activation radionuclides. The modeling of such phenomena requires identification of the "barriers" for the technology of interest. The "barriers" provide mechanisms for the retention of the fission products during normal operation and accident conditions. The process for the development of modeling and simulation tools for non-LWR applications is similar to LWR applications. Once the LBEs are selected and the modeling tools are available, the actual simulation effort can be initiated. These requirements are described in the following subsections.

3.12.1 Requirements for Source Term Modeling and Simulation

Since the publication of "NRC Non-Light Water Reactor (non-LWR) Vision and Strategy – Staff Report: Near-Term Implementation Action Plans," November 2016 (ADAMS Accession No. ML16334A495) [1], there has been dialogue between NRC staff, ACRS, DOE, and industry representatives on computer codes and tools to perform source term modeling and simulation for non-LWRs.

The NRC plan was presented to ACRS on May 1, 2019 (ADAMS Accession No. ML19143A120 [40]) and October 3, 2019 to discuss the NRC staff's ongoing code development to support independent analysis for licensing of non-LWR designs. In its letter of November 4, 2019 (ADAMS Accession No. ML19302F015 [41]), ACRS emphasized that, ideally, the tools for staff confirmatory analysis should be as independent as practical, validated, understood by the staff, and usable on the staff's computer resources. The ACRS stated that the staff also needed to become sufficiently familiar with applicants' codes to support timely reviews of submitted analyses. The ACRS stated that four principles should underlie the strategy: simplicity, completeness, working the problem backwards from the source term, and scaling down the level of effort of licensing review proportionately as the hazard decreases. The staff likewise advocates the strategies underlying these principles.

The staff's source term evaluation model for non-LWR applications is shown in Figure 3-3. This model is technology-inclusive because it relies on the same codes with the suite of physics models needed for the different non-LWR technologies. A detailed description of these codes and the development process, including identification of technical gaps, is provided in NRC's "Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis" (ADAMS Accession No. ML20030A178 [42]).

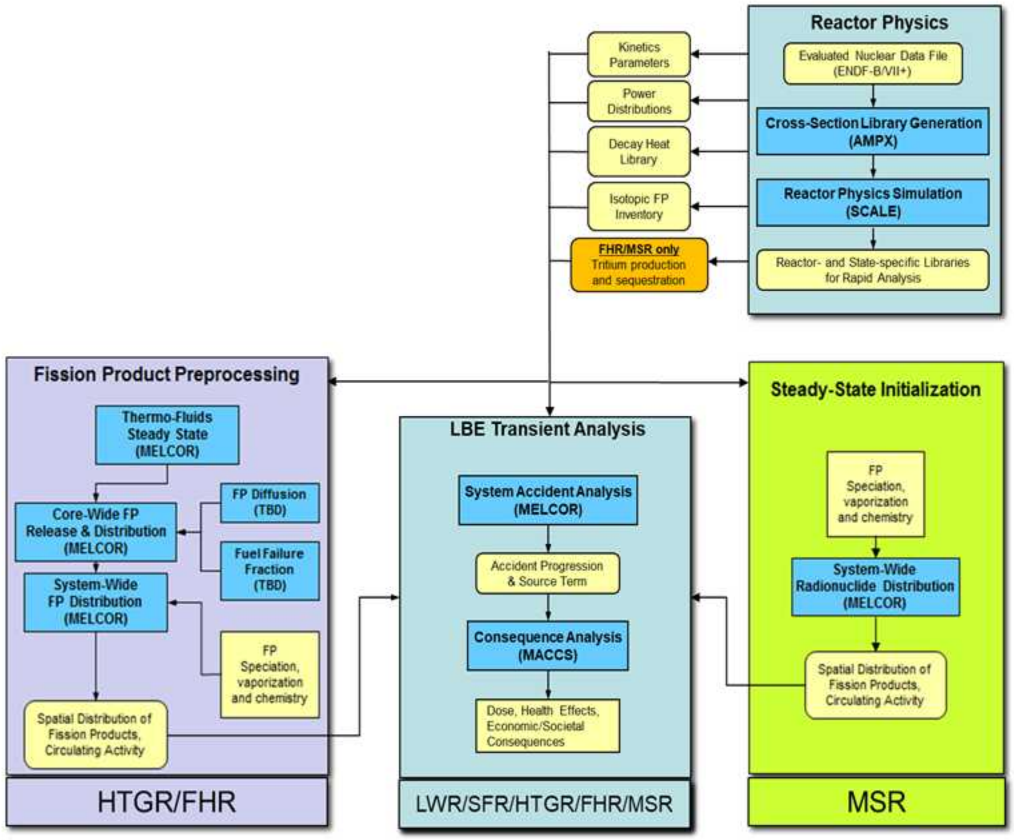


Figure 3-3 NRC evaluation model plan for source term characterization.

In 2020, the NRC began analysis of severe accident progression and source term for three representative advanced reactor designs. This effort is focusing on severe accident phenomenology and source term development and was presented at an advanced reactor stakeholder meeting on February 20, 2020 (ADAMS Accession No. ML20040E155 [43]). The three designs, which have publicly available data, are the following: (1) an HTGR, (2) a liquid-metal-cooled heat pipe reactor plant model (e.g., Los Alamos National Laboratory MegaPower reactor), and (3) a molten-salt-cooled pebble bed reactor plant model (e.g., University of California-Berkeley's Mark I Pebble Bed Fluoride-Salt-Cooled High-Temperature Reactor). In the first phase of this effort, MELCOR is being used to demonstrate how beyond design basis accident progression and source terms can be characterized for the selected three non-LWR design concepts. In the second phase, the MELCOR study results will be used to inform NRC staff, promoting the knowledge and insights needed to:

- Understand beyond DBEs for non-LWR technologies
- Develop guidance to support staff review of non-LWR applications in a timely and efficient manner.

In the final phase of this effort, workshops will be held to inform stakeholders on the staff's approach to perform independent source term analysis for the three representative non-LWR designs to promote dialogue between NRC and stakeholders. The intent of these workshops is to provide sufficient information to reduce uncertainty in the review process for non-LWR vendors developing design-specific source terms.

ACRS was briefed by the U.S. Department of Energy (DOE) concerning the capabilities of DOE computer codes and by industry representatives. The DOE presentation to the ACRS on August 21, 2018

(ADAMS Accession No. ML18254A164 [44]) outlined the DOE strategy for advanced (non-LWR) reactor safety analysis and involved various areas, including neutronics analysis capabilities, fuels modeling capabilities, thermal-hydraulic/system analysis, and source term assessment codes. For the source term analysis, an example involving application of DOE codes for a liquid-metal reactor application is shown in Figure 3-4.

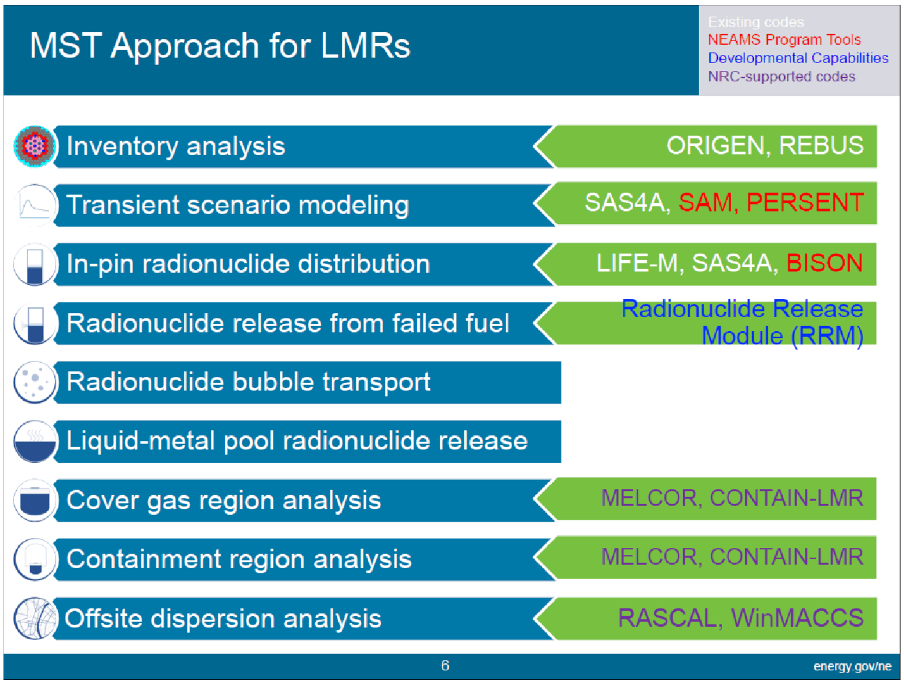
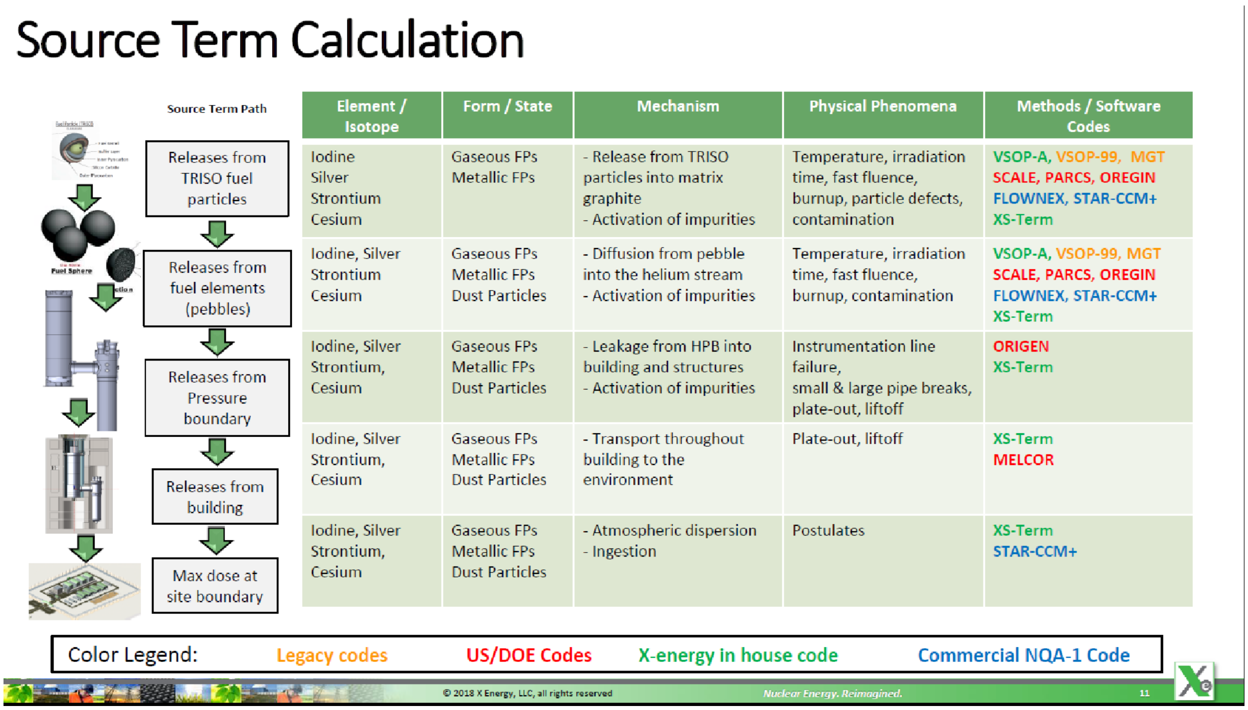


Figure 3-4 DOE code strategy liquid-metal reactor example for source term characterization.

ACRS was briefed on November 16, 2018 (Transcript at ADAMS Accession No. ML18340A016 [45]) by industry representatives working in MSR, SFR, and HTGR source term methodology. The vendors were engaged in efforts to characterize the source term, due to its importance in the safety analysis. The degree of computer code development and technical approach by different vendors varied.

As an example of a vendor's approach to characterize source term, Figure 3-5 below (Page 394 of ACRS Transcript) shows that the vendor X-Energy is applying a combination of in-house developed codes, such as XSTERM, and NRC codes such as SCALE and MELCOR. (The codes labeled in the Figure 3-5 as "US/DOE" are NRC codes that are being developed by the DOE national laboratories and the University of Michigan for NRC staff independent analysis).



Source Term Calculation

Element / Isotope	Form / State	Mechanism	Physical Phenomena	Methods / Software Codes
Iodine Silver Strontium Cesium	Gaseous FPs Metallic FPs	- Release from TRISO particles into matrix graphite - Activation of impurities	Temperature, irradiation time, fast fluence, burnup, particle defects, contamination	VSOP-A, VSOP-99, MGT SCALE, PARCS, OREGIN FLOWNEX, STAR-CCM+ XS-Term
Iodine, Silver Strontium Cesium	Gaseous FPs Metallic FPs Dust Particles	- Diffusion from pebble into the helium stream - Activation of impurities	Temperature, irradiation time, fast fluence, burnup, contamination	VSOP-A, VSOP-99, MGT SCALE, PARCS, OREGIN FLOWNEX, STAR-CCM+ XS-Term
Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Leakage from HPB into building and structures - Activation of impurities	Instrumentation line failure, small & large pipe breaks, plate-out, liftoff	ORIGEN XS-Term
Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Transport throughout building to the environment	Plate-out, liftoff	XS-Term MELCOR
Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Atmospheric dispersion - Ingestion	Postulates	XS-Term STAR-CCM+

Figure 3-5 X-Energy plan for source term characterization.

In general, as shown in the discussion above, the prediction of source term often involves the use of multiple codes that “answer” to different functional requirements:

- Reactor Physics Computer Models:
    - Calculate radionuclide inventories and power distributions in the design.
- Fuel Performance Computer Models:
    - Calculate thermal and stress histories for fuel and identify fuel failure and radionuclide release.
- System Analysis Computer Models:
    - Calculate the progression of accident and radionuclide transport.
    - Requires boundary conditions from fuel performance analysis.
- Radionuclide Transport Models (linked to system analysis models):
    - Calculate radionuclide release and transport within the reactor and surrounding structures.
    - Calculate radionuclide transport from the reactor to the EAB and transport in the atmosphere (plume dispersion).
- Dosimetry Computer Models (linked to radionuclide transport models):
    - Calculate doses within and outside the site boundaries during normal operation and accident conditions. Used to determine whether the plant design meets offsite dose limits and criteria and risk goals.
- Uncertainty Assessment Computer Models:

- Categorize the uncertainties associated with the events' source terms and select the most impactful ones to be considered.
- These models are used in conjunction with the previously mentioned models to characterize the quantification and propagation of uncertainties and perform sensitivity analysis.

3.12.2 Evaluate LBEs Source Term Calculations Against F-C Target

The risk significance of individual LBEs is evaluated against the F-C target (see Figure 1-1). The uncertainties in mechanistic source term determinations and risk assessments are evaluated quantitatively in conjunction with the analytic and testing programs.

3.12.3 Evaluate Cumulative Risk Against QHOs and 10 CFR 20

The following are definitions of the Quantitative Health Objectives (QHOs) taken directly from the NRC 1986 Safety Goal Policy Statement [46]:

- "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor 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 U.S. population are generally exposed."
- "The risk to the population in the area of nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes."

The average individual risk of prompt (or early) fatality and latent cancer fatality that is calculated in the PRA to compare with the safety goals and the QHOs is the total plant risk incurred over a reactor year. This means the PRA results need to demonstrate that the total plant risk, i.e., the risk summed over all of the accident sequences in PRA, needs to satisfy both the latent cancer QHO and the early fatality QHO. The safety goals, and consequently, the QHOs are phrased in terms of the risk to an 'average' individual in the vicinity of a nuclear power plant per reactor year. The latent cancer QHO is defined in terms of the risk to an average individual within 10 miles and the early fatality QHO in terms of the risk to an average individual within 1 mile of the plant. Therefore, the PRA results need to show that the total integrated risk from the PRA sequences satisfy both the latent cancer QHO and the early fatality QHO. The following objectives should be met in evaluating cumulative risk:

- The total frequency of exceeding a EAB dose of 100 mrem (annual cumulative exposure limits in 10 CFR 20) from all LBEs should not exceed 1/plant-year.
- The average individual risk of early fatality within 1 mile of the EAB from all LBEs shall not exceed 5×10^{-7}/plant-year to ensure that the NRC safety goal QHO for early fatality risk is met.
- The average individual risk of latent cancer fatality within 10 miles of the EAB from all LBEs shall not exceed 2×10^{-6}/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

3.12.4 Identify Risk Significance of LBEs and Perform MST Calculations Against Regulatory Criteria

LBEs are classified in NEI-18-04, which is endorsed by RG 1.233, as risk-significant if the LBE EAB dose exceeds 2.5 mrem over 30 days and the frequency of the dose is within two orders of magnitude of the F-C target. Each design will establish barriers to the release of radioactive material from the fuel, RCS, or other systems, to maintain doses to below the criteria defined for various anticipated or postulated conditions. The specific conditions for each barrier's leakage, temperature, pressure, and time response will be design and event specific. The success of a barrier or combination of barriers in preventing releases from each source within the SRIR may simplify the assessment and preclude the need to assess offsite consequences.

In lieu of event-specific assessments, the analysis of offsite consequences or the leakage past specific barriers may be based on an MCA. Likewise, the leakage from an individual barrier could be assumed for an individual LBE, based on the worst conditions for that barrier from all LBEs. The definition of the MCA or events, as applicable, should be agreed upon between the applicant and the NRC consistent with the technology and safety characteristics of the design. For an MST, the timing, magnitude, and the form of radionuclides released into the barriers and the resulting temperature, pressure, and other environmental factors (e.g., combustible gas) in the barriers during the event should be analyzed mechanistically, with uncertainty considered. Using conservative assumptions is permitted in the MST and MCA dose calculations. For example, the timing of closure and the allowable leak rate is then established such that the worst two-hour dose at the EAB and the dose at the outer edge of the LPZ for the duration of the event do not exceed 25 rem TEDE.

3.13 Select a Final List of LBEs

Since the regulatory structure for advanced reactor technology licensing makes use of PRA, the selection of LBEs may not be a one-time licensing step, carried out at the time of initial plant licensing and remaining fixed. Instead, it is expected that both the selection of LBEs and the safety classification of SSCs may change as the reactor design is evolved and matured, and over the lifetime of the plant operations as new information and operational experience add to, and reshape, the risk insights from maintaining and updating the PRA.

The LBE evaluation provides feedback on whether additional improvements on design and operation should be considered. Such improvements could be motivated by a desire to increase margins against the F-C target criteria, reduce uncertainties in the LBE frequencies or consequences, limit the need for restrictions on siting or emergency planning, or enhance the performance against DID criteria. If improvements are needed, then go back to 3.6. If no improvements are needed, the final list of LBEs and safety-related structures, systems and components is established.

3.14 Documentation of Source Terms and Dose Rates

A document will be prepared to show the calculations of the source terms and dose rates for use in licensing, such as for the bounding analysis case or for the final list of LBEs. This information will be submitted to the NRC for approval as part of an application for a licensing action. The methodology used and scenarios analyzed for the source term and dose rate calculations should be presented in the document. The results from risk-informed and performance-based mechanistic source term calculations should include uncertainty quantification, as applicable, in both in the PRA models and in the mechanistic source term calculations.

4. SUMMARY

A risk-informed, performance-based, technology-inclusive determination of source terms for dose-related assessments for advanced nuclear reactor facilities is developed in this report to support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495 [1]) and the NRC's response to the NEIMA Public Law No: 115-439, of January 2019 [2]. This approach uses a graded process that allows both non-mechanistic source term calculation methods, which adopt conservative approaches and assumptions based on known physical and chemical principles, and, more importantly, the risk-informed and performance-based mechanistic source term calculation methods, which consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of LBEs, to be used for the design and licensing of advanced nuclear technologies.

The source terms developed with this graded approach and radionuclide inventories elsewhere in the facility that are determined during source term analysis can be used to address licensing issues to support the 10 CFR 52 Combined License (COL) application process. They can also be used for other purposes, including equipment environmental qualification, control room habitability analyses, and assessments of severe accident risks in environmental impact statements. The graded approach presented in this report for source term determination is, to the extent possible, generic to any of ongoing reactor designs and future reactor designs. It provides information on the review of the regulatory foundation for use of conservative bounding source terms as well as event-specific mechanistic source terms for advanced nuclear reactor designs.

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APPENDIX A: EXAMPLE OF ANALYSIS

This appendix provides an overview of how the methodology might be applied to an advanced reactor design, using a high-temperature gas-cooled reactor as a representative example. For each step in the methodology, a brief overview of the corresponding action or activity is given, and some representative examples of the kind of analysis and output expected from each step are given. These are not intended to be complete. In many cases, numerical values are used as example inputs or outputs of a calculation or analysis; it is important to note that these are only hypothetical and for the purpose of illustration only. They do not represent the results of actual analysis nor are necessarily representative of any particular reactor design or this reactor type generally.

Step 1: Identify Regulatory Requirements

Applicable regulatory requirements and dose limits have been outlined in Section 3.1 and Table 3-1. These are applicable to any reactor type, including an HTGR. For the purpose of this example, consider a prospective site of location and size that dictates the EAB be at most 300 m from the reactor. In the proceeding analysis, the applicant must demonstrate that that the regulatory requirements outlined in Table 3-1 are met for this particular EAB.

Step 2: Identify Reference Facility Design

The reference facility design is that described in [47], a single module 600 MWt thermal prismatic Modular HTGR (MHTGR), with a 700°C helium coolant outlet temperature. The reactor produces high-temperature steam via a steam generator. Barriers and processes important to the transport of fission products in the reactor are illustrated schematically in Figure A- 1.

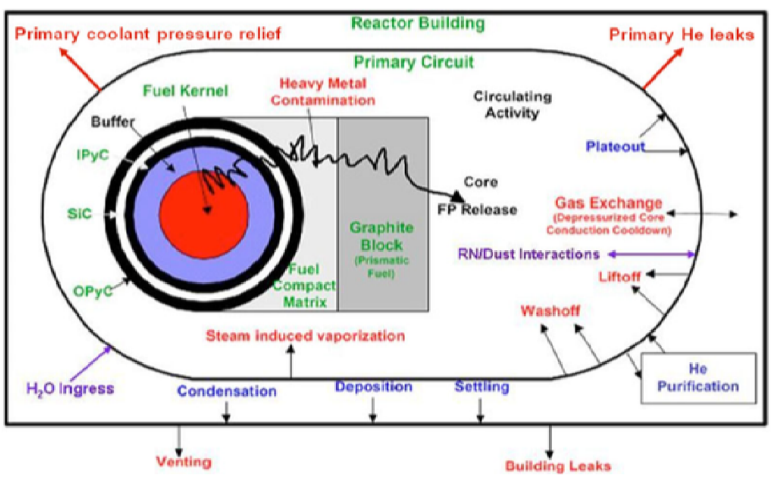

Figure A- 1 Barriers to fission product transport in a HTGR [18].

The reactor uses TRISO fuel, and this constitutes the primary barrier to fission product release. To be released from a fuel particle, radionuclides must be transported through and out of the fuel kernel itself and subsequently through each of the buffer, inner pyrolytic carbon, SiC, and outer pyrolytic carbon layers. Small fractions of the fuel with defects in one or more of the layers may dominate the release for a given radioisotope. Fission products that escape the fuel itself may be retained in the surrounding fuel compact matrix and graphite block; fission products that are transported through these materials are released to the primary coolant. Fission products circulating in the primary coolant may be removed by a coolant purification system; some fraction of these will be deposited on surfaces throughout the primary circuit but are retained by this barrier under normal operating conditions. Accidents involving a breach of the primary circuit may allow circulating activity or re-entrained deposits to be released to the reactor building, where some will be deposited via condensation, settling, or other mechanisms or removed by filters before they can be released through building leaks or vents to the environment.

Step 3: Define Initial Radionuclide Inventories

The applicant defines initial radionuclide inventories via a series of analyses:

A. Neutronics analysis to obtain the spatially varying neutron flux and energy spectrum in the core.

B. Radionuclide generation rates from fission (with input from the neutronics analysis), accounting for activation, decay, etc.

C. The radionuclide generation rates constitute a mass source input to a fission product transport code, which determines how these are distributed throughout the reactor system at the initiation of the accident. Such a code would incorporate models for transport through all the barriers delineated in Figure A- 1, including the kernel and multiple layers of both intact and defective or failed TRISO fuel; models for transport through matrix and graphite materials, and release from these to the primary coolant; and models for transport throughout the primary circuit, incorporating any models necessary to describe the mechanisms involved in deposition or resuspension of radionuclides on/from surfaces. The end results are fission product inventories deposited on different components and portions of the primary circuit, plus inventories remaining in all parts of the TRISO fuel, matrix, and graphite materials in different regions of the core. Table A-1 shows the initial core fission product inventories for the 600 MWt thermal MHTGR reactor design.

Table A- 1 Initial Core Fission Product Inventories for the 600 MWth MHTGR [47, 48].

Fission Product Class	**Characteristic Nuclide**	**Inventory (Curies)**
Noble gases	^{133}Xe	3.63E+07
	^{85}Kr	1.90E+05
	^{88}Kr	1.85E+07
I, Br, Te, Se	^{131}I	2.00E+07
	^{133}I	3.60E+07
	^{132}Te	2.71E+07
Cs, Rb	^{137}Cs	1.69E+06
	^{134}Cs	1.90E+06
Sr, Ba, Eu	^{90}Sr	1.69E+06

Ag, Pd	^{110m}Ag	2.81E+04
	^{111}Ag	2.96E+06
Sb	^{125}Sb	2.35E+05
Mo, Ru, Rh, Tc	^{103}Ru	3.61E+07
La, Ce groups	^{144}Ce	2.33E+07
	^{140}La	3.27E+07
Pu, actinides	^{239}Pu	4.66E+03

Step 4. Perform Bounding Calculations

Among many other isotopes, the applicant finds, as a result of the analysis in Step 3, that 1.7 MCi of ^{90}Sr are present in the reactor at the initiation of the accident scenario. Schedule C of 10 CFR Part 30 indicates that a release of 1% of the ^{90}Sr inventory must not exceed 90 Ci if no emergency plan is to be considered. In this case, 1% of the ^{90}Sr inventory is 17,000 Ci, far in excess of the 90 Ci limit. Without consideration of any other isotopes, simple bounding analysis is insufficient in this case, and the applicant should proceed with a mechanistic source term analysis.

Step 5. Conduct SHA and Perform Simplified Calculations

Conduct SHA

The applicant performs a SHA on the system to identify the SSCs and barrier penetration pathways and estimate the component and/or interaction likelihood of failures and severity. A team of experts consisting of the designers and subject matter experts are gathered and queried for this task. A partial outcome of such an analysis may include Table A-2, which uses an FMEA. FMEAs use a risk priority number (RPN) to quantify the priority of the hazards, should the design team have the capability to address them. The scale in this FMEA is on a 1–10 for severity, frequency, and detection. The high end of the severity scale for a reactor indicates that a release through the final barrier can be expected. Frequency in the SHA step is relative and estimated. Note that detection is inverse, in that it represents the inability to detect the fault/hazard.

The SHA can inform a redesign, based on recommended actions that, if implemented, will change the likelihood of the hazard, the SSCs in the barrier penetration pathway, or even the barrier penetration pathway itself. The hypothetical examples given in Table A-2 show that two recommended actions are viable for the four failure modes listed. Recommended actions are determined and are generally done by using either an RPN threshold (such as above RPN > 50 in this example) or something that is easy and cost effective to implement that lessens a high severity failure mode, such as adding filters to the building ventilation system in this hypothetical example. The other possible improvement actions in this example, such as possibly increasing the durability of the steam generator tubes or redesigning TRISO fuel, are not economically viable and are left as is.

The SHA is kept current with any design changes. Each new iteration of the SHA risk-informs the design, the simplified source terms quantification or it will inform the PRA. The SHA in this example will be used to inform the PRA.

Table A- 2 Example of FMEA Results.

Process Function	**Potential Failure Mode**	**Potential Causes/ Mechanisms of Failure**	**Severity**	**Frequency**	**Detection**	**RPN**	**Recommended Action**	**Implemented / Date**	**Severity**	**Frequency**	**Detection**	**RPN**
Transport of coolant	Pipe Rupture	Loss of coolant	7	4	2	56	Vibration dampeners	Yes / 29Oct20	7	2	2	28
Heat exchange	Steam generator tube break	Water ingress	7	3	2	42						
Fission Product Retention	FP release from fuel barriers	Particle failure during accident	9	1	2	18						
Environmental Controls	Building overpressure	Venting to environment	9	1	1	9	Add filters	Yes / 10Oct20	6	1	1	6

Perform Simplified Calculations

For the 600 MWth MHTGR, previous safety analyses indicate that breaks in the helium pressure boundary and water ingress events pose the greatest challenges with respect to offsite dose consequences [47, 48]. Step 7 has a more detailed description of these two accidents. The applicant adopts a simplified approach in which attenuation factors (inverse of release fraction) are assigned to a series of barriers to release. The attenuation factors are based on experimental data in conditions intended to bound the range of temperatures experienced in bounding accidents, and the NRC must approve of the specific methodology applied in this case. Some of the determined attenuation factors for non-intact fuel (TRISO failure) are shown in Table A-3. It is the retention in the fuel kernel itself that leads to attenuation in this case.

Table A- 3 Attenuation Factors for non-intact fuel (TRISO failure) for simplified source term calculations [47].

Fission Product Class	**Attenuation Factors: Accident Release from Non-Intact Fuel**	
Confidence Limit	**50%**	**95%**
Noble Gases	10	5
I,Br,Se,Te	10	5

Cs, Rb	1	1
Sr,Ba,Eu	1	1
Ag, Pd	1	1
Sb	1	1
Mo,Ru,Rh,Tc	100	50
La, Ce	100	50
Pu, Actinides	1000	500

If the results from the scoping analysis indicate that the NRC Siting and EPA PAG plume exposures criteria are met, the applicant could skip steps 6–13 and proceed to the last Step. In the example above, no additional retention of certain fission products (including ^{90}Sr from the preceding step) can be assumed for failed fuel during accident conditions. Therefore, the applicant proceeds with the mechanistic approach beginning with the next step.

Step 6. Consider Risk-informed System Design Changes

Based on the recommendations of the FMEA in the preceding step, the applicant decides to add vibration dampeners, in order to decrease the frequency of pipe rupture events that would lead to a loss of coolant. Realizing that such a loss of coolant could result in a building overpressure that necessitates venting directly to the environment, the applicant additionally decides to incorporate filters that would retain fission products and thereby reduce the severity of such an occurrence.

Step 7. Select Initial List of LBEs and Conduct PIRT

Based on the findings of the FMEA in Step 5, the applicant identifies an initial list of LBEs "which may not be complete but are necessary to develop the basic elements of the safety design" [22]. Two events in the initial list of LBEs include those described in [47, 48]:

1. A break in the helium pressure boundary with loss of forced cooling:
    a. Leak or break in the helium pressure boundary piping up to the largest connecting pipe
    b. Reactor trip
    c. Loss of heat transport to the energy conversion system
    d. Loss of shutdown cooling
    e. Immediate depressurization of helium in the helium pressure boundary
    f. Opening of the RB vent to relieve helium pressure.

2. A water ingress event:
    a. Steam generator tube break
    b. Reactor trip
    c. Loss of heat transport to the energy conversion system

g. Over-pressurization of the helium pressure boundary through the vessel system relief valve
h. Opening of the reactor building vent to relieve helium and water/steam pressure.

These are thought to encompass all of the relevant transport phenomena that might occur in HTGR accidents, including those resulting from steam interactions and transport in the reactor building plus all of the same transport phenomena occurring inside the helium pressure boundary, core, and fuel that occur during less severe accidents. They are therefore sufficient to develop the basic elements of safety design.

The applicant proceeds to conduct a PIRT to identify the phenomena relevant to the progression of these scenarios and the importance and current knowledge base of these. Such a PIRT has been conducted for the HTGR and is documented in [37]. Some example entries are given in Table A-4.

Table A- 4 PIRT Sample Results.

Phenomenon	**Importance**	**Rationale**	**Knowledge Level**	**Rationale**	**Model Status**
FP transport through fuel block	5 – High	Effective release rate coefficient (empirical constant) as an alternative to first principles (IC and Trans.)	1 – Low 4 – Medium	Depends on specific graphite; expected from material PIRT	Major need
Steam attack on graphite	5 – High	If credible source of water present; design dependent (Trans.)	1 – Low 4 – Medium	Historical data	Major need for severe accidents
Aerosol/dust deposition	5 – High	Gravitational, inertial, thermophoresis, electrostatic, diffusional, turbophoresis (Trans.)	5 – Medium	Reasonably well-developed theory of aerosol deposition by most mechanisms except inertial impact in complex geometries; applicability to NGNP unclear	Minor Mod

Step 8. Establish Adequacy of MST Simulation Tools

At this stage, the applicant has already developed a code intended to model all aspects of fission product transport, in normal and off-normal conditions. As a result of the PIRT findings in the preceding step, the applicant has identified that:

1. Their existing model for fission product transport in graphite is rather uncertain and not adequately informed by relevant experiment data.
2. It does not presently include any model for steam interaction with graphite.
3. The applicant has a model for aerosol transport in the reactor building, but there is a question as to whether some of the deposition mechanisms apply to HTGRs and other mechanisms (i.e. inertial impaction) are thought to be important.

In response to the first two findings, the applicant plans some additional post-irradiation experiments on their fuel and graphite materials. The first involves heating compacts with failed particles in order to observe the resultant distribution of mobile fission products that are transported into the graphite; this data is used to update the models for fission product transport in graphite. The second involves heating fuel compacts and graphite in a furnace in helium atmospheres with varying amounts of steam, at temperatures representative of a severe accident condition. The data collected during these experiments is used to inform a graphite-steam oxidation model that the applicant develops and incorporates into their code.

In response to the third finding, the applicant identifies a large body of data on inertial impaction of dust and aerosols in the existing literature and uses these to develop and implement a model for this into their code. They also review the applicability of the various other deposition mechanisms. Finding some uncertainty as to whether these are applicable or not, the applicant decides to conservatively model only gravitational, inertial, and diffusional deposition.

Step 9. Develop and Update PRA Model

The applicant conducts a PRA to take the hazards and severities identified in the SHA, determine the frequencies of initiators and the probability of failure of mitigating actions, and place them into logic trees. Frequencies were defined within ranges in the SHA. In the PRA, the initiating event frequency (per year) is quantified for each of the events identified. The mitigating systems are modeled as success or failure in fault trees, based on the probability of failure of required components and operator actions. Following the guidance in Section 3.9, the total URF for all sequences from one initiator to UR is the URF for one event. The total sequences from all initiators that lead to UR are summed for a total URF.

The PRA is developed to the current state of knowledge of the design and of the initiating event frequencies and resulting URFs before moving on to the selection of LBEs. The PRA should be kept current to provide a tool to use for risk-informed decision-making and to provide sequences that lead to UR for inclusion in the modeling and simulation step.

In some instances, the results of the modeling will in turn validate or invalidate the sequences developed in the PRA. A sequence tested through modeling that is thought to lead to UR may not, or vice versa. The PRA is then updated to reflect the new information gained through modeling and simulation, and the LBE selection step and modeling is performed again, as necessary, until all are in agreement.

Step 10. Identify or Revise the List of LBEs

Based on the evaluation of the risk information, the applicant expands the list of LBEs to include normal operation and a comprehensive set of AOOs, DBEs, and BDBEs. For example, if the scenarios leading to loss of coolant outlined in Step 7 are determined by the PRA to have an initiating event frequency between 10^{-2} and 10^{-4} per year, these would be classified as DBEs. Examples of LBEs at the far ends of the frequency-consequence spectrum that might be considered include:

- Analysis of tritium transport during normal operations. Tritium generation rates in the HTGR are not high, but tritium is uniquely mobile and may be able to diffuse through parts of the primary helium circuit even during normal operation.
- A severe accident such as a large-break loss-of-coolant accident that results in significant air or steam ingress into the graphite or core. This event is determined to be very unlikely and beyond the design basis.

Step 11. Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

External hazards are site specific and not design specific. The external hazards considered for the PRA model of the MHTGR use a full-scope PRA treatment of internal and external hazards. However, it is expected that the selection of LBEs performed in Step 10 is based on a PRA that includes internal events but has not yet been expanded to address external hazards. The external events encompass all potential hazards applicable to the site and could include seismic, flooding, high winds, and external fires. It is reasonable to expect that safety function failures will be dominated by events and conditions that exceed the design basis envelope for passive SSCs. Extreme external hazards represent one way this can occur. The DBEHL are defined in this step, which can include the ground motion peak acceleration "g" values for seismic events, maximum wind speed for high winds, maximum flood level for external flooding, and the possible damage from the wildfire events. The safety-related SSCs of the MHTGR are required to be capable of performing their reactor safety functions in response to external events within the DBEHL, and there will be no new LBEs introduced by external hazards.

Step 12. Perform Source Terms Modeling and Simulation for LBEs

As discussed in Section 3.12 above, the applicant uses their fission product transport code (e.g., XSTERM) or the NRC's fission product transport code (MELCOR), supported by neutronic, thermal-hydraulic, and other analysis tools as necessary, to perform an analysis of the LBEs identified in Step 10. At this stage, the code has been revised and informed by the additional experiment data collected as a part of Step 8, verified, and validated. For the two DBAs outlined in Step 7, the following source terms are calculated:

Table A- 5 Example source terms for a break in the He pressure boundary [47].

Fission Product Class	**Nuclide**	**Short Term Release (curies)**			**Long-Term Release (curies)**		
		50%	**Mean**	**95%**	**50%**	**Mean**	**95%**
Noble gases	^{133}Xe	3.99E+01	3.99E+01	3.99E+01	4.92E+01	6.44E+01	1.68E+02
	^{85}Kr	2.10E-01	2.10E-01	2.10E-01	3.39E-01	4.50E-01	1.21E+00
	^{88}Kr	2.06E+01	2.06E+01	2.06E+01	1.38E-04	1.82E-04	4.91E-04
I, Br, Te, Se	^{131}I	5.51E-02	1.61E-01	6.10E-01	2.85E+00	6.11E+00	2.24E+01
	^{133}I	1.00E-01	2.90E-01	1.08E+00	1.17E+00	2.64E+00	9.72E+00
	^{132}Te	7.44E-02	2.17E-01	8.08E-01	3.12E+00	6.51E+00	2.41E+01
Cs, Rb	^{137}Cs	1.17E-01	3.43E-01	1.33E+00	1.15E-01	3.31E-01	1.28E+00
	^{134}Cs	1.93E-02	5.54E-02	2.10E-01	1.32E-01	3.82E-01	1.49E+00
Sr, Ba, Eu	^{90}Sr	1.57E-03	4.56E-03	1.72E-02	1.79E-01	4.71E-01	1.76E+00
Ag, Pd	^{110m}Ag	3.86E-02	1.13E-01	4.22E-01	8.61E-01	2.30E+00	8.51E+00
	^{111}Ag	7.96E-01	2.28E+00	8.93E+00	5.48E+01	1.72E+02	6.49E+02
Sb	^{125}Sb	7.27E-04	2.12E-03	8.15E-03	1.78E-03	5.23E-03	2.05E-02
Mo, Ru, Rh, Tc	^{103}Ru	8.05E-04	2.34E-03	8.68E-03	7.19E-01	1.96E+00	7.54E+00

La, Ce groups	^{144}Ce	9.74E-03	2.75E-02	1.05E-01	4.57E-02	1.30E-01	5.01E-01
	^{140}La	7.46E-04	2.16E-03	8.13E-03	2.88E-02	7.65E-02	2.92E-01
Pu, actinides	^{239}Pu	1.90E-07	5.74E-07	2.10E-06	8.97E-07	2.60E-06	1.01E-05

Table A- 6 Example source terms for a water ingress event [47].

Fission Product Class	**Nuclide**	**Short Term Release (curies)**			**Long-Term Release (curies)**		
		50%	**Mean**	**95%**	**50%**	**Mean**	**95%**
Noble gases	^{133}Xe	3.99E01	3.99E+01	3.99E+01	1.08E02	1.54E02	4.42E02
	^{85}Kr	2.10E-01	2.10E-01	2.10E-01	7.31E-01	1.06E00	3.08E+00
	^{88}Kr	2.03E+01	2.03E+01	2.03E+01	3.05E-04	4.32E-04	1.25E-03
I, Br, Te, Se	^{131}I	1.10E+00	1.90E+00	6.56E+00	3.16E+00	6.90E+00	2.48E+01
	^{133}I	2.01E+00	3.36E+00	1.12E+01	1.31E+00	2.85E+00	1.02E+01
	^{132}Te	1.48E+00	2.54E+00	8.52E+00	3.29E+00	7.00E+00	2.54E+01
Cs, Rb	^{137}Cs	2.37E+00	4.06E+00	1.35E+01	2.59E-01	7.06E-01	2.67E+00
	^{134}Cs	3.77E-01	6.36E-01	2.16E+00	2.92E-01	7.81E-01	3.09E+00
Sr, Ba, Eu	^{90}Sr	3.12E-02	5.28E-02	1.72E-01	3.62E-01	9.59E-01	3.58E+00
Ag, Pd	^{110m}Ag	7.60E-01	1.30E+00	4.31E+00	8.89E-01	2.01E+00	7.33E+00
	^{111}Ag	1.57E+01	2.65E+01	8.81E+01	5.59E+01	1.53E+02	5.60E+02
Sb	^{125}Sb	1.51E-02	2.52E-02	8.35E-02	1.95E-03	5.52E-03	2.17E-02
Mo, Ru, Rh, Tc	^{103}Ru	1.57E-02	2.66E-02	8.79E-02	6.91E-01	1.69E+00	6.42E+00
La, Ce groups	^{144}Ce	1.98E-01	3.33E-01	1.08E+00	4.82E-02	1.13E-01	4.08E-01
	^{140}La	1.45E-02	2.46E-02	8.36E-02	2.81E-02	6.66E-02	2.44E-01
Pu, actinides	^{239}Pu	3.70E-06	6.23E-06	2.09E-05	9.07E-07	2.13E-06	7.80E-06

In Tables A-5 and A-6, a “short term” release indicates the prompt release during depressurization of fission products that were present in the primary circuit as a result of normal operations, and a "long-term" release indicates a delayed release associated with heatup of the fuel over the course of the accident. In addition to the radionuclide inventories and timing of the release, mechanistic calculations should address the thermal energy associated with, and physical and chemical forms of, those radionuclides; for example, radionuclides may exist as vapors or may be adsorbed on dust.

Using these mechanistic source terms, the applicant performs atmospheric transport calculations to determine transport to the EAB, followed by dose calculations based on that remaining fraction of radionuclides transported there. The results are summarized in Table A-7:

Table A- 7 Example calculated dose comparison with regulatory criteria [48].

Regulatory Criteria (scenario)	**Event Scenario**	**Exposure**	**Calculated Dose (rem)**	**Regulatory Criteria (rem)**
EAB at 400m (TEDE)	Break in Helium Pressure Boundary	Worst 2 hours	0.02	25
LPZ at 400m (TEDE)	Break in Helium Pressure Boundary	Cloud Passage	1.32	25
EAB at 400m (TEDE)	Water Ingress Event	Worst 2 hours	0.46	25
LPZ at 400m (TEDE)	Water Ingress Event	Cloud Passage	6.43	25
EPA PAG Plume Exposure Related Dose (TEDE)	Break in Helium Pressure Boundary	4 days	0.04	1
EPA PAG Plume Exposure Related Dose (TEDE)	Water Ingress Event	4 days	0.05	1
EPA PAG Plume Exposure Related Dose (Thyroid)	Break in Helium Pressure Boundary	4 days	0.18	5
EPA PAG Plume Exposure Related Dose (Thyroid)	Water Ingress Event	4 days	0.25	5

The applicant finds that doses resulting from all the LBEs do not exceed the frequency-consequence targets illustrated in Figure 1-1. The cumulative risk is also assessed, and it is found that:

- The total frequency of exceeding a EAB dose of 100 mrem is less than 1/plant-year.
- The average individual risk of early fatality within 1 mile of the EAB from all the LBEs is less than 5×10^{-7}/plant-year.
- The average individual risk of latent cancer fatality within 10 miles of the EAB from all the LBEs is less than 2×10^{-6}/plant-year.

The regulatory limits identified in Section 3.12.3 have therefore been met. Finally, the applicant identifies all events classified as "Risk-significant LBEs," i.e. those within two orders of magnitude of the frequency-consequence targets in Figure 1-1, as areas for potential future improvement.

Step 13. Review LBEs List for Adequacy of Regulatory Acceptance

The LBE evaluation performed in the previous steps provides feedback on whether additional improvements on design and operation should be considered. Such improvements could be motivated by a desire to increase margins against the F-C target criteria, reduce uncertainties in the LBE frequencies or consequences, limit the need for restrictions on siting or emergency planning, or enhance the performance against DID criteria. The applicant concludes that no improvements are needed for the MHTGR, and the final list of LBEs and safety-related SSCs is established.

Step 14. Document Completion of Source Term Development

Having found, at the completion of the analyses, that all regulatory requirements have been met, the applicant documents these and submits the documentation to the NRC for approval. As the PRA is updated to reflect any changes that occur as part of a continuing design process or modifications to the plant during its operating life, the list of LBEs is revisited and steps 6–14 are repeated as/if necessary based on the updated set of LBEs and PRA results.