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# Simplified Approach for Scoping Assessment of Non-LWR Source Terms

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## ABSTRACT

This report describes a structure to aid in evaluation of release mitigation strategies across a range of reactor technologies. The assessment performed for example reactor concepts utilizes previous studies of postulated accident sequences for each reactor concept. This simplified approach classifies release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance. It is not, however, intended to develop quantitative estimates of radiological release magnitudes and compositions to the environment. Rather, this approach is intended to identify the characteristics of a reactor design concept's release mitigation strategies that are most important to different classes of accident scenarios. It uses a scoping methodology to provide an approximate, order-of-magnitude, estimate of the radiological release to the environment and associated off-site consequences. This scoping method is applied to different reactor concepts, considering the performance of barriers to fission product release for these concepts under sample accident scenarios. The accident scenarios and sensitivity evaluations are selected in this report to evaluate the role of different fission product barriers in ameliorating the source term to the environment and associated off-site consequences.

This report applies this structure to characterize how release mitigation measures are integrated to define overall release mitigation strategies for High Temperature Gas Reactors (HTGRs), Sodium Fast Reactors (SFRs), and liquid fueled Molten Salt Reactors (MSRs).

To support this evaluation framework, factors defining a chain of release attenuation stages, and thus an overall mitigation strategy, must be established through mechanistic source term calculations. This has typically required the application of an integral plant analysis code such as MELCOR. At present, there is insufficient evidence to support a priori evaluation of the effectiveness of a release mitigation strategy for advanced reactor concepts across the spectrum of events that could challenge the radiological containment function. While it is clear that these designs have significant margin to radiological release to the environment for the scenarios comprising the design basis, detailed studies have not yet been performed to assess the risk profile for these plants. Such studies would require extensive evaluation across a reasonably complete spectrum of accident scenarios that could lead to radiological release to the environment.

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# ACRONYMS AND DEFINITIONS

| Abbreviation | Definition |
|---|---|
| AEC | United States Atomic Energy Commission |
| ARE | Aircraft Reactor Experiment |
| ART | Aircraft Reactor Test |
| CRBR | Clinch River Breeder Reactor |
| DLOF | Depressurized Loss of Flow |
| DOE | United States Department of Energy |
| ELAP | Extended Loss of AC Power |
| EPRI | Electric Power Research Institute |
| FFTF | Fast Flux Test Facility |
| FSAR | Final Safety Analysis Report |
| HTGR | High Temperature Gas Reactor |
| INL | Idaho National Laboratory |
| LOHR | Loss of Heat Removal |
| LPF | Leak Path Factor |
| MAAP | Modular Accident Analysis Program |
| MACCS | MELCOR Accident Consequence Code System |
| MCCI | Molten Core-Concrete Interaction |
| MSBR | Molten Salt Breeder Reactor |
| MSR | Molten Salt Reactor |
| MSRE | Molten Salt Reactor Experiment |
| NEI | Nuclear Energy Institute |
| NGNP | Next Generation Nuclear Plant |
| NRC | United States Nuclear Regulatory Commission |
| ORNL | Oak Ridge National Laboratory |
| PCL | Primary Coolant Leak |
| PLOF | Protected Loss of Flow |
| PRA | Probabilistic Risk Assessment |
| PRISM | Power Reactor Innovative Small Module |
| PSAR | Preliminary Safety Analysis Report |
| PSER | Preliminary Safety Evaluation Report |
| PSID | Preliminary Safety Information Document |
| RN | Radionuclide |

| Abbreviation | Definition |
|---|---|
| SAWA | Severe Accident Water Addition |
| SAWM | Severe Accident Water Management |
| SFR | Sodium-cooled Fast Reactor |
| SNL | Sandia National Laboratories |
| SOARCA | State-of-the-Art Reactor Consequence Analyses |
| TRISO | Tristructural Isotropic |
| ULOF | Unprotected Loss of Flow |
| UTOP | Unprotected Transient Overpower |

# 1. INTRODUCTION

## 1.1. Background

The Nuclear Regulatory Commission (NRC) is developing a technology-inclusive, risk-informed, performance-based framework for licensing and regulating non-light water reactors (non-LWRs). This approach requires an initial evaluation of release mitigation strategies for hypothetical accident scenarios for non-LWR design concepts. This evaluation utilizes a simplified or scoping evaluation of radiological releases to the environment (i.e., the source term to the environment) and off-site consequences. Currently relevant reactor concepts considered in this evaluation are the High-Temperature Gas-cooled Reactor (HTGR), the Sodium-cooled Fast Reactor (SFR), and Molten Salt-fueled Reactor (MSR). Assessing source terms in a simplified manner is useful to develop a regulatory strategy for evaluating key technical and policy issues, as well as regulatory engagement plans.

Each reactor concept may have different designs, in some cases corresponding to different vendors. In order to perform adequate analyses for a specific reactor concept, this document focuses on common features across the range of designs proposed for each reactor type. In the context of evaluating release mitigation strategies using a scoping source term, design-specific details are generally not relevant unless a specific design employs distinct features to ameliorate off-site radiological risk. The scope includes a pebble bed HTGR using Tristructural Isotropic (TRISO) fuel, pool-type SFR using metallic fuel, and MSR using liquid fuel and fluoride salts. This document outlines a structure for characterizing radiological release mitigation strategies, leveraging DOE's safety analysis approach for radiological source term and dose consequence calculations [1].

The evaluation of LWR source terms to the environment and off-site consequences have evolved to utilize computer models based on mechanistic models of fission product release and transport. These mechanistic models have developed over more than 30 years and have been integrated into computer codes such as NRC's MELCOR or the Electric Power Research Institute's (EPRI's) Modular Accident Analysis Program (MAAP). These codes have enabled the development of methods that calculate a mechanistic source term for LWRs. This capability has been utilized in a range of regulatory evaluations and rulemakings, including the recent Containment Protection and Release Reduction (CPRR) rulemaking by both the NRC [2] and industry [3].

The concept of a mechanistic source term was first introduced in SECY-93-092 [4]. A mechanistic source term calculation involves the quantitative evaluation of all relevant fission product release and transport processes that govern how radionuclides migrate from fuel, through a plant, and ultimately to the environment. This evaluation must be done across a range of risk-significant accident scenarios and account for physical processes that contribute to retention or transmutation of fission products in different regions of the plant. A mechanistic source term calculation must have the following supporting attributes.

- Sufficient knowledge is available regarding reactor and fuel performance under normal and off-normal conditions. This requires an appropriate technical basis established through research, development, and testing programs. Such a technical basis instills confidence in the capability of a mechanistic analysis to calculate a credible source term. Structured approaches to address areas of uncertainty and their potential impact have only recently received attention. For a given reactor design, uncertainty could conceivably affect the consequence spectrum and release mitigation strategy. Primary examples of assessing uncertainty in this respect are provided by

NRC evaluations for CPRR rulemaking [2] and by complementary evaluations performed by industry [3].

- A sufficient technical basis exists to characterize fission product transport through all barriers and pathways to the environment. For source term calculations, it is especially important to have an adequate understanding of how containment function can mitigate radiological release to the environment. It is critical that such calculations be as realistic as possible to adequately evaluate efficacy and limitations of barriers to fission product release.
- Sufficient understanding of event scenarios exists – for a given reactor design – to establish that selected events leading to radiological release adequately bound severe accident and design-dependent uncertainties.

The level of modeling detail embedded in these computer codes reflects the state-of-knowledge regarding LWR accident progression and source terms. A similar state-of-knowledge does not presently exist for non-LWRs, but is expected to evolve through the process of design development and licensing. Given the different level of knowledge between LWR and non-LWR source term phenomena, an initial scoping source term evaluation approach affords insight into how different reactor designs ameliorate off-site radiological risk.

All reactor designs implement radiological release mitigation strategies to limit the potential for off-site radiological consequences. A strategy involves a set of measures designed to either reduce the likelihood of or consequences from a radiological release event. Examples of release mitigation measures include

- Design measures to achieve high reliability for reactivity control and thus significantly reduce the likelihood of reactivity excursion events contributing to plant risk
- Introduction of highly reliable emergency core cooling systems to eliminate the potential for extensive fuel damage in the event of loss of coolant inventory due to failure of reactor pressure boundary components
- Robust reactor enclosure structure to prevent or mitigate release of fission products to the environment
- Filtered venting systems to remove radiological contaminants from controlled discharge of effluent to the environment
- Pressure suppression pools to control containment overpressure excursions and also scrub fission product aerosols from the containment atmosphere

A release mitigation strategy combines one or more of these measures to limit offsite radiological risk for a spectrum of radiological release events. For example, in the case of a BWR Mark I, mitigation of radiological risk in the event of an Extended Loss of AC Power (ELAP) progressing to ex-vessel damage involves the following measures

- Operation of the RCIC system for an extended period following reactor shutdown to provide time for compensatory measures to be staged and initiated
- Preservation of containment integrity through opening of a hardened wetwell vent
- Scrubbing of fission product aerosols from the wetwell atmosphere by discharge of contaminated reactor and drywell gases through the large mass of water in the wetwell
- Deployment of reliable Severe Accident Water Addition (SAWA) after 8 hours from the time of the initiating event to stabilize ex-vessel damage conditions

- Preservation of the wetwell vent through Severe Accident Water Management (SAWM), which reduces the SAWA water injection rate 4 hours after SAWA commencement

This example of a release mitigation strategy is described in NEI-13-02 [5], with the technical basis established in Reference [3]. This work is a current example of the use of mechanistic source term evaluation in a full range Probabilistic Risk Assessment (PRA) to quantify the risk reduction attributable to a range of release mitigation strategies.

## 1.2. Scope

For non-LWR designs, the level of phenomenological knowledge and modeling capabilities may not be initially sufficient to support a mechanistic source term calculation. Throughout the regulatory review process, various stages of evaluating a reactor design must be applied. While acceptable, it may not always be feasible to have mechanistic source term calculations available during the initial assessment phase of a reactor design. At this stage, it is useful to assess release mitigation strategies to identify issues requiring enhanced regulatory scrutiny. Such initial evaluations do not require detailed evaluations of the source term. Rather, understanding of the critical measures supporting effective release mitigation is essential during initial evaluation. This helps to prioritize review efforts. It also aids in assessing where the level of knowledge (from experiments, probabilistic modeling or deterministic modeling) required to support a set of release mitigation strategies must be enhanced.

This report does not present a means of preparing mechanistic source terms for advanced reactors. This is to be provided through separate guidance reflecting the evolution of the state-of-knowledge in the modeling of non-LWR accident progression and source term modeling. This report focuses on describing a framework by which to generically assess release mitigation strategies using scoping source term calculations.

The simplified source term evaluation structure presented in this report provides an initial, primarily qualitative, means to identify the dominant considerations that affect a release mitigation strategy. Areas of significant uncertainty can be targeted through the structured evaluation of the important measures that form an overall release mitigation strategy.

This report is organized as follows:

- Section 2 summarizes the reference reactor designs provided as examples in this report and gives an overview of available documentation for assessing associated release mitigation strategies
- Section 3 provides a summary of the framework for generically evaluating non-LWR release mitigation strategies through a scoping source term evaluation
- Section 4 applies this framework to understand the ways in which HTGR, SFR and MSR design concepts mitigate releases for selected design-specific scenarios

# 2. REVIEW OF REACTOR CONCEPTS AND REFERENCE DOCUMENTS

This section provides a summary of historical consequence studies and code assessments for HTGRs, SFRs, and MSRs.

Many of the evaluations that have been performed for the non-LWR concepts considered in this report have focused on potential source terms for design basis accidents. These events are defined to ensure that a design is sufficiently robust at *preventing* escalation to the point where fission product release to the environment is possible. While these demonstrate the overall robustness of the design, they do not consider the important role of accident mitigation in contributing to an appropriate level of defense-in-depth. As noted in INSAG-10 [6],

> "Prevention of accidents remains the highest priority among the safety provisions for future plants. As already indicated in INSAG-3 [7], concerning the estimated probability of severe core damage, figures below $10^{-5}$ per plant year ought to be achievable. However, values that are much smaller than this would, it is generally assumed, be difficult to validate by methods and with operating experience currently available. Improved mitigation is therefore an essential complementary means to ensure public safety."

This is one important reason for the consideration of LWR source terms based on core melt events. Severe accident source terms were considered in regulatory evaluations as a result of WASH-1400 [8] and the Three Mile Island accident. The role of beyond design basis accidents in defining overall radiological risk for non-LWRs is captured through the risk-informed, performance-based approach adopted by the Licensing Modernization Project (LMP) [9] for developing licensing bases for non-LWRs. To implement the LMP, it is necessary to develop a set of beyond design basis event scenarios informed by a Probabilistic Risk Assessment (PRA). This effort is still in development for a range of non-LWR concepts. As a result, this report focuses on characterizing scenarios that have been previously identified, as well as investigating the impact on consequences associated with impairment of fission product retention barriers.

## 2.1. High Temperature Gas-Cooled Reactors

HTGRs have been studied for decades and the state of HTGR knowledge has benefited from domestic licensed power reactors (i.e., Peach Bottom 1 and Fort St. Vrain), preliminary safety information documents (PSIDs), subsequent NRC preliminary safety evaluation reviews (PSERs), and a relatively recent and robust DOE and industry licensing program under the Next Generation Nuclear Plant (NGNP) program. This section provides a brief review of HTGR release and consequence studies as well as code assessments.

HTGR reactors are typically categorized as prismatic or pebble bed reactors. Both the prismatic and pebble bed reactors utilize tristructural isotropic (TRISO) fuel. One of the main differences, at least with regards to core configuration, is that prismatic cores place the TRISO particles into fuel compacts whereas the pebble bed reactors place the TRISO particles into fuel pebbles. The fuel pebble and TRISO fuel particle are illustrated in Figure 2-1. Two fuel types have been proposed for the TRISO particle kernel: uranium oxide ($UO_2$) and uranium oxycarbide (UCO). $UO_2$ has more extensive experimental and civilian experience. UCO has much less experience, but the primary benefit of UCO fuel is the ability to better control carbon monoxide production and internal pressure buildup. Surrounding the fuel kernel are several protective layers meant to encapsulate fission products. Immediately surrounding the fuel kernel is a porous carbon buffer to

accommodate the buildup of fission product gases and gases from chemical reactions of various carbon layers. The inner pyrocarbon (IPyC) layer has two major functions: (1) shields the kernel from chlorine released during fabrication (SiC deposition) and (2) accommodates the mechanical stress distribution within the particle. The silicon carbide (SiC) layer is the major fission product barrier. The outer pyrocarbon (OPyC) layer encloses the SiC layer and isolates the SiC layer from the matrix.

A fuel pebble contains between 11,000 and 15,000 TRISO fuel particles, depending on the design. Fuel compacts contain about 10,000 particles. There are two primary sources of pressure build-up in the fuel kernel during operation. These influence the mechanical integrity of the fuel kernel.

- Fission gas retention in the kernel begins to diminish up to 20% fuel burnup. Up to this burnup, krypton, xenon, iodine, and cesium will be held up in the kernel.
- CO pressure buildup depends on oxygen potential and temperature. Oxygen potential depends on $UO_2$ burnup.

Although UCO helps reduce the formation of gaseous CO, cesium mobility is greater.

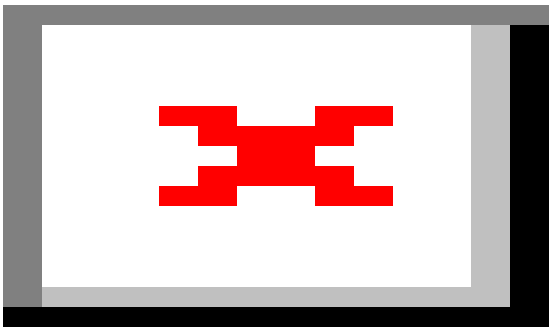

**Figure 2-1. Illustration of fuel pebble (top) and TRISO fuel particle (bottom) [10]**

In addition to pressurization of a fuel kernel, thermal gradients across a fuel particle can induce additional mechanical stresses. The amoeba effect is the asymmetrical movement of the fuel kernel due to temperature gradients across the particle. The temperature gradient causes carbon to move within the particle which pushes the kernel up against the SiC layer. This effect is greatly reduced for pebble bed and prismatic reactors with relatively low temperature gradients and burnups.

A number of design factors relevant to performance of a fuel pebble were identified in NUREG-6844 [10]:

- Matrix material specification – binds fuel particles and determines fuel element properties.
- Particle packing factor – determines in part the nuclear and thermal power properties. Higher packing factor likely to lead to more particle damage fraction.

- Unconfined heavy metal outside SiC layer – results in FPs in the primary circuit.
- Particle distribution in fuel element – inhomogeneous distribution can lead to hot spots.
- Particle overcoat – protects the particle during fabrication
- Fuel free zone – fuel pebble needs a fairly strong outer layer to protect the inner-fueled region from damage as the pebble must be repeatedly dropped several meters into the pebble bed core.

Unlike an LWR, an HTGR does not have core structures that would experience melting under accident conditions. This precludes the potential for early melt formation (i.e., prior to actual fuel melting) that in LWRs leads to an early challenge to core coolable geometry.

Kernel heatup accident factors identified in NUREG-6844 [10] include:

- Maximum fuel temperature
- Temperature vs time transient conditions (time-dependent variations of fuel with time will determine diffusive release)
- Energy transport: conduction within kernel
- Thermodynamic state of fission products (chemical state of fission products determines how they migrate and vapor pressures)
- Condensed phase diffusion
- Gas phase diffusion
- Oxygen flux
- Grain growth
- Buffer carbon-kernel interaction

### 2.1.1. Previous HTGR Source Term Studies

A number of scoping source term analyses for HTGR plants (both for prismatic and pebble bed reactor designs) have been performed. Many of these source term studies have been performed to demonstrate that the fission product release for design basis accidents (DBAs) is very unlikely. However, in some of these source term studies, the authors assumed key scenarios that can cause them to be classified as beyond-design basis accidents (BDBAs).

- US Advanced Gas Reactor Program – experiments that highlight TRISO fuel failures and RN release. The US Advanced Gas Reactor (AGR) Program ran experiments that focused on the behavior of TRISO particles during accident scenarios. Several tests were run to explore the TRISO behavior where peak fuel temperatures are in the range of 1600 and 1800 °C.
- mHTGR Source Term and Containment Report [11]
- The NGNP produced a number of reports on the topic of HTGR source terms

### 2.1.2. HTGR Inventory and Release References

Table 2-1 and Table 2-2 present a summary of fission product inventory and release considerations from past HTGR studies.

**Table 2-1: Inventory Overview for previously studied HTGRs**

| Power Level | Inventory Elements | Reactor/Program | Reference |
|---|---|---|---|
| 250 MWth | Cs, Sr, Ag, I with | HTR-PM | [12] |

| Power Level | Inventory Elements | Reactor/Program | Reference |
|---|---|---|---|
| | chemical forms | | |
| 600 MWth | Kr, Xe, I, Te, Se, Cs, Sr, Ag, Sb, Ru, La, Ce, Pu | HTGR-700C/NGNP | [13] |
| 600 MWth | Kr, Xe, I, Te, Se, Cs, Sr, Ag, Sb, Ru, La, Ce, Pu | HTGR-900C/NGNP | [13] |
| 250 MWth | Kr, Xe, I, Te, Se, Cs, Sr, Ag, Sb, Ru, La, Ce, Pu | HTGR-700C/NGNP | [13] |
| 350 MWth | Full Level 3 PRA | mHTGR | [14] |

**Table 2-2: Release Estimates from HTGRs**

| Power Level | Information on Releases | Reactor/Program | Reference |
|---|---|---|---|
| 250 MWth | Primary circuit circulating concentrations | HTR-PM | [12] |
| 600 MWth | Fuel Failure Rates, Attenuation Fractions, Releases | HTGR-700C/NGNP | [13] |
| 600 MWth | Fuel Failure Rates, Attenuation Fractions, Releases | HTGR-900C/NGNP | [13] |
| 250 MWth | Fuel Failure Rates, Attenuation Fractions, Releases | HTGR-700C/NGNP | [13] |
| 350 MWth | Full Level 3 PRA | mHTGR | [14] |

Based on past operational and manufacturing experience of pebble bed reactors and TRISO fuel, contamination and fuel defects are inherent to HTGRs. In the NGNP scoping analysis, these were identified and discussed. The three main sources considered were heavy metal contamination, silicon carbide (SiC) defects, and in-service failures. The understanding of heavy metal contamination and SiC defects is based on current fuel fabrication experience. Under accident conditions, the extent to which the impact of fuel defects on the source term depends on the accident scenario.

## 2.2. Sodium Fast Reactors

SFRs have been studied for a number of decades and the state-of-knowledge for SFRs has benefited from

- Domestic licensed power reactors (i.e., Fermi 1)
- Research reactor design, construction, licensing and operating experience
- Preliminary Safety Information Documents (PSIDs) and subsequent NRC Preliminary Safety Evaluation Reports (PSERs)
- Relatively recent DOE laboratory studies of safety gaps

Past studies for SFRs include

- The PSID [15] and PSER [16] for the Power Reactor Innovative Small Module (PRISM) reactor design
- PRISM source term evaluations that have been published in PRISM licensing documents as well as public conferences
- The Final Safety Analysis Report (FSAR) for the Fast Flux Test Facility (FFTF)
- Recent DOE-sponsored efforts as part of the Advanced Reactor Technology program such as ANL-ART-3, ANL-ART-38 and ANL-ART-49 [17]

### 2.2.1. SFR Plant Characteristics

The SFR considered in this report follows the reference design described in ANL-ART-3 [18]. This is a pool-type design with metal fuel. This is illustrated in Figure 2-2

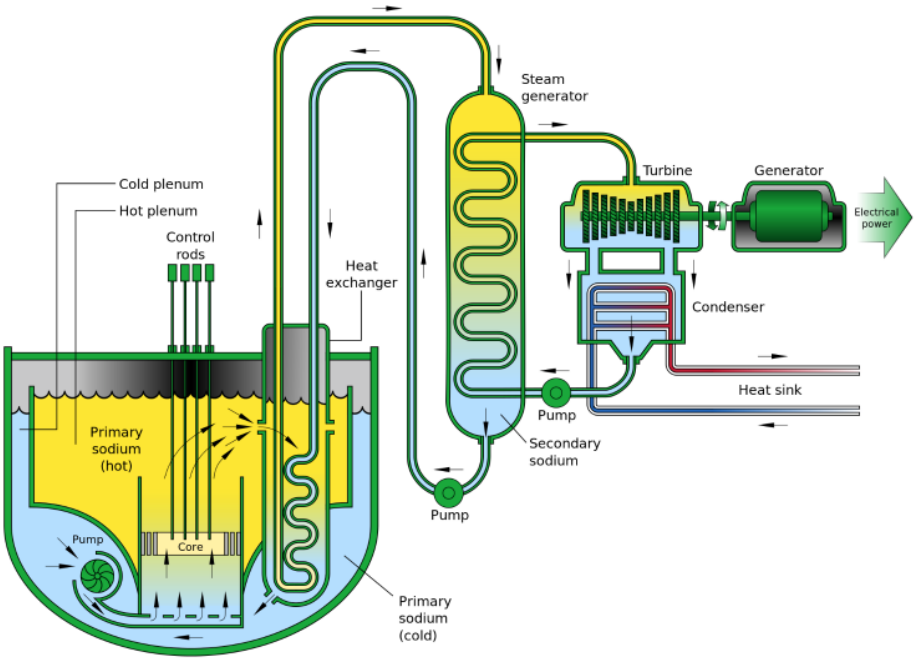


**Figure 2-2 – Reference Pool-Type SFR Design [19]**

The upper portion of the reactor vessel above the sodium pool is an inert cover gas. The reactor vessel is housed within a structure enclosing the reactor consisting of a

- Guard vessel around the actual reactor surrounding the reactor vessel and an atmosphere of inert gas
- Upper enclosure above the top of the reactor vessel

This is illustrated in Figure 2-3.

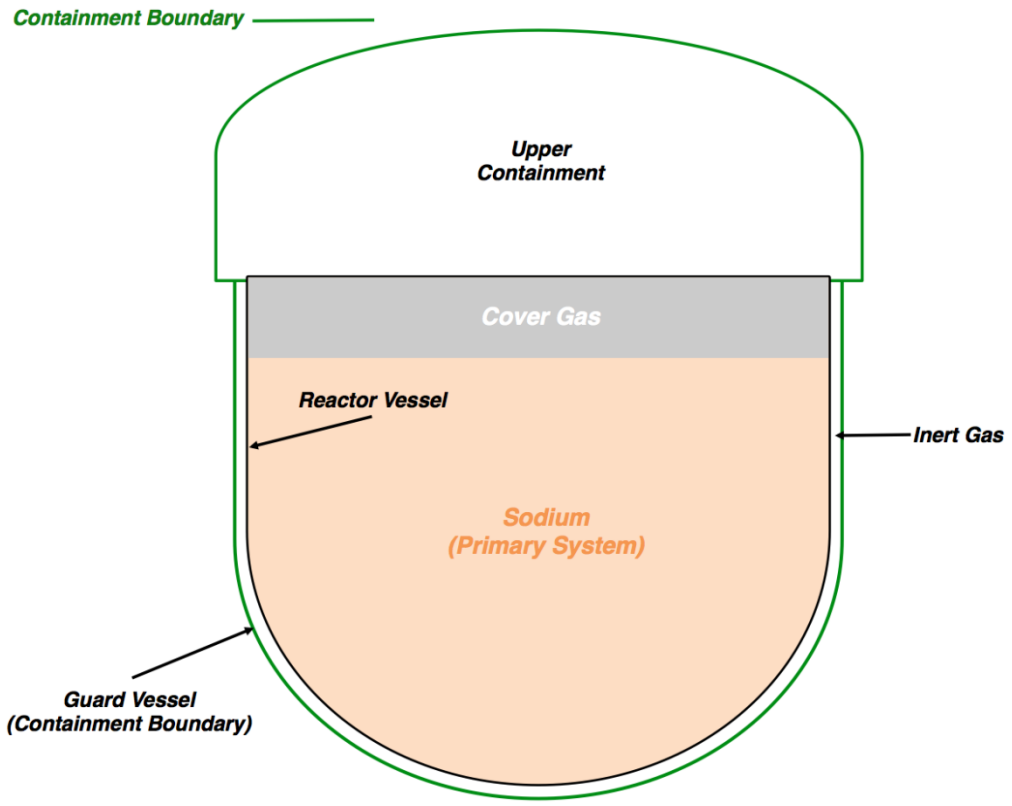


**Figure 2-3 – Illustration of Reference SFR Pool-Type Reactor Enclosure Boundary [18]**

The SFR reactor vessel and reactor enclosure would typically be located within a reactor building as illustrated in Figure 2-4.

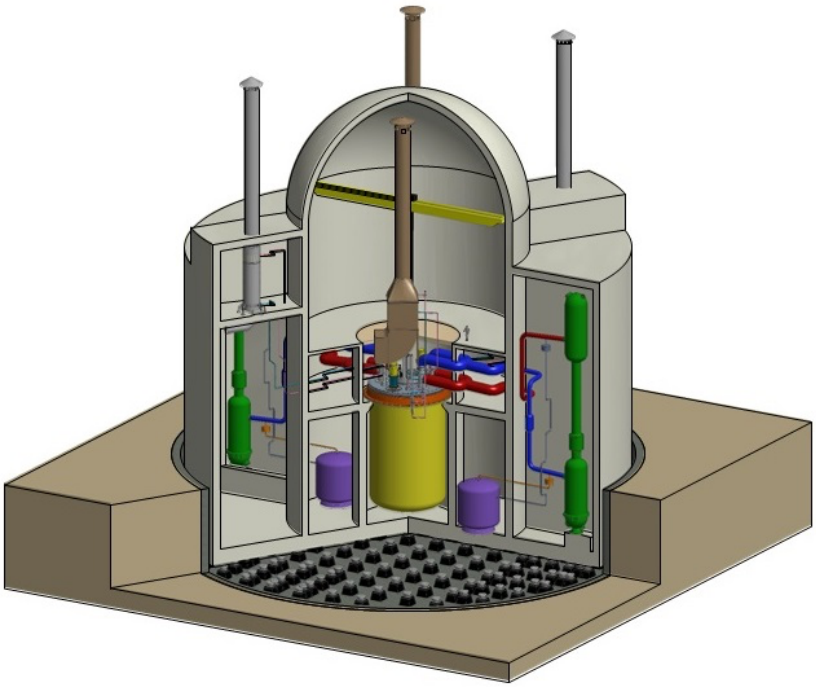

**Figure 2-4 – Illustration of SFR Reactor Building [20]**

### 2.2.2. SFR Release and Consequence Studies

Despite past U.S. experience with SFRs, none have been licensed by the NRC. The fast breeder reactor Fermi I was licensed by the Atomic Energy Commission (AEC). As a result, no source term analysis for an SFR has been conducted for a reactor that has ultimately been licensed by the NRC. Other SFR designs, specifically the Experimental Breeder Reactor II (EBR-II) and the Fast Flux Test Facility (FFTF), were built and operated by the DOE. Subsequent SFR development efforts progressed designs to various stages. However, none were ultimately licensed and constructed.

The Clinch River Breeder Reactor (CRBR) prepared a Preliminary Safety Analysis Report (PSAR) that received initial review from the NRC. This design used oxide fuel in a loop-type primary system arrangement. The source term assessment for this design used for siting purposes was based on TID-14844 [21], which assumed a large instantaneous release into the reactor enclosure. No retention mechanisms are credited in this approach. A focus during review of this design was the potential for energetic events to develop due to positive reactivity insertion events that could arise upon occurrence of fuel melting. These types of events have been termed Hypothetical Core Disruptive Accidents (HCDAs). For such an energetic event, vaporization of fuel together with significant energy release failing both the primary and reactor enclosure boundaries could give rise to substantial fission product release to the environment. The significant uncertainties associated with analyzing such energetic events resulted in regulatory challenges that contributed to cancellation of the CRBR.

Following CRBR cancellation, advanced SFR designs focused on metal fuel in a pool-type primary system arrangement. Two advanced SFR designs were proposed as part of the Advanced Liquid Metal Reactor (ALMR) project: the PRISM and the Sodium Advanced Fast Reactor (SAFR). The adoption of a pool-type reactor at atmospheric pressure removed the potential for high pressure reactor vessel failure; this was a key source of challenge to reactor enclosure boundary integrity in the earlier CRBR. The use of metal fuels furthermore introduced additional sources of negative reactivity feedback effects beneficial to mitigating the potential occurrence of an HCDA. Both the SAFR and PRISM designs submitted PSIDs to the NRC. Comments from the NRC on these PSIDs were communicated through PSERs.

The SAFR PSID reported a design basis source term that involved considerably less challenge to reactor fuel than traditionally assumed for LWRs. Based on the metal alloy fuel having enhanced heat transfer capabilities and reliable decay heat removal systems, it was argued that extensive fuel melting was incredible. Thus, melting of a single assembly was assumed to be the most credible degree of fuel damage leading to a bounding source term estimate. For the reported source term analysis, 100% of the assembly fission product inventory was assumed to be released into the sodium pool and no credit was allowed for retention of released fission products in the sodium. Subsequent to this fuel release, fission products were assumed to migrate into the reactor enclosure either instantaneously (i.e., a 100% instantaneous leakage rate out of the reactor vessel) or over a day (i.e., a 100% reactor vessel leakage rate over a day).

The PRISM PSID developed a source term that was intended to bound uncertainties in the calculation of a mechanistic source term. This bounding approach was adopted because of a) limited data regarding metal alloy fuel performance under accident conditions, and b) limited characterization of credible fission product release scenarios. As a result, the PRISM PSID source term assumed that total fuel damage occurred that included spent fuel. The inventory assumed was scaled from oxide fuel assuming limited fission product retention in the sodium. Release to the reactor enclosure was approximately instantaneous, similar to an HCDA scenario.

The NRC reviews of these PSIDs highlighted some areas for further consideration.

- The SAFR PSER highlighted the need for additional justification of fission product release fractions and retention factors. The more limited extent of fuel damage (i.e., limited to a single assembly) may not adequately capture scenarios where positive reactivity insertion could lead to more substantial core melting.
- The PRISM PSER noted that oxide fuel release fractions were not realistic for metal fuels. No detailed review of the assumed retention factors was performed as part of the initial phase of review.

### *2.2.3. SFR Inventory and Release References*

Table 2-3 and Table 2-4 provide a summary of how fission product inventory and release were considered in past SFR studies.

**Table 2-3: Inventory Estimates for SFRs**

| Power Level | Information on Inventory | Reactor/Program | Reference |
|---|---|---|---|
| 471 MWth (per module)<br>9 modules for the overall plant | Inventory elements not provided, but activity levels postulated for several design-basis events. | PRISM | [15] |
| 1000 MWth | Kr, Xe, I, Br, Cs, Rb, Te, Sb, Se, Ba, Sr, Ru, Rh, Pd, Mo, Tc, Co, La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, Ce, Pu, Np, U. | Reference SFR | [17] |

**Table 2-4: Release Estimates from SFRs**

| Power Level | Information on Releases | Reactor/Program | Reference |
|---|---|---|---|
| 425 MWth | Most severe core damage end state: 100% fuel melt, 10% vapor formation.[1,2] Primary sodium cold trap leak postulated. | PRISM | [15] |
| 1000 MWth | Two transient accident scenarios assessed (PLOF and UTOP); distribution of RNs in fuel pin analyzed; fuel failure times predicted; 100% RNs and 10% RNs released in bubbles considered. | Reference SFR | [17] |
| [1] Non-mechanistic analyses<br>[2] Judgements based on experimental work performed on FFTF oxide fuel and extrapolated to PRISM metallic fuel. | | | |

## 2.3. Molten Salt Reactors

Of the three reactor types being studied, MSRs lack substantial operational experience, detailed source term studies, and licensing reviews compared to HTGRs and SFRs. Most MSR facility safety insights stem from the Aircraft Reactor Experiment (ARE), the Aircraft Reactor Test (ART), and the Molten Salt Reactor Experiment (MSRE). Each of these projects generated a hazards summary report for consideration by the U.S. Atomic Energy Commission (U.S. AEC) and the MSRE generated and updated a detailed safety analysis. Both the ARE and MSRE were built and operated,

while the ART was canceled during construction. This section provides a brief review of MSR release and consequence studies and code assessments that may be leveraged to calibrate MSR release estimates and develop scaling laws. Each of the historical MSR hazard summaries also included a brief study of the effects of dispersal of the beryllium used as a moderator and reflector. The hazard from beryllium is due to its chemical toxicity and is not addressed in this report. Some aspects of this study were also informed by preliminary designs for the Molten Salt Breeder Reactor (MSBR).

**NOTE:** This report focuses on molten salt-fueled reactors. Other designs have been proposed where the molten salt is only used as the coolant, carrying away heat generated within solid fuel. In NRC non-LWR licensing strategy documents, such as Reference [22], these types of reactors fall under the class Fluoride Salt-Cooled High Temperature Reactor (FHR).

### 2.3.1. MSR Plant Characteristics

The MSR plant used for this study is the MSRE because, although it may not be representative of current proposed MSRs, it is the best-documented and tested MSR design. It operated from 1965 to 1969 and had a general layout as seen in Figure 2-5. The MSRE fuel salt was a lithium-beryllium fluoride salt with dissolved $UF_4$ and $ZrF_4$ [23, p. 7]. The fuel salt was pumped through a graphite core geometry which allowed criticality and then through a salt-salt heat exchanger within the reactor cell. Radionuclides were mostly contained in the fuel salt although a fraction of xenon and krypton were stripped in the primary salt pump and sent to an off-gas system to decay.

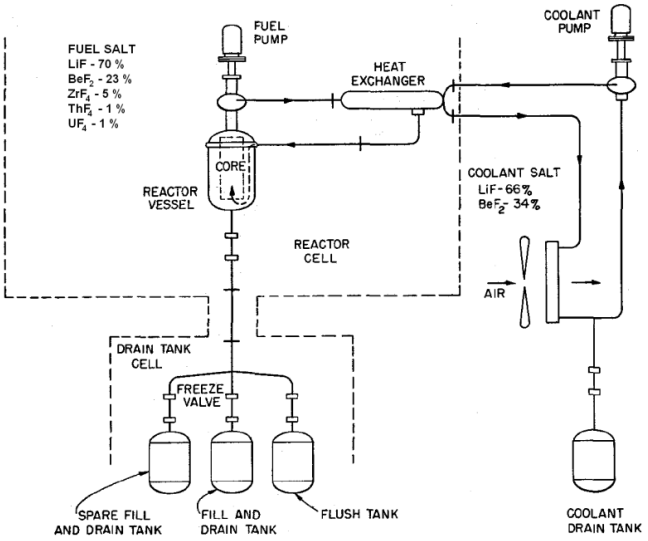


**Figure 2-5: MSRE Plant Layout [23, p. 6]**

When the primary loop required draining, freeze valves were opened to allow the entire fuel salt volume to drain into a drain tank. To travel from the reactor vessel to the drain tank, the fuel salt flows through a pipe between the reactor cell and the drain tank cell. The drain tank was cooled using steam thimbles to remove decay heat. Designers estimated that a dump from the full reactor power of 8 MWt would initially require 100 kW of cooling and would result in a peak fuel temperature of approximately 1400°F four hours after shutdown [24, p. 6]. A water reservoir provided cooling for approximately the first six hours after shutdown in the event of a coolant flow failure.

A break in the primary loop would result in fuel salt being spilled onto the floor of the reactor cell. The reactor cell itself was a cylindrical steel vessel [23, p. 178]. It sat within a larger cylindrical steel shield vessel and the annulus between the vessels was filled with sand and water. A large mass of reinforced concrete held the shield vessel. A sufficiently large spill of molten salt into the reactor cell would cause steam to form in the annulus and be vented into the coolant cell.

A loss of salt from the drain tank (see Figure 2-5) would result in a spill onto the floor on the drain tank cell. The bottom and sides of the drain tank cell were lined with steel [23, p. 182]. The structure of the cell was made of reinforced concrete. Water could be introduced to this spilled salt if the water reservoir or a steam thimble in the drain tank leaked. This is the main avenue for pressurization of the drain tank cell.

Both the reactor cell and the drain tank cell were rated to 40 psig and were kept inerted with a 95% nitrogen atmosphere [23, p. 239]. An over-pressurization of either cell (20 psig) would have opened rupture discs to the vapor-condensing system where steam would be condensed and non-condensable gases would be held up. The vapor-condensing system, which is shown in Figure 2-7, would route steam and gases from the cells through a tank of water to cool gases and condense steam. The vapor-condensing system was located outside of the reactor building but ultimately vented back to the building stack.

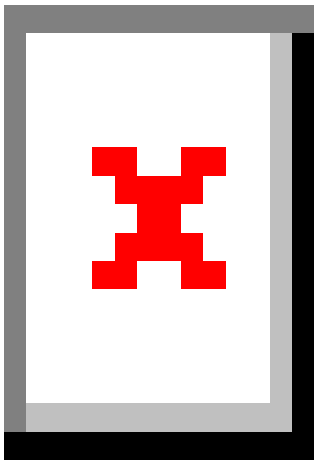

**Figure 2-6: MSRE Fuel Salt Drain Tank [23, p. 29]**

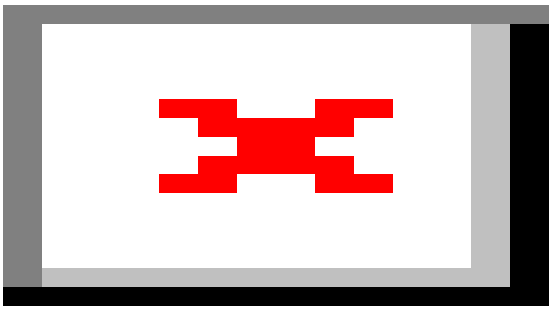

**Figure 2-7: MSRE Vapor-Condensing System [23, p. 190]**

### *2.3.2. MSR Release and Consequence Studies*

This section describes MSR release and consequence studies. Reports on the MSRE form a substantial portion of current understanding of potential MSR risk. These cover a wide range of engineering design information, including a preliminary hazards analysis report. This includes an updated and extended safety analysis [23]. Table 2-5 lists MSR accidents whose consequences have been evaluated.

**Table 2-5: Consequence Estimates for MSRs**

| Power Level | Information on Consequences | Reactor/Program Accident | Reference |
|---|---|---|---|
| 8 MWth | Airborne doses calculated for multiple locations for case of vaporization of all fuel salt. | MSRE Maximum Credible Accident | [23] |
| 2.5 MWth | Airborne and waterborne doses calculated for multiple locations for case of a reactor-equivalent weight of TNT exploding and dispersing all radioactive materials | ARE Ultimate Catastrophe | [25] |
| 60 MWth | Doses calculated for multiple locations for case of 1% of fuel leaking into NaK in heat exchanger | ART Heat Exchanger Leak | [26] |
| 60 MWth | Doses calculated for | ART Primary System | [26] |

| Power Level | Information on Consequences | Reactor/Program Accident | Reference |
|---|---|---|---|
| | multiple locations for case of all fuel leaking into reactor building cell | Leak | |

### 2.3.3. MSR Release and Consequence Code Assessments

There are currently no MSR specific codes to evaluate fission product releases from MSRs. The MELCOR code has introduced functionality to treat molten salt. Continuing work to update MELCOR in the near-term to enable evaluation of molten salt systems is summarized in Reference [22].

### 2.3.4. MSR Inventory and Release References

Table 2-5, Table 2-6 and Table 2-7 summarize fission product inventory and release insights from past MSR studies.

It should be noted that gaseous fission products are removed from the fuel salt in each of the historical MSR designs. Thus, an inventory is created in the off-gas system and a decrease of inventory in the fuel salt results. For conservatism, historical hazard reports considered this inventory to be available for dispersal in the limiting accident. After MSRE was shutdown, a mass balance identified an unaccounted loss of iodine and tritium from the system.

**Table 2-6: Inventory Estimates for MSRs**

| Power Level | Inventory Elements | Reactor/Program | Reference |
|---|---|---|---|
| 8 MWth | Inventory elements not postulated but activity levels are postulated. | MSRE | [23] |
| 2.5 MWth | Broad inventory categories given and activity levels are postulated. | ARE | [25] |
| 60 MWth | Inventory elements not postulated but activity levels are postulated. | ART | [26] |
| 236 MWth | Ag, Cs, Ce, Eu, I, Kr, Pd, Sr, Xe | Pebble Bed Fluoride Salt Cooled High-temperature Reactor (FHR)[1] | [27] |

[1] The retention of fission products in the molten salt shares common radiochemistry with molten salt-fueled systems.

**Table 2-7: Release Estimates from MSRs**

| Power Level | Information on Releases | Reactor/Program | Reference |
|---|---|---|---|
| 8 MWth | Releases into secondary containment (or reactor cell). 4,000 lbs of salt in 15 sec from recirculation line or 10,000 lbs of salt in about 370 seconds from drain line. 10,000 lbs | MSRE | [23] |

| Power Level | Information on Releases | Reactor/Program | Reference |
|---|---|---|---|
| | of salt in about 280 seconds from simultaneous breaks. | | |
| 2.5 MWth | Releases into secondary containment. Limiting accident assumes instantaneous dispersal of all salt into secondary containment. | ARE | [25] |
| 60 MWth | Releases into secondary containment. Limiting accident assumes all salt vaporized and buildings ruptured. | ART | [26] |
| 236 MWth | (The types of releases in [27] should not be applied due to fundamental differences in reactor fuel, reactor core design, and primary coolant thermal-hydraulics in accident scenarios. However, the fluoride salt chemistry insights can be applied, albeit with caution due to the presence of uranium and plutonium in the fuel salt) | Pebble Bed Fluoride Salt Cooled High-temperature Reactor (FHR) | [27] |

The ARE hazards summary report considers the release of overheated fuel with the simultaneous mixing of NaK and water in the heat exchanger pit to be the worst possible accident, or "Ultimate Catastrophe" [25]. This event releases 1.75E+9 cal and is assumed to release 6.5E+7 curies within 1000 seconds of initiation. The release is broken into categories in Table 2-8.

**Table 2-8: ARE Ultimate Catastrophic Source Term**

| Activity Category | Source Term (Ci) | Reference |
|---|---|---|
| Noble Gases | 3.2E+6 | [25] |
| Halogens | 3.6E+6 | [25] |
| Sr-90 | 80 | [25] |
| Total | 6.5E+7 | [25] |

The ART hazards summary report calculates that the energy necessary to raise the fuel to its boiling point and vaporize it is approximately 6E+5 kcal [26]. The resulting cloud is assumed to contain 3.3E+8 curies of activity, which is 50% of the equilibrium inventory of the reactor. A breakdown of the radionuclides assumed to be within the cloud is not given.

The MSRE hazards report suggests that 10% of solid fission products (6.8E+5 Ci) and 10% of iodine (2.5E+5 Ci) will be released from the salt to the secondary container in the maximum credible accident. It is further assumed that 50% of the iodine released from the salt plates out on the container walls. Regarding noble gases, 0.87E+5 Ci of xenon and 2.88E+5 Ci of krypton are expected to be released for a total noble gas release of 3.75E+5 Ci [23, p. 245].

The maximum activity within the reactor building was calculated to be 183 Ci at 1.1 hours with the fan on. If the fan is off, the maximum is 932 Ci at 4 hours. The activities of different radionuclide classes within the reactor building were calculated for times immediately following the accident. The calculations were repeated for the building fan being on and off as seen in Table 2-9 and Table 2-10, respectively.

**Table 2-9: Activity in Building with Fan On [23, p. 263]**

| Time After Accident (hr) | Fraction of Activity Released from Building per Day | Iodine | | Noble Gases | | Solids | |
|---|---|---|---|---|---|---|---|
| | | Release Rate (Ci/day) | Activity in Building (Ci) | Release Rate (Ci/day) | Activity in Building (Ci) | Release Rate (Ci/day) | Activity in Building (Ci) |
| 0.25 | 3.5x10-3 | 436 | 9.7 | 1308 | 29.1 | 2.4 | 53 |
| 0.5 | 5.4x10-3 | 675 | 15 | 2025 | 45 | 3.7 | 82 |
| 0.75 | 6.4x10-3 | 800 | 17.8 | 2400 | 53.4 | 4.4 | 97 |
| 1 | 6.8x10-3 | 855 | 19 | 2565 | 57 | 4.6 | 103 |
| 1.1 | 7.0x10-3 | 874 | 19.4 | 2622 | 58.2 | 4.8 | 105 |
| 1.2 | 6.8x10-3 | 855 | 19 | 2565 | 57 | 4.6 | 103 |
| 1.5 | 6.6x10-3 | 828 | 18.4 | 2484 | 55.2 | 4.5 | 100 |
| 2 | 5.8x10-3 | 729 | 16.2 | 2187 | 48.6 | 3.9 | 88 |
| 2.5 | 4.75x10-3 | 594 | 13.2 | 1782 | 39.6 | 3.2 | 72 |
| 3 | 3.6x10-3 | 455 | 10.1 | 1365 | 30.3 | 2.5 | 55 |
| 3.5 | 1.4x10-3 | 176 | 6.8 | 528 | 20.4 | 1 | 37 |
| 4 | 1.3x10-3 | 160 | 3.55 | 480 | 10.65 | 0.9 | 19.3 |
| 5 | 1.9x10-4 | 24.1 | 0.535 | 72 | 1.6 | 0.13 | 2.9 |
| 6 | 3.0x10-5 | 3.8 | 0.0845 | 11.4 | 0.24 | 0.02 | 0.46 |
| 7 | 4.7x10-6 | 0.6 | 0.013 | 1.8 | 0.039 | 0.003 | 0.071 |
| 8 | 7.2x10-7 | 0.1 | 0.002 | 0.3 | 0.006 | 0.0005 | 0.011 |

**Table 2-10: Activity in Building with Fan Off [23, p. 264]**

| Time After Accident (hr) | Fraction of Activity Released from Building per Day | Iodine | | Noble Gases | | Solids | |
|---|---|---|---|---|---|---|---|
| | | Release Rate (Ci/day) | Activity in Building (Ci) | Release Rate (Ci/day) | Activity in Building (Ci) | Release Rate (Ci/day) | Activity in Building (Ci) |
| 0.25 | $9.7x10^{-6}$ | 1.21 | 12.1 | 3.63 | 36.3 | 6.58 | 65.8 |
| 0.5 | $1.86x10^{-5}$ | 2.33 | 23.3 | 7 | 70 | 12.7 | 127 |
| 0.75 | $2.72x10^{-5}$ | 3.39 | 33.9 | 10.2 | 102 | 18.4 | 184 |
| 1 | $3.5x10^{-5}$ | 4.37 | 43.7 | 13.1 | 131 | 23.8 | 238 |

| Time After | Fraction of | Iodine | | Noble Gases | | Solids | |
|---|---|---|---|---|---|---|---|
| 1.5 | $4.9x10^{-5}$ | 6.07 | 60.7 | 18.2 | 182 | 33 | 330 |
| 2 | $6.0x10^{-5}$ | 7.48 | 74.8 | 22.4 | 224 | 40.6 | 406 |
| 2.5 | $6.8x10^{-5}$ | 8.54 | 85.4 | 25.6 | 256 | 46.4 | 464 |
| 3 | $7.4x10^{-5}$ | 9.3 | 93 | 27.9 | 279 | 50.5 | 505 |
| 3.5 | $7.6x10^{-5}$ | 9.49 | 94.9 | 28.5 | 285 | 51.6 | 516 |
| 4 | $7.9x10^{-5}$ | 9.88 | 98.8 | 29.6 | 296 | 53.7 | 537 |
| 5 | $7.86x10^{-5}$ | 9.84 | 98.4 | 29.5 | 295 | 53.5 | 535 |
| 8 | $7.8x10^{-5}$ | 9.71 | 97.1 | 29.1 | 291 | 52.8 | 528 |
| 10 | $7.7x10^{-5}$ | 9.63 | 96.3 | 28.9 | 289 | 52.4 | 524 |
| 16 | $7.5x10^{-5}$ | 9.4 | 94 | 28.2 | 282 | 51.1 | 511 |
| 24 | $7.25x10^{-5}$ | 9.08 | 90.8 | 27.2 | 272 | 49.4 | 494 |

# 3. ASSESSMENT APPROACH

In order to develop a framework to evaluate the capabilities of reactor designs to mitigate the release of fission products to the environment, it is necessary to consider:

- The physical processes that characterize transport of radionuclides released from fuel through various regions of a nuclear power plant until ultimately being released into the environment, and
- The range of operational, anticipated operational occurrences, design basis events and beyond design basis events that result in fission product release from radioactive fuel material.

This report considers only the evaluation of radiological release mitigation for fuel in a nuclear reactor. There are also conditions which lead to the release of radionuclides from radioactive fuel that has been discharged from a reactor (i.e., spent fuel). This report does not consider those types of events as the design characteristics for advanced reactor concepts is less well-defined.

The discussion presented below provides a basis for the structured release mitigation strategy evaluation approach proposed in this report. This structured approach generalizes the range of release mitigation measures that could be utilized in any reactor design into a number of factors reflecting the different barriers to fission product release. This approach adopts that formulated for DOE radiological facilities [1], [28]. It is adopted since it provides a high-level means of characterizing the barriers that intercept and remove fission products moving along a pathway discharging to the environment. In this manner, it is able to more readily adapt to a range of different reactor types. A release mitigation strategy can thus be assessed in terms of the contribution of various release mitigation measures limiting the overall fission product release to the environment. This is summarized in terms of the following multi-factor formula that correlates fission product release to the environment, of a radionuclide X ($R_E^X$), with retention by various barriers encountered generically across a range of reactor designs.

$$R_E^X = I_X\left(1 - f_d^X\right)\left(1 - f_{fs}^X\right)\left(1 - f_r^X\right)\left(1 - f_e^X\right)\left(1 - f_l^X\right)$$

where:

$R_E^X$ is the total release of radionuclide $X$ is given by the above expression

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

Fraction of radionuclide $X$ retained through dissolution in the fuel ($f_d^X$)

Fraction of radionuclide $X$ retained by fuel system structures enclosing the fuel ($f_{fs}^X$)

Fraction of radionuclide $X$ retained in the reactor heat removal system ($f_r^X$)

Fraction of radionuclide $X$ retained in the reactor enclosure ($f_e^X$)

Fraction of radionuclide $X$ retained along leakage pathways connecting the reactor enclosure with the environment ($f_l^X$)

This overall process of reduction of fission products being transported through multiple barriers interrupting a fission product transport pathway to the environment is illustrated in Figure 3-1. Note that the factors $f_e^X$ and $f_l^X$ correspond to the overall leak path factor identified in Figure 3-1. They

have been separated out in this context because of their relevance to generically treating a range of scenarios in which fission products released from fuel and the reactor system may or may not bypass the reactor enclosure.[1]

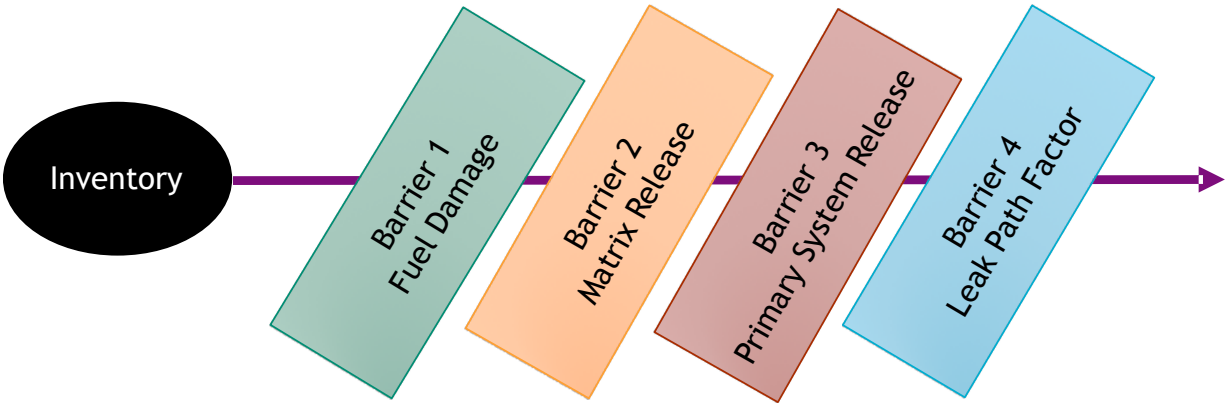


**Figure 3-1. Illustration of Source Term Reduction through Multiple Barriers to Fission Product Release**

The manner in which the different barriers can interface with each other is somewhat more complicated than illustrated in Figure 3-1. For example, in situations where a fraction of fuel cladding has been breached, the release from fuel will generally involve some fuel with the fuel clad bypassed. For that fraction, the fuel matrix provides the primary barrier to fission product release from the fuel. This is schematically illustrated in Figure 3-2.

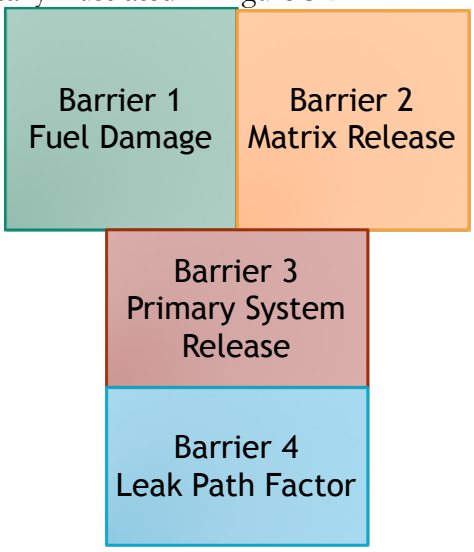


**Figure 3-2. Illustration of Impact of Fuel Clad Failure in Fraction of Fuel on Fission Product Transport Pathways into Primary System**

## 3.1. Phenomenological Considerations

Events that progress to unmitigated fission product release from radioactive fuel will not necessarily progress to substantial fission product release to the environment. The characterization of the magnitude of fission product release to the environment requires the evaluation of a number of physical processes and for some reactor designs, operator actions/inactions. These physical processes do not only relate to the transport of fission product vapors or aerosols, but also involve a

[1] Scenarios with a bypass of the reactor enclosure correspond to bypass scenarios in LWRs.

number of thermal hydraulic, thermo-chemical, and thermo-mechanical processes. The overall magnitude of fission product release is thus sensitive to the interplay of a number of different phenomena.

The evaluation of phenomena that influence how a reactor design responds to limit the extent of fission product release to the environment requires assessment of the following characteristics:

- Fission product release from fuel or fuel debris,
- Fission product transport through various regions of the plant and ultimately to the environment.

### 3.1.1. Fission Product Release

The magnitude of and radionuclide composition in a release from fuel or fuel debris depend on the phase of accident progression. Fission product release from fuel or fuel debris can generally be divided into release of the following fission product populations.

- Release of noble gases that migrate away from fuel material upon formation
- Release of fission products that have already moved past the surface of the fuel
- Release of fission products that are dissolved in the fuel

The first two populations represent the most mobile fission product. Across reactor designs, the magnitude of initial fission product release will be driven by fission products in these populations.

The third population of fission product gases are typically less mobile. These fission product gases must be transported to the surface of fuel to be released into interfacing fluid (either liquid or gas depending on the reactor design). The manner by which these fission products are transported to the fuel surface depends on the state of the fuel. In solid fuel systems, prior to significant core melting, fission product gases typically diffuse within the solid fuel and fuel matrix. In cases where fission product gases are dissolved in molten fuel, transport to fuel surfaces must consider additional transport mechanisms beyond thermally-assisted diffusion, such as:

- Convective transport of fission products within the bulk molten fuel
- Mass diffusion through boundary layers separating the molten fuel from the interfacing fluid
- Vaporization of fission products from the surface of molten fuel
- Transport due to sparging gas flows through molten fuel

Transport via diffusion is influenced by the temperature of the fuel; at higher fuel temperatures, these diffusing fission products have greater thermal energy driving their diffusive motion toward the surface of the fuel. Once reaching the surface of the fuel, these fission product gases move into the interfacing structure or fluid. Ultimate release to fluids interfacing with the fuel structure determines the magnitude of release out of the fuel.

### 3.1.2. Fission Product Transport

To frame a phenomenological discussion of fission product transport within the reactor module, and ultimately to the environment, it is useful to describe transport in terms of the overall progression of an accident. For water-moderated reactors, integral plant response codes are used to quantitatively evaluate fission product transport through the plant and to the environment. Performance assessment of engineered systems, physical barriers or inherent fission product

retention mechanisms is often performed by evaluating the fraction of radionuclides from the initial core inventory that is prevented from ultimately reaching the environment.

As an example, for Boiling Water Reactors (BWRs), the suppression pool is one mechanism by which fission products are removed from the containment atmosphere. Discharges into BWR suppression pools are through spargers that ensure discharge flows are broken into bubbles. Fission products in the discharge flows thus enter the pool in these bubbles as a sparging flow. A number of different processes contribute to the removal of these fission product aerosols in discharged bubbles, such as condensation of supersaturated steam.

The effectiveness of fission product scrubbing in BWR suppression pools is often expressed using a decontamination factor (DF)[2]. This factor represents the ratio of fission product mass remaining in the suppression pool to the total fission product mass entering the suppression pool. It is important to note that in this example, the BWR suppression pool does not have the same effectiveness under all conditions. Containment thermal hydraulic conditions have a critical impact on the performance of BWR suppression pools. If, for example, the suppression pool water temperature is near saturation, the DF for the suppression pool is relatively low (DF values less than 10). For a subcooled suppression pool, however, significantly more steam condensation occurs resulting in much more substantial fission product retention (DF values in excess of 100).

## 3.2. Accident Scenario Considerations

The release of fission products is typically limited under normal operating conditions for currently operating water-moderated reactors. Accident scenarios across different reactor types can be classified by the nature of critical safety function failures.

- Reactivity control – failure to maintain the generation of energy within the fuel at a rate capable of being removed by the heat removal system (either active or passive)

  These events can be particularly challenging from the perspective of fission product release and transport. Because damage to fuel occurs with the reactor at elevated power, there is a much greater amount of energy in flows that transport fission products away from the fuel into the heat removal system, the reactor enclosure and potentially the environment. These types of events often exhibit earlier fission product release to the environment and typically larger release magnitudes.

- Heat removal – failure of the heat removal system that is designed to carry power generated within the fuel to an ultimate heat sink

  The accident scenario plays a critical role in determining the amount of radioactive fuel material that is damaged and from which fission products can be released. In the DOE fuel cycle facility source term evaluation methodology [1], this is referred to as the fraction of material at risk.

  As an example, consider an accident in a Pressurized Heavy Water Reactor (PHWR) in which a feeder stagnation flow blockage occurs. In this scenario, a single fuel channel experiences an interruption of coolant flow while the reactor remains at power with the primary system at full system pressure. The severe mismatch between power generation and heat removal in the

[2] The decontamination factor is a quantitative metric that provides the ratio of radiological contamination in an effluent entering a volume to the radiological contamination remaining in the effluent after leaving a volume. This gives an estimate of the effectiveness of various radiological scrubbing measures at removing fission products from an effluent. The higher the decontamination factor, the more effect a scrubbing measure is at removing fission products from an effluent stream.

affected fuel channel leads to fuel heatup and eventual melting. Impingement of molten fuel on the pressure tube leads to rupture of the fuel channel. With safety systems available, fuel in other fuel channels remain undamaged. This type of single channel event releases a much smaller fraction of fission products than an event which affects the majority of fuel in the core. It serves as an example of how different reactor designs can require design-specific attention regarding the fraction of fuel that is affected by a loss of cooling function.

- Reactor enclosure boundary – failure of the boundary that divides the plant from the environment

  The available mitigation measures to prevent or ameliorate potential fission product releases to the environment are captured through definition of the characteristics of the reactor enclosure boundary performance. For example, the effectiveness of scrubbing systems such as the BWR suppression pool, the availability of a filtered vent system, and the operation of sprays that would remove fission product vapors from the enclosure atmosphere are examples of performance characteristics. The potential for bypass of the reactor enclosure boundary is also relevant. Should an open pathway exist from the reactor, there will be limited to no attenuation of fission product release in the reactor enclosure.

## 3.3. Generalized Characterization of Radiological Release Mitigation Strategies

The characteristics of fission product release are typically defined in terms of

- The magnitude of radionuclide release
- The timing of radionuclide release

The magnitude of radionuclide release usually focuses on the quantity of specific radionuclide releases in an event.

In addition to release magnitude, timing is critical to assessment of off-site emergency mitigation measures. The current release phases considered for LWRs factor in the general timing of LWR accident progression. For example, gap release is expected to occur relatively early in an event that challenges fuel cladding integrity. In-vessel release occurs somewhat later once fuel melting commences. Ex-vessel releases, however, only occur once fuel debris breaches the reactor vessel lower head and discharges into containment. These release phases primarily reflect the distinct nature of radionuclide release. Gap release is principally dominated by noble gases, while in-vessel release is characterized by more substantial release of volatile fission products (such as iodine or cesium) from the fuel. Ex-vessel release typically leads to release of the less volatile fission products from the fuel debris due to phenomena associated with molten core-concrete interaction (MCCI).

This characterization of release phases is not possible in a generalized, technology-inclusive approach. However, it is reasonable to represent releases such that issues related to reactor siting and the exclusion area boundaries are taken into consideration. In the discussion of off-site consequences, the focus will be on presentation of off-site dose as a function of distance from the site. This is most consistent with the evaluation of consequences in the LMP, which considers dose at the exclusion area boundary as the primary consequence metric.

For the purpose of using a scoping source term to estimate off-site dose consequence, it is reasonable to restrict consideration to the release magnitude. Ignoring release timing introduces a level of conservatism in the evaluation since no credit is taken for radioactive decay. However, for non-LWR concepts, delayed progression to fission product release from damaged fuel presents an

important release mitigation measure. When relevant, accident progression timing is mentioned below in a qualitative manner.

### 3.3.1. Initial Fission Product Inventory

The available inventory for release can be significantly lower when dealing with a reactor that is lower power than typical operating LWRs. For sufficiently small initial inventories, dispersion in the environment could reduce the off-site dose appreciably relative to that for a higher power reactor. This may be a factor when evaluating the appropriate exclusion area boundary.

### 3.3.2. Components of Fission Product Release from Fuel

The release of fission product from a fuel system can be generally characterized into the following components.

1. Release from fuel of undissolved fission products (i.e., noble gases)
2. Release of fission products dissolved in fuel material
3. Release of fission products generated in fuel system structures enclosing fuel
4. Transport of activated materials from fuel system structures into interfacing fluid
5. Release through the boundary between fuel system and interfacing fluid (e.g., release due to clad failure in water-moderated reactor designs)

- Limited transport through interfacing boundary (e.g., prior to clad failure in in water-moderated reactor designs)
- Diffusion-limited transport through interfacing boundary
- Unmitigated fission product gas transport through interfacing boundary

The first two fuel release modes are typically referred to as matrix release for water-moderated reactors.

The third fission product release mode reflects situations where fuel particles have contaminated encapsulating structures. Under irradiation, fission product gases will be generated.

The fourth fission product release mode reflects situations where irradiation of fuel structures generates activated material that can be transported into the interfacing fluid.

The final release regime is governed by the integrity of barriers separating the fuel system from the interfacing working fluid that removes heat generated within fuel.

In the case of water-moderated reactors, the physical barrier that separates the fuel matrix from the interfacing coolant is the fuel rod cladding. For High Temperature Gas-Cooled Reactors (HTGRs) using TRISO fuel, the nature of this cladding barrier is different, consisting of a set of composite material layers. TRISO fuel is formed as a matrix packed with TRISO particles. A TRISO particle consists of an inner fuel kernel (made either of UCO or $UO_2$) which is coated in three layers of material (an inner porous carbon buffer followed by a silicon carbide layer and finally an outer pyrolytic carbon layer). TRISO particles are packed into a graphite matrix. There are thus a number of structural barriers in a TRISO fuel system that impede motion of fission product gases released from the fuel out of the fuel system.

HTGRs present additional complications. It is possible for fissile material to migrate out of the TRISO particles into the fuel pebble. This is captured by the third fission product release mode. Finally, the fourth release mode captures the generation of graphite dust that is readily transported throughout the reactor heat removal system.

In molten fuel systems, where the radioactive material is carried within a circulating molten fuel, the release of fission products occurs from free surfaces of the molten liquid into the cover gas. As in the case of solid fuel systems, fission products are either in gaseous form or dissolved in the molten salt.

For these systems, the noble gases are continually released into the cover gas. Unlike the solid fuel system, the lack of a fuel cladding allows these gaseous fission products to be continually released into the cover gas as they are generated in the molten fuel. These reactors typically require additional systems that collect these gaseous fission product releases into the cover gas during normal operation. In the event of an accident where a cover gas purification system is lost, the noble gases in the molten salt will be released relatively quickly into the cover gas space. The accumulated inventory in a purification system may also be available to be rapidly released into the containment in the event of an accident.

The release of fission products that are dissolved in the molten fuel involves a number of different effects.

- Entrainment of contaminated molten salt droplets in the gas flows
  In this process gas bubbles flowing through the molten fuel reaching the surface will burst. The contaminated droplets formed will be entrained in gas flows within the cover gas.
- Vaporization of fission products from the molten fuel
  Fission products in the molten fuel have a vapor pressure that can be used to estimate the partial pressure of fission products in the cover gas above the molten fuel. As an example, assuming a radiological chemical compound $X$, its vaporization out of the molten salt is represented by the chemical process

$X(salt) \rightarrow X(gas)$

The equilibrium constant for this chemical reaction can be expressed as

$$K_{eq}(T) = \frac{P_X}{[X(salt)]\gamma_X}$$

where $P_X$ is the partial pressure of $X$ in the cover gas, $[X(salt)]$ is the concentration of $X$ in the molten fuel and $\gamma_X$ is the activity coefficient of $X$ in the molten fuel.

### 3.3.3. Fission Product Transport Regimes

Once fission product gases are released from the fuel system, they enter an interfacing fluid. Transport through this fluid influences the fission product inventory entering the enclosure volume around the reactor system. Fission product gases in the atmosphere of the heat removal system can readily move through leaks or openings into the reactor enclosure. The transport within the reactor system enclosure determines the atmospheric mass of fission products available to be released to the environment, or potentially effluent scrubbing systems.

The following are mechanisms by which fission product gases can be removed from a gas atmosphere within the heat removal system.

- Condensation/deposition of fission product gases on liquid pools in the heat removal system
- Deposition of fission product gases on heat removal system structural surfaces

Fission product gases in the heat removal system can evolve into the atmosphere of the heat removal system through the following mechanisms.

- Vaporization of fission products off liquid pools in the heat removal system
- Resuspension processes that move deposited fission product gases from structural surfaces to the heat removal system atmosphere

A similar set of processes occur that determine the amount of fission product gases remaining in the atmosphere of the reactor enclosure volume. Within this volume, however, it is necessary to consider aerosol formation and growth through condensation/evaporation/agglomeration mechanisms.

## 3.4. Identification of Fission Product Release Barriers

Nuclear power plant safety design is based on a concept of multiple levels of defense against fission product release to the environment. These levels of defense are referred to as barriers – they prevent the transport of fission products from one region of the plant to another. These can be physical barriers, like vessel walls. Alternatively, barriers can also be physical processes that limit the degree to which radionuclides can move from one region of the plant to another.

As an example, in an LWR, the fuel cladding serves as a barrier to release of fission products from fuel rods. When the fuel cladding fails, fission products can more readily leave the damaged fuel rod. Depending on the nature of the event, fuel cladding failure may be localized to a single fuel rod or occur across multiple fuel assemblies. In many beyond design basis accidents, the extent of damage to the core is such that the majority of fuel assemblies experience fuel cladding failure.

As an additional example, in a Pressurized Heavy Water Reactor (PHWR), with fuel arranged in discrete fuel bundles within an array of horizontal fuel channels, fuel cladding still serves as the first barrier to fission product release.

In both of the above examples from current production reactor designs, the extent of fission product release is also affected by the fraction of fuel that experiences conditions which can cause cladding failure. In the assessment of Department of Energy (DOE) fuel cycle facility source terms, this characteristic of fission product release is normally referred to as the damage ratio or the fraction of affected radioactive material [1].

Beyond the fuel cladding there are additional barriers that impede the motion of radionuclides into the environment. The evolution of fission products from reactor fuels can typically be characterized in the following manner.

Some classes of fission products can evolve as gases and do not remain bound in the nuclear fuel. A key example of these fission product gases are the noble gases such as Xe and Kr. In typical solid fuel systems for water-moderated reactors, these gases migrate to the gap between the fuel pellet and the fuel cladding. Failure of the fuel cladding results in immediate release of this gap inventory.

Other fission products remain bound within the nuclear fuel. These classes of fission products can only be released from fuel through a process of mass transport. In solid fuel systems, mass transport occurs through diffusion within the solid fuel matrix, such that migration of these fission products out of fuel is driven by diffusion through the fuel matrix and out through damaged cladding. In molten-fueled systems, fission products dissolved in the fuel will be convected within the bulk pool toward the surface of the molten pool in contact with cover gas. Diffusion across the surface boundary and ultimately vaporization allow these dissolved fission products to move into the cover gas. Alternatively, fission products dissolved in molten fuel can also be entrained by rising bubbles and transported into the cover gas when these bubbles leave the molten pool and burst.

The temperature of the fuel strongly determines the rate at which different classes of these fission products can be transported out of the fuel into interfacing fluid. Those that are able to leave solution and are transported into interfacing fluid at lower temperatures are typically referred to as volatile fission products. Classes having low transport rates out of solution into interfacing fluid, however, tend to have low diffusional release rates. These are classified into two broad categories of semi-volatile and non-volatile fission products.

Once released from the fuel, fission products can be held-up either in the atmosphere of the heat removal system or deposited on reactor structures. Those remaining in the atmosphere of the heat removal system can leak into the reactor enclosure, or potentially the environment through bypass flow paths from the reactor to the environment. Fission products transported into the reactor enclosure can be held-up in the atmosphere or deposited on structures. The leakage of fission products in the containment atmosphere to the environment may or may not be attenuated due to

- Deposition in small cracks in the containment structure through which leakage occurs [29] [30]
- Retention in a scrubbing system through which containment effluent must first pass (e.g., a filtered vent)

In order to characterize the inherent capabilities of different reactor systems to prevent release of radionuclides to the environment, it is useful to identify the following retention fractions.

- Fraction of fission product inventory retained through dissolution in the fuel ($f_d$)
- Fraction of fission product inventory retained by fuel system structures enclosing the fuel ($f_{fs}$)
- Fraction of fission product inventory retained in the reactor heat removal system ($f_r$)
- Fraction of fission product inventory retained in the reactor enclosure ($f_e$)
- Fraction of fission product inventory retained along leakage pathways connecting the reactor enclosure with the environment ($f_l$)

In the case of a water-moderated reactor, these different retention volumes correspond to the following fission product retention barriers

- Fuel matrix
- Fuel cladding (in most cases this corresponds to the fraction of the fuel that has been damaged as a result of the postulated accident)
- Primary system
- Containment
- Leakage path scrubbing mechanisms such as filtered vents or natural deposition in unintentional leakage pathways (e.g., small cracks in containment structures)

These different factors can be related to the fission product inventory that escapes to the environment. If $I_X$ is the initial fission product inventory at the time of the reactor accident for radionuclide $X$, then the total release of radionuclide $X$ is given by the expression

$$R_E^X = I_X\left(1 - f_d^X\right)\left(1 - f_{fs}^X\right)\left(1 - f_r^X\right)\left(1 - f_e^X\right)\left(1 - f_l^X\right)$$

This expression does not provide a means to mechanistically calculate a source term for different reactor types. Each factor in the expression is not a priori known for any accident scenario. As discussed above, there are a number of different physical processes that contribute to identifying the magnitude of any one of these factors. As with existing LWR source term analyses, an integral plant response code such as MELCOR is often used to quantitatively evaluate the magnitude of

radionuclide release to the environment. Such mechanistic source term analyses are normally interpreted using the above expression to better understand for a specific accident scenario the magnitude of fission product release from the fuel, the regions of the plant where fission products are retained and the role of any engineered system in scrubbing fission products.

In this sense, the usefulness of the above expression is not in its ability to quantitatively evaluate a reactor source term. Rather, its value is in organizing the information from supporting mechanistic calculations to enable a structured assessment of a particular reactor's design features that are critical to limiting the extent of radiological release to the environment. In this manner, it is possible during design or a subsequent regulatory review to establish the most critical elements of a reactor concept in reducing off-site consequence, as well as better resolve areas where knowledge gaps may impact conclusions about a design's robustness.

To illustrate how this type of simplified structure might be used to assess a new design's capabilities to limit the extent of radiological release to the environment, consider an example comparison of a BWR and negative containment pressure PHWR.

In a BWR, the factor $f_e^X$ would typically be large for non-noble gas radionuclides. This reflects the role of the suppression pool in scrubbing fission products from the containment atmosphere. This factor would tend to be significantly lower in accident sequences progressing to fission product release from the drywell (i.e., an unfiltered release pathway).

By comparison, a negative pressure PHWR relies on a Vacuum Building to provide holdup of fission products with some scrubbing due to sprays. Like the containment in these designs, the Vacuum Building is not designed to maintain large positive pressures. Thus, a filtered vent consisting of a charcoal bed and HEPA filter is provided to scrub radiological contamination from effluents. The filtered vent is used to prevent over-pressurization of the Vacuum Building and containment. A Vacuum Building spray system is provided to remove aerosols from the Vacuum Building atmosphere before venting to the environment. Once the Vacuum Building sprays have been initiated, however, the water source is exhausted so long-term discharges through the filtered vent will not benefit from any enhanced retention within the containment envelope. In this design, the factor $f_l^X$ is larger reflecting the role of contaminated release scrubbing as a key means of limiting the extent of off-site consequence. Typically, this factor is more important to determining the extent of off-site release than retention within containment $f_e^X$.

The actual evaluation of the magnitude and composition of a source term for a given accident scenario, however, relies on integral plant analyses using a code like the Modular Accident Analysis Program for CANDU (MAAP-CANDU). In fact, the importance of $f_l^X$ relative to $f_e^X$ may not be the same across all accident scenarios. In the event of a LOCA with fuel failures mitigated by Emergency Core Cooling, the initial release from fuel will be transported with the discharged steam from the primary system. The resulting pressurization of the containment will result in the Vacuum Building ducts opening and contaminated steam being discharged into the Vacuum Building. The sprays will tend to scrub non-noble gaseous fission products from the atmosphere and thus reduce the contamination discharged through the filtered vent pathway. In this type of scenario, the role of the filtered vent is somewhat less significant than noted above.

However, in the event of loss of heat sinks scenario for this negative pressure PHWR, the Vacuum Building ducts will open prior to any contaminated steam discharge into containment. This occurs due to heatup of the containment atmosphere due to heat losses from the PHWR primary system.

Upon opening of the Vacuum Building ducts, sprays will be initiated and all spray water will be exhausted. By the time fission product release to the containment occurs in this scenario, sprays will not be available. Thus, the importance of filtered venting is greater for this type of scenario because of how it challenges the assumed safety requirements for the spray system. This type of nuanced plant response is revealed by integral plant response analyses used to calculate mechanistic source terms; however, application of the above equation can still prove helpful to structuring information provided by a mechanistic calculation for the assessment of design performance.

This illustration provides the context for application of this type of simplified barrier approach. This also illustrates that the simplified barrier approach can offer insight into the effectiveness of radiological retention mechanisms as a function of accident scenario.

## 3.5. Additional Considerations for Non-LWRs

The above discussion focused on evaluating how a design ameliorates potential radiological release to the environment. From accident scenarios to phenomenology, the quantitative evaluation of a source term must account for interplay among several factors. The state-of-knowledge, not discussed above, is also a critical factor in assessing the capability of a design to mitigate the extent of off-site consequence.

In this section, the impact of key gaps in knowledge with respect to characterizing potential accident scenarios and phenomena for non-LWRs is briefly discussed. These key gaps must be considered prior to assessing the effectiveness of a range of release mitigation strategies of a reactor design.

### *3.5.1. Understanding of Accident Scenarios*

As noted above, systems or processes in a design that mitigate off-site consequences have variable efficacy depending upon accident scenario. The range of accident scenarios that can lead to potential off-site release for LWRs has been studied over several decades.

Knowledge of and experience with current LWRs gave way to the inherently robust and passively safe advanced reactor design concepts. In some cases, this has been achieved by fundamentally changing typical LWR design characteristics, such as the operation of the reactor system with a working fluid at low pressure. In other cases, this has been achieved through the introduction of inherent safety features; from exploiting negative reactivity feedback effects to utilization of passive heat removal mechanisms to prevent temperature excursions that could lead to fuel damage.

From this perspective, non-LWR concepts are generally robust (or have significant safety margin) with respect to internal failures of components of the engineered system. Thus, the risk associated with these designs, when evaluated with an internal events Probabilistic Risk Assessment (PRA), is generally low relative to the current operating fleet of LWRs. However, the risk profile associated with non-LWR concepts has not been exhaustively studied. Since many involve inherent safety features for design robustness, risk will likely be associated with challenges to the functionality of these safety features.

While such challenges may not be evident from an internal events PRA, they may become relevant when considering other hazards—from external events (such as seismic, flooding, high winds, etc.) to malicious acts. An exhaustive survey of event scenarios has generally not been considered across the range of advanced reactor design concepts. Thus, the true risk spectrum necessary to support risk-informed decision-making is generally not well-known at present for these non-LWR concepts. However, assessing the manner in which designs ameliorate off-site consequences through a

structured consideration of barriers, as provided above, will be useful in identifying where off-site consequences could emerge due to a challenge to critical release mitigation features of a design.

### 3.5.2. Phenomenological Considerations

The change in the working fluid, whether it is a solid or molten fuel system, introduces considerably new radiochemistry that may either limit or potentially enhance the release of fission products from a molten fluid. Two examples of this are presented below.

**Retention of Molecular Iodine in NaCl Molten Salt Systems**

The vaporization of fission products from the molten fuel must also account for a number of chemical reactions between radionuclides and the molten fuel. Consider as an example a molten fuel system that uses $NaCl$ as the molten salt. In this case, fission products such as iodine may be preferentially retained in the salt through the chemical reaction

$$I_2(salt) + Cl^- \rightarrow I_2Cl^-$$

**Fission Product Release from NaCl Molten Salt System in the Presence of a Radiation Field**

The above consideration of vaporization from a molten pool assumed only a thermally driven release of radionuclides. In a radiation field, additional chemical reactions occur that may lead to vaporization of fission products that are otherwise non-volatile at the temperatures of interest. Consider the following set of reactions in an $NaCl$ salt system that would lead to enhanced release of Ru into the cover gas.

$$Cl^- + (\gamma \; or \; n^0) \rightarrow Cl + e^-$$

$$Cl + Cl^- \rightarrow Cl_2^-$$

$$2Cl_2^- \rightarrow Cl_3^- + Cl^-$$

$$Cl_3^- \leftrightarrow Cl_2(salt) + Cl^-$$

The reaction of $Ru$ with $Cl_2(salt)$ could lead to the evolution of $RuCl_2(gas)$ through the following example reaction

$$Ru(salt) + Cl_2(salt) \rightarrow RuCl_2(gas)$$

Thus, radiochemistry is an essential area where knowledge must be enhanced. As noted above, while fission product retention in a working fluid different from water may be enhanced, there are also potential phenomena that could lead to increased transport of radionuclides out of the working fluid of the reactor heat removal system.

## 3.6. Characterization of LWR Source Terms

The following discussion provides an example of how LWR source terms can be characterized in terms of the barrier approach. Since there has been significant evaluation of LWR fission product release and transport phenomenology, an adequate technical basis is available to establish reasonable estimates for the extent to which individual barriers reduce fission product release to the environment.

### 3.6.1. Fuel Cladding Barrier

The fuel cladding barrier becomes ineffective under scenarios where clad failure has occurred. Under many accident scenarios arrested at clad failure, only a fraction of the fuel rods in the core experience cladding failure. This factor, typically termed the fraction of fuel at risk, is not considered further in this example, for simplicity. Beyond the fuel cladding, fission products are retained within the fuel matrix; that is, migration of fission products out of the fuel matrix toward the region of failed cladding typically occurs with higher mass transfer resistance at lower temperatures. The role of the fuel matrix is considered further below. Thus, for this example of fuel cladding failure, there is no effective reduction of fission product release due to the fuel cladding barrier. The following set of reduction factor complement tables are provided for completeness. All reduction factor complements provided in Table 3-1, Table 3-2, and Table 3-3 are zero, indicating the ineffectiveness of the fuel cladding barrier for accidents progression to fuel cladding failure.

This is shown in Table 3-1 for accidents progressing to fuel cladding failure.

**Table 3-1. Fuel Cladding Reduction Factor Complements for Cladding Failure LWR Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0 |
| Halogens (I, Br) | 0 | 0 |
| Alkali Metals (Cs, Rb) | 0 | 0 |
| Tellurium Group (Te, Sb, Se) | 0 | 0 |
| Barium, Strontium (Ba, Sr) | 0 | 0 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0 |
| Cerium Group (Ce, Pu, Np) | 0 | 0 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0 |

Fuel cladding failure accidents can progress to more severe core damage if decay heat removal is not adequately restored prior to initiation of core degradation. As long as the decay heat removal function is restored prior to challenge to the RPV lower head, as occurred at TMI-2, such core degradation accidents can be arrested with core materials retained within the RPV. Fission product release from the fuel is ultimately not affected by the fuel cladding for these scenarios, for similar reasons to accidents progressing to fuel cladding failure. Releases from the fuel materials, however, are mitigated by retention of fission products within the degraded fuel material or matrix. This is discussed further below. As a result, for these accident scenarios, fuel cladding plays no role in reduction of fission product release from fuel. This is shown in Table 3-2.

**Table 3-2. Fuel Cladding Reduction Factor Complements for In-Vessel LWR Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0 |
| Halogens (I, Br) | 0 | 0 |
| Alkali Metals (Cs, Rb) | 0 | 0 |
| Tellurium Group (Te, Sb, Se) | 0 | 0 |

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Barium, Strontium (Ba, Sr) | 0 | 0 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0 |
| Cerium Group (Ce, Pu, Np) | 0 | 0 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0 |

Accidents progressing to extensive in-vessel core damage have the potential to challenge the integrity of the RPV lower head. Upon breach of the RPV lower head, degraded core materials that have accumulated in the lower head will be able to relocate into the containment. Such accident scenarios progress to a state of ex-vessel damage. This is the general state of damage encountered at all three damaged Fukushima Daiichi units, to varying degrees. Fission product release from the fuel is ultimately not affected by the fuel cladding for these scenarios, for similar reasons to accidents progressing to fuel cladding failure and in-vessel damage. Releases from the fuel materials, however, are mitigated by retention of fission products within the degraded fuel material or matrix. This is discussed further below. As a result, for these accident scenarios, fuel cladding plays no role in reduction of fission product release from fuel. This is shown in Table 3-3.

**Table 3-3. Fuel Cladding Reduction Factor Complements for Ex-Vessel LWR Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0 |
| Halogens (I, Br) | 0 | 0 |
| Alkali Metals (Cs, Rb) | 0 | 0 |
| Tellurium Group (Te, Sb, Se) | 0 | 0 |
| Barium, Strontium (Ba, Sr) | 0 | 0 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0 |
| Cerium Group (Ce, Pu, Np) | 0 | 0 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0 |

### *3.6.2. Fuel Matrix*

In the case that the fuel matrix is challenged, LWR accidents evolve in such a way that the fuel cladding would be already breached. As a result, the fuel cladding does not serve to reduce the amount of fission product release. The reduction factor $f_{fs}^{X}$ for fuel structures (i.e., fuel cladding) is 0 so that the only reduction of fission product release into the reactor vessel is through retention in the fuel/fuel debris either inside or outside the vessel.

For simplicity, it is assumed that the effectiveness of the reactor vessel at retaining fission products is limited. As shown in Figure 3-3 and Figure 3-4, iodine and cesium may be retained in the long-term at fractions of 10% and 15%, respectively, within the reactor vessel. As noted in the Surry SOARCA [32], fission products released from the fuel are primarily swept into the containment. Thus, the reactor vessel reduction factors are assumed to be 0 in this simplified example. The

reduction factors presented below for fuel matrix retention, however, are based on information presented in [31] that has been extracted from NUREG-1465 [33]. These reduction factors do implicitly consider retention inside the reactor vessel as they are intended to provide an estimate of the release into containment. However, given the role of the reactor vessel, they are reasonable estimates of the overall effectiveness of the fuel matrix at mitigating release of fission products.

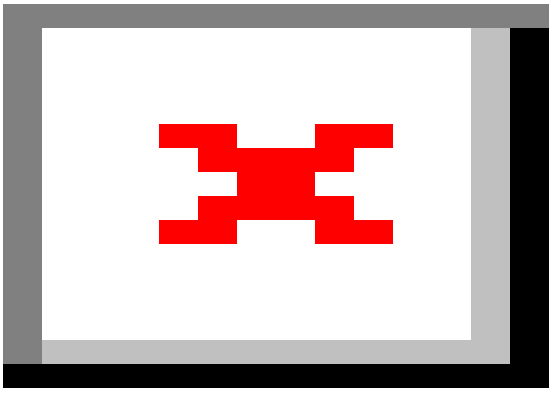

**Figure 3-3. Illustration of Iodine Distribution History for Surry Unmitigated Long-Term Station Blackout [32]**

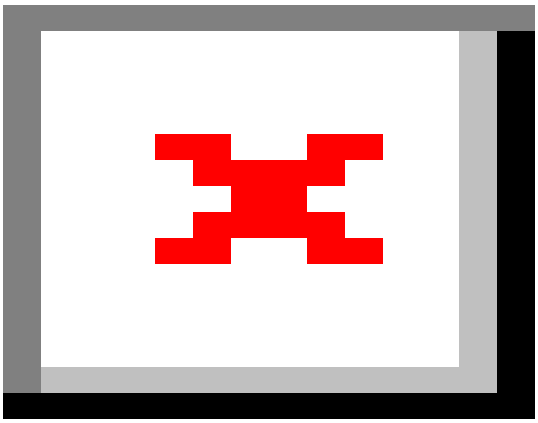

**Figure 3-4. Illustration of Cesium Distribution History for Surry Unmitigated Long-Term Station Blackout [32]**

For events that have been arrested at fuel cladding failure, the fuel matrix is effective at limiting fission product release from the fuel considerably. For noble gases, halogens, and alkali metals, about 95% of the fission product inventory is expected to be retained within the fuel (i.e., only 5% of the fission product inventory is expected to be released from the fuel). For other lower volatility radionuclide groups, it is expected that all the fission product inventory is retained within the fuel for events arrested at fuel cladding failure. This is shown in Table 3-4.

**Table 3-4. Fuel Matrix Reduction Factor Complements for LWR Fuel Cladding Failure Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0.95 | 0.95 |
| Halogens (I, Br) | 0.95 | 0.95 |
| Alkali Metals (Cs, Rb) | 0.95 | 0.95 |
| Tellurium Group (Te, Sb, Se) | 1 | 1 |
| Barium, Strontium (Ba, Sr) | 1 | 1 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 1 | 1 |
| Cerium Group (Ce, Pu, Np) | 1 | 1 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 1 | 1 |

By contrast to scenarios arrested at fuel cladding failure, accidents that progress to more extensive in-vessel core damage experience severe fuel temperature excursions. At temperatures realized for these classes of accident, fission product migration out of degraded fuel materials will be larger. The fuel matrix will serve as a less effective barrier limiting fission product release from fuel.

**Table 3-5. Fuel Matrix Reduction Factor Complements for LWR In-Vessel Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0.05 | 0.05 |
| Halogens (I, Br) | 0.75 | 0.65 |
| Alkali Metals (Cs, Rb) | 0.80 | 0.75 |
| Tellurium Group (Te, Sb, Se) | 0.95 | 0.95 |
| Barium, Strontium (Ba, Sr) | 0.98 | 0.98 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0.9975 | 0.9975 |
| Cerium Group (Ce, Pu, Np) | 0.9995 | 0.9995 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0.9998 | 0.9998 |

For events that progress to ex-vessel core damage, the fuel debris leaves the reactor vessel and enters containment. The interaction of core debris with the concrete containment floor promotes additional release of fission products from the fuel matrix into containment. For scenarios progressing to this degree of core damage, the total retention of fission products in the fuel matrix can be characterized by the reduction factors provided in Table 3-6.

**Table 3-6. Fuel Matrix Reduction Factor Complements for LWR Ex-Vessel Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 1 | 1 |
| Halogens (I, Br) | 0.70 | 0.75 |
| Alkali Metals (Cs, Rb) | 0.65 | 0.65 |
| Tellurium Group (Te, Sb, Se) | 0.75 | 0.75 |
| Barium, Strontium (Ba, Sr) | 0.90 | 0.90 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0.9975 | 0.9975 |
| Cerium Group (Ce, Pu, Np) | 0.9995 | 0.9995 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0.9995 | 0.9995 |

### *3.6.3. Reactor Vessel*

As discussed above, the role of the reactor vessel will be somewhat limited at mitigating the release of fission products into the containment. It is assumed that this barrier plays a relatively minor role and thus reduction factors are assumed to be zero for all radionuclide groups.

**Table 3-7. Reactor Vessel Reduction Factor Complements for Spectrum of LWR Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0 |
| Halogens (I, Br) | 0 | 0 |
| Alkali Metals (Cs, Rb) | 0 | 0 |
| Tellurium Group (Te, Sb, Se) | 0 | 0 |
| Barium, Strontium (Ba, Sr) | 0 | 0 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0 |
| Cerium Group (Ce, Pu, Np) | 0 | 0 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0 |

### *3.6.4. Containment*

Fission products released into containment typically will be subject to a number of mechanisms that promote their removal from the containment atmosphere. Such removed fission products are those that settle on to containment surfaces, or deposit in containment water pools. The effectiveness of these fission product removal mechanisms typically depend on the amount of time fission products are "held up" within containment before migrating to a pathway through which they can be released to the environment. In addition, the pathway through which fission products travel before being released to the environment can also play an important role in promoting more removal from the effluent stream. When characterizing fission product reduction factors attributable to the containment barrier, it is important to distinguish

- The time fission products are held up within containment
- The release pathway

Note that reduction factors apply to fission products other than noble gases. It is typically assumed that noble gases will be released from the containment without any possibility of removal. These fission products are in the form of vapors that are not subject to the same removal mechanisms acting on other radionuclides that form aerosols within containment.

For simplicity, Table 3-8 only presents reduction factors for a scenario in which releases occur through the following pathways

- BWR release through the wetwell vent, assuming a saturated suppression pool
- PWR release through a deliberate vent, assuming 20 hours of hold up prior to initiation of venting[3]

For the BWR release, it is assumed that fission product removal is primarily achieved through retention of aerosols in the suppression pool. Minimal removal is assumed due to settling processes within the drywell. From Reference [31], this reduction factor is 0.05.

[3] This assumes that the containment capacity is exhausted at 24 hours and fission product release into containment has largely occurred by 4 hours.

For the PWR release, it is assumed that the only mechanisms available to remove fission product aerosols is natural settling processes during the prolonged hold-up period. From Reference [31], the combined reduction over multiple time periods is about 0.006. Assuming the lowest rate of settling over the entire hold-up period gives a reduction factor of 0.05. This will be assumed for simplicity.

**Table 3-8. Containment Reduction Factor Complements for Spectrum of LWR Accident Scenarios (based on Reference [31])**

| Radionuclide Group | BWR | PWR |
|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0 |
| Halogens (I, Br) | 0.95 | 0.95 |
| Alkali Metals (Cs, Rb) | 0.95 | 0.95 |
| Tellurium Group (Te, Sb, Se) | 0.95 | 0.95 |
| Barium, Strontium (Ba, Sr) | 0.95 | 0.95 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0.95 | 0.95 |
| Cerium Group (Ce, Pu, Np) | 0.95 | 0.95 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0.95 | 0.95 |

### *3.6.5. Sample Consequence Evaluation*

Using the above reduction factors, a sample PWR consequence evaluation was performed with the MELCOR Accident Consequence Code System (MACCS) computer code [34]. MACCS was used in order to evaluate dose at a range of distances from the plant. The fission product inventory was assumed based on the Surry SOARCA study [32].

The dose is normalized to the peak dose determined by this calculation. Subsequent evaluations of non-LWR concepts consider off-site dose with respect to this reference dose. The assumed release has the following characteristics

- A plume of radionuclides is discharged to the environment at a constant mass flow rate representative of a typical plume in the Surry SOARCA study [32]
- The plume is released into the environment from an elevated point consistent with that used for elevated releases in the Surry SOARCA study [32]
- A single release is assumed to occur over a time period of 1-hour

These characteristics are applied to subsequent off-site dose assessments for the non-LWR concepts, which are presented below. Note that the assumed fission product inventories for the non-LWR concepts considered below are reduced relative to those typical for an LWR to reflect the lower thermal power of the concepts considered. This is only an approximation, with more detailed core inventory calculations being performed as part of the U.S. NRC effort to develop modeling and simulation tools/methods for non-LWRs [22].

The off-site dose decays with distance from the site due to fission product deposition and dilution of the radiological material as a result of plume spreading. By about 10 km, the dose has decayed by two orders of magnitude. Beyond about 30 km, the dose has been reduced by three orders of magnitude.

# 4. RELEASE MITIGATION STRATEGY ASSESSMENT OF REACTOR CONCEPTS

## 4.1. HTGR Release Mitigation Strategy Assessment

Experience with HTGRs has been gained through a range of different reactors.

- The Dragon Reactor was a 20 MW(t) reactor constructed in the United Kingdom that first achieved criticality on August 23, 1964. This reactor operated at a system pressure of 2 MPa(a) with core inlet/outlet temperatures of 350°C/750°C.
- The Peach Bottom Unit 1 reactor was 115 MW(t) plant located in Pennsylvania. It operated at a system pressure of 2.4 MPa(a) with core inlet/outlet temperatures of 350°C/750°C.
- The AVR was a 46 MW(t) German reactor that served as a prototype of the pebble bed concept. It operated at a system pressure of 1.1 MPa(a) with core inlet/outlet temperatures of 270°C/950°C. In total, the reactor operated with 100,000 spherical pebbles each having a diameter of 6 cm.
- The Fort St. Vrain HTGR operated at a power level of 842 MW(t) and was the first reactor to use stacked columns of prismatic fuel elements. It operated at a system pressure of 4.8 MPa(a) with core inlet/outlet temperatures of 405°C/775°C.
- The Thorium High Temperature Reactor (THTR-300) was a German reactor that operated at a power level of 750 MW(t). It operated at a system pressure 4 MPa(a) of with core inlet/outlet temperatures of 404°C/777°C.
- The High Temperature Test Reactor (HTTR) was a Japanese reactor that operated at a power level of 300 MW(t). It operated at a system pressure of 4 MPa(a) with core inlet/outlet temperatures of 395°C/850-950°C.
- The High Temperature Test Reactor (HTR-10) was a Chinese reactor that operated at a power level of 10 MW(t). It operated at a system pressure of 3 MPa(a) with core inlet/outlet temperatures of 250°C/700°C.

While this experience base is more extensive than other advanced reactor concepts, this is still relatively limited compared to LWR operational experience.

### *4.1.1. Phenomenological Considerations*

A detailed consideration of HTGR source term modeling has been developed as part of the NRC vision and strategy for non-LWR modeling capabilities in Reference [22]. The relevant physical processes governing fission product release and transport are illustrated in Figure 4-1.

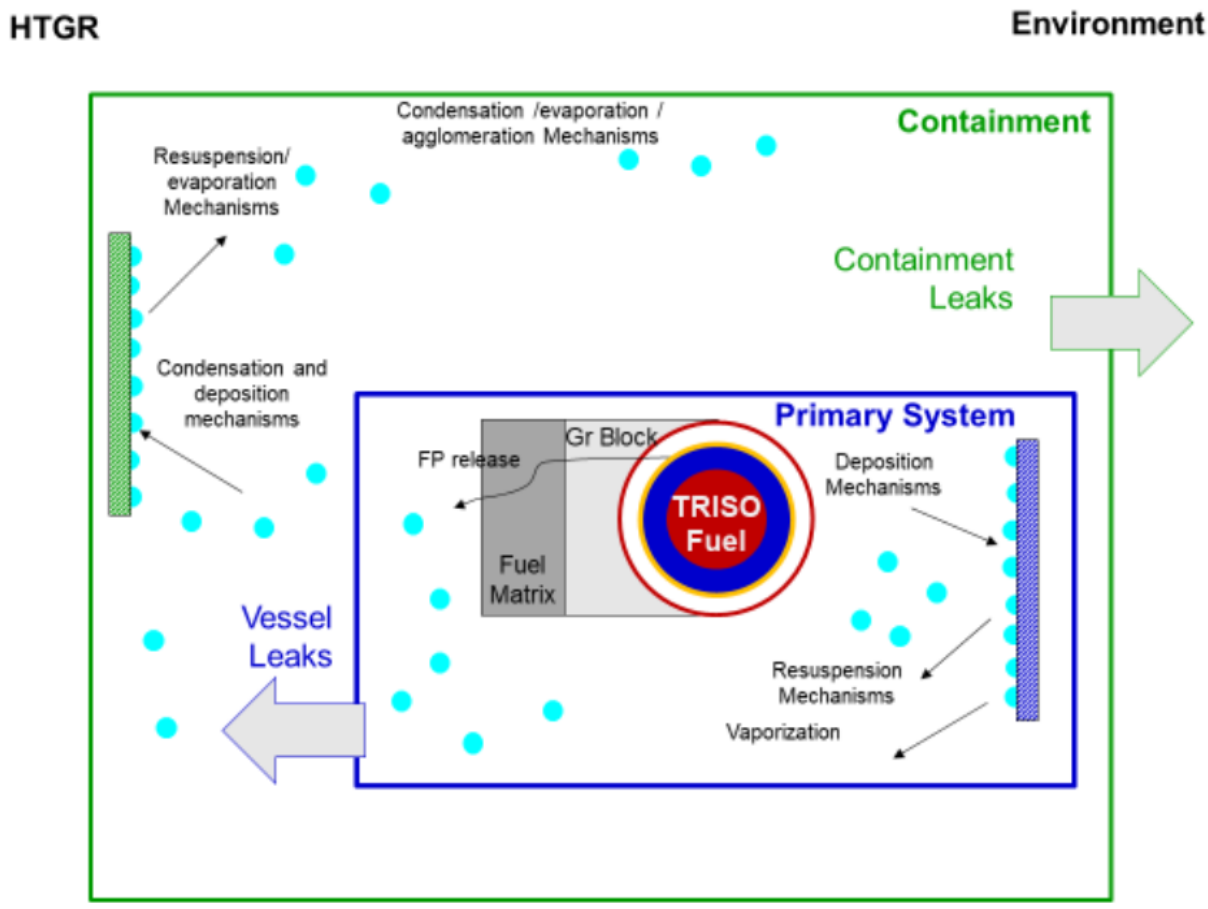


**Figure 4-1. Summary of Radionuclide Release and Transport Processes for High Temperature Gas-Cooled Reactors [22]**

### *4.1.2. Initiating Events*

A number of PRA studies have been conducted to evaluate HTGR concepts. The PRA studies for mHTGR and NGNP are used to establish a set of initiating events to consider for this reactor concept. The reference initiating event chosen for consideration in this report is the break in the Helium Pressure Boundary (HPB). Earlier fission product release could occur in this scenario due to depressurization of the reactor. There is the potential for long-term release as a result of core heatup leading to thermally-driven fission product release from the fuel.

In the evaluations considered here, focus is placed on medium and large breaks. These break sizes give rise to a more rapid depressurization of the primary system. They also tend to give rise to more severe core temperature excursions. Such breaks are examples of a depressurized loss of forced circulation (DLOFC). A pressurized loss of forced circulation (PLOFC) is an event that would correspond to a much smaller HPB break without a rapid depressurization of the heat removal system.

This event is characterized as follows:

- An initiating leak or break in the HPB piping
- Reactor trip

- Loss of heat transport to the steam generator as a result of the loss of forced circulation in the primary system when heat transport system pumps trip
- Shutdown cooling is assumed to be lost
- The reactor building vent is assumed to open successfully to vent helium discharged from the primary system, arresting the pressure excursion in the building

A range of break sizes have been considered in previous studies.

This event does not fall within the more severe range of events that could be considered for this reactor concept. Air ingress into the primary system, as an example, is not considered. Such events would lead to potentially more severe core heatup transients.

### *4.1.3. Reference Initial Fission Product Inventory*

A reference inventory for the HTGR for illustrative purposes is extracted from work to develop a source term for the NGNP concept. This reactor concept was a 250 MW(t) pebble bed reactor which is consistent with some of the more prominent proposals for HTGR advanced reactors being developed by industry. The fission product inventory for NGNP is presented in Table 4-1, and compared to a typical inventory for the Peach Bottom BWR from the SOARCA study. As is typical with these lower power reactors, the initial fission product inventory is lower than the high power LWRs in the commercial nuclear fleet in the United States.

The available inventory for release is lower when dealing with lower power reactors than many of the operating LWRs. For sufficiently small initial inventories, the dispersion of radiological release in the environment could be sufficient to reduce the off-site dose sufficiently to allow a reduction in the exclusion area boundary.

Table 4-1 provides a summary of a reference HTGR fission product inventory in comparison to a typical LWR inventory extracted from the Peach Bottom SOARCA study. While there is a reduced inventory available for release, this reduction in inventory combined with dispersion in the environment would not be sufficient to eliminate the potential for offsite public health consequences.

**Table 4-1 – Representative HTGR Fission Product Inventory**

| Radionuclide Class | Representative Element | Member Elements | Peach Bottom SOARCA Radioactive Mass [kg] | Reference HTGR Radioactive Mass [kg] |
|---|---|---|---|---|
| Noble Gas | Xe | He, Ne, Ar, Kr, Xe, Rn, H, N | 531.7 | 53 |
| Alkali Metals | Cs | Li, Na, K, Rb, Cs, Fr, Cu | 323.0 | 32 |
| Alkaline Earths | Ba | Be, Mg, Ca, Sr, Ba, Ra, Es, Fm | 235.6 | 24 |
| Halogens | I | F, Cl, Br, I, At | 19.9 | 2 |
| Chalcogens | Te | O, S, Se, Te, Po | 49.1 | 5 |
| Platinoids | Ru | Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni | 342.8 | 34 |

| Radionuclide Class | Representative Element | Member Elements | Peach Bottom SOARCA Radioactive Mass [kg] | Reference HTGR Radioactive Mass [kg] |
|---|---|---|---|---|
| Early Transition Elements | Mo | V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W | 400.2 | 40 |
| Tetravalent | Ce | Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C | 1555.5 | 156 |
| Trivalents | La | Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf | 1793.7 | 179 |
| Uranium | U | U | 132794.0 | 13279 |
| More Volatile Main Group | Cd | Cd, Hg, Zn, As, Sb, Pb, Tl, Bi | 6.6 | 1 |
| Less Volatile Main Group | Sn | Ga, Ge, In, Sn, Ag | 9.6 | 1 |

### *4.1.4. Source Term Behavior in Given Accident Scenarios*

The evaluation of the contributions to fission product release mitigation in the HTGR concept are itemized as follows:

6. Retention of fission products dissolved in the fuel matrix of TRISO particles

   This mode of release mitigation is strongly influenced by the temperature excursion in the HTGR core. The loss of coolant scenario has been found to give rise to a more severe core temperature transient than other considered scenarios in past analyses. For example, the peak core temperature was found in one analysis to reach nearly 1600°C for a loss of coolant accident, somewhat less than 400°C higher than the peak core temperature found for a loss of forced circulation event without loss of coolant inventory.

   Core temperature excursions in an HTGR, once heat removal function has been lost, are strongly influenced by radial conduction within the packed bed or prismatic block fuel matrix as well as radiative heat loss from fuel to the reactor vessel wall. The heat loss to the reactor vessel wall is dissipated in the reactor building. Since the magnitude of radiative heat loss from fuel is very dependent on the temperature gradient between the core and the vessel wall, the core must reach relatively high temperatures for radiation heat loss to reach levels comparable with decay heat. At the same time, the decay heat in the reactor core is decreasing following reactor shutdown. Should radiative heat losses exceed the decay heat level, the core temperature excursion will be terminated and it is possible for the core to gradually cool off. Radiation only becomes an appreciable heat transfer pathway when the core reaches very high temperatures. At a certain point, the temperature will reach a point where radiative heat losses are of the same magnitude as decay heat. The HTGR is distinguished from LWRs by the fact that this occurs at a temperature lower than that required to initiate core damage.

   In an HTGR, there are three distinct phases of the core thermal excursion. An example MELCOR calculation of the core temperature excursion is provided in Figure 4-2. This example shows the distribution of temperatures across the radial extent of the reactor core. As expected,

the temperature excursion is most severe in the center of the core given the thermal resistance to radiation provided by peripheral regions of the core surround the core center. The peripheral region is in direct thermal contact with the vessel wall and thus the thermal resistance to radiative heat loss is lower. As a result, this portion of the core exhibits the least severe thermal excursion. This accident can be categorized into three phases based on the thermal response of the core.

- Prolonged core heatup
- Brief stabilization of the peak core temperature at a quasi-steady level
- Long-term temperature decrease in the reactor core

In the first phase, the core heats up from the initial temperature at the time of reactor trip. This heatup occurs because the radiative heat loss remains lower than the level of decay heat in the core. In the very initial phase of core heatup, there is a more rapid increase in core temperature as radiative heat losses are quite small compared to decay heat. The rate of heatup during this initial portion of the core heatup transient is governed by the thermal inertia in the system, which for an HTGR is relatively high. Thus, core heatup in an HTGR is a relatively slow process. For loss of coolant accidents, the core heats up by only 400K over a period of a few hours. After this initial temperature increase, the magnitude of radiative heat loss from the core increases substantially relative to the decay heat level. As a result, the remaining heat up of the core occurs over a period of greater than 1 day before reaching a state where the radiative heat losses balance the decay heat level. For a loss of coolant accident, the peak temperature reached at the end of this phase is nearly 1800K.

The second phase of the core temperature excursion is characterized by a period somewhat less than a day where the radiative heat losses from the core are approximately equal to the level of decay heat. This period lasts for a limited time as a result of the gradual decrease in decay heat in the reactor core.

The final, long-term phase of the core temperature excursion is characterized by continuous decrease in the core temperature. This occurs due to the gradual decrease of decay heat such that radiative heat losses exceed heat generation within the fuel.

The determination of the core temperature excursion is critical to evaluating the potential for fission products dissolved in TRISO particles to be released due to thermally-driven diffusion. From evaluated temperature excursions for loss of coolant or loss of forced circulation accidents, there is limited thermally-driven diffusion of the dissolved fission products. This is a critical feature of the HTGR; the temperature excursions for the range of events catered for in the design concept are not severe enough to promote thermally-driven fission product release.

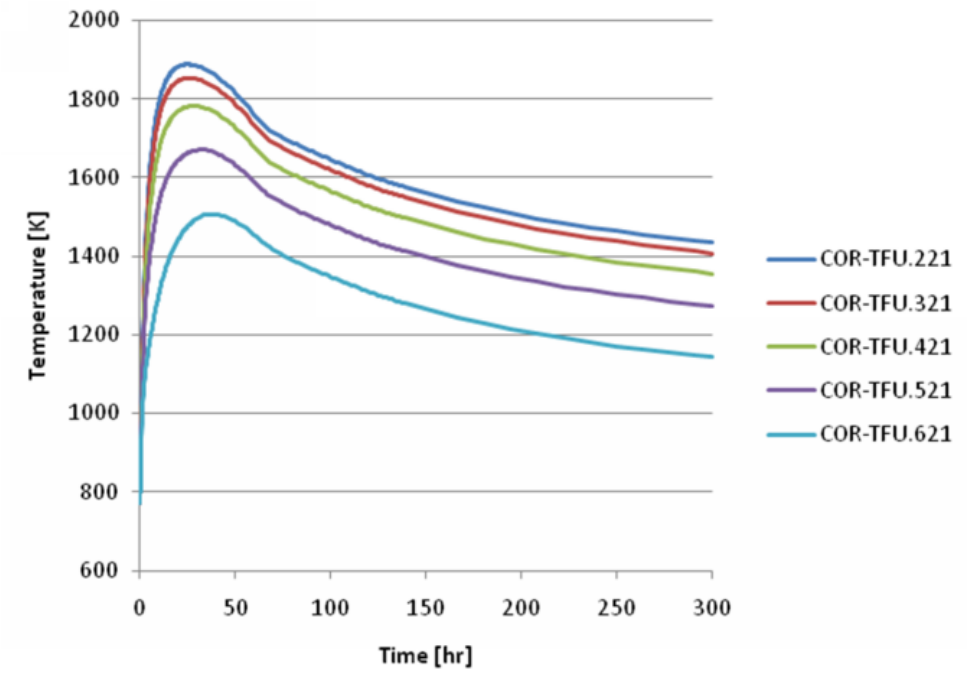


**Figure 4-2 – Variation of Fuel Temperature across Core Radius for Depressurized Loss of Forced Circulation Event (DLOFC)**

7. Retention of fission products, including noble gases, within the TRISO fuel pebble
   A TRISO particle is assumed to have the following structure (illustrated in Figure 4-3).

- A fuel matrix made from either UCO or $UO_2$
- An inner porous carbon buffer
- A middle SiC layer
- An outer pyrolytic carbon layer

  A large number of TRISO particles are embedded in a graphite sphere that forms the TRISO pebble.

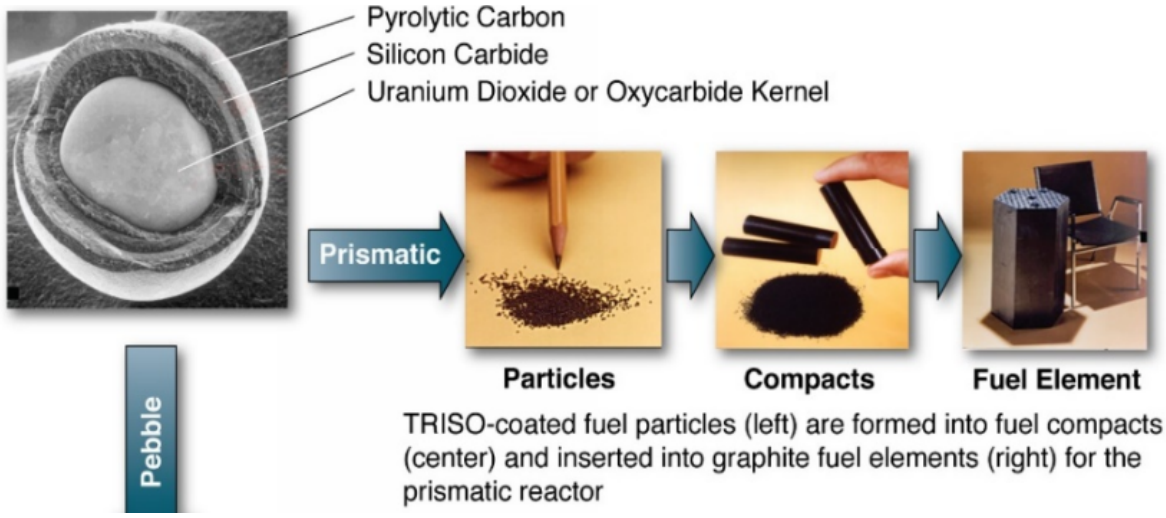


**Figure 4-3 – Illustration of TRISO Fuel Particle and Fuel Pebble Design [36]**

Fission products will migrate during operation throughout the carbonaceous matrix of the TRISO particle. An illustrative fission product distribution is provided in Figure 4-4. Fission products are generally effectively retained beneath the SiC middle layer in the TRISO particle during operation. In intact TRISO particles, the SiC is designed to function as a pressure vessel that retains fission product gases generated within the fuel kernel. However, operational experience and TRISO fuel testing have found that some fraction of these particles can be initially defective or already-failed at the time of the event. Since there are billons of TRISO particles in an HTGR core, even a very small number of TRISO particle failures could result in a moderate release of fission products either during operation or following the onset of the accident.

Thus, TRISO particle integrity is a particularly important consideration for the HTGR design. With highly reliable TRISO particles, very limited release into the coolant could be expected. noble gases generated in the fuel kernel, with highly reliable TRISO particles, would be expected to remain in the TRISO particle under accident conditions that do not thermally challenge the integrity of the three layers surrounding the fuel kernel. Furthermore, the diffusion of fission product gases dissolved in the fuel kernel will be relatively minor if core temperature excursions are arrested with peak fuel temperatures below about 1700°C. For the range of design basis events, fuel temperature excursions are not sufficient to cause thermally-induced TRISO particle failures or lead to enhanced thermal diffusion of fission products out of the fuel.

TRISO particle reliability and the fuel thermal transient are critical factors influencing fission product release. TRISO particle reliability is affected by fuel fabrication processes and quality control. The thermal transient is affected by the accident scenario. For example, air ingress scenarios (e.g., due to cross-duct breaks) give rise to a more severe fuel thermal transient because of graphite oxidation [35].

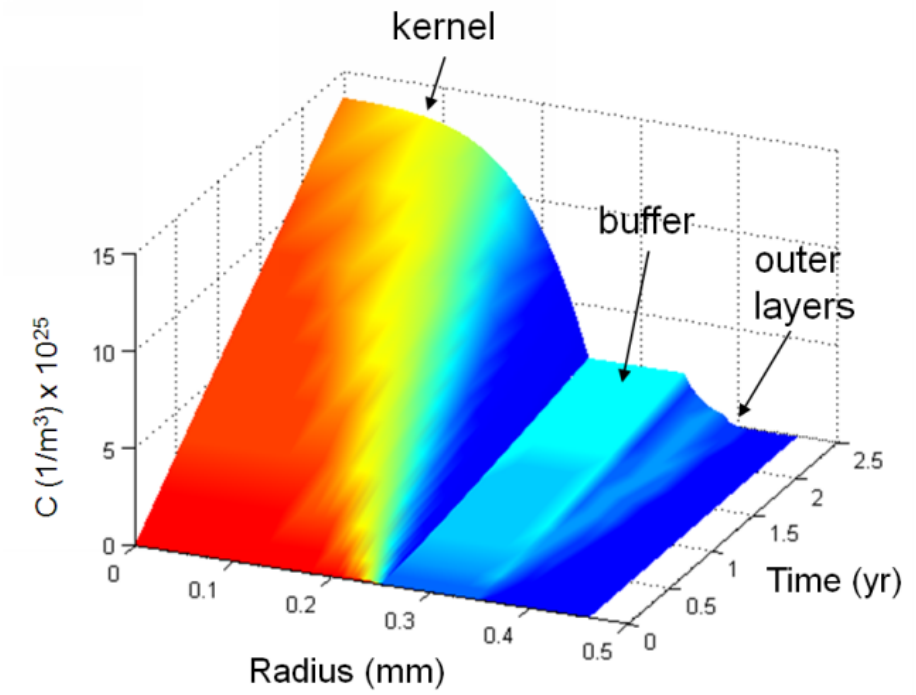


**Figure 4-4 – Illustrative Fission Product Species Distribution in a TRISO Particle [22]**

8. Retention of fission products within the heat removal

   The HPB break scenario is among the most limiting from the perspective of the release mitigation effectiveness of the heat removal system barrier. The opening in the HPB provides a direct path for radiological release out of the heat removal system, which will tend to limit the fission product residence time. This reduces the extent to which natural deposition mechanisms will facilitate retention of fission products on surfaces within the reactor and heat removal system. The role of the heat removal system is relatively small for this type of scenario.

   Furthermore, the break in the heat removal system leads to an immediate release of activity. This includes activated graphite dust formed during reactor operation. This activity is typically quite low relative to the activity in the fuel. For example, for the mHTGR the coolant activity is about $3\times10^{3}$ Ci, compared to a core inventory of $2\times10^{9}$ Ci.

9. Retention of fission products within the reactor enclosure

   Similar to the consideration of retention in the heat removal system, the reactor enclosure will play a more minimal role for this type of event. The opening of the building vent to control the pressure excursion establishes a temporary leak path to the environment. This allows fission products released from the heat removal system during HPB depressurization to migrate into the environment with relatively limited residence time in the reactor enclosure. The role of the reactor enclosure in release mitigation for this type of scenario is thus initially more limited. Subsequently, it can be expected to provide hold-up of fission products that have migrated out of the heat removal system. Since fission product release will occur in the longer-term as a result of heatup of TRISO fuel pebbles, the reactor enclosure will serve as a barrier to mitigate longer-

term releases to the environment. Its effectiveness can thus be approximated in a manner similar to LWR containment volumes due to fission product hold-up.

10. Retention of fission products along the release pathway to the environment
    The HTGR concept considered is not assumed to incorporate a filter to remove contamination from the discharges through the reactor building vent. There is thus negligible release mitigation along the discharge pathway to the environment. With enclosure vents assumed to be relatively large, there will be limited retention along this path due to turbulent deposition processes (see, for example, Reference [29] and Reference [30]). Leakage of fission products from the reactor enclosure is assumed to have an attenuation based on RASCAL guidance [31].

### 4.1.5. Summary of Dominant Release Mitigation Factors

The dominant factors achieving release mitigation for the HTGR concept are:

- The core temperature excursion is mitigated due to conductive and radiative heat losses accommodating all sources of energy generation in the core (e.g., due to decay heat generation and potentially any graphite oxidation). Should energy generation sources exceed or heat removal mechanisms be less effective than that considered in previous studies informing the reference assessments, it is likely that more severe core temperature excursions would occur. Such conditions would result in the potential for more substantial fission product release than considered possible based on current studies that have primarily evaluated the robustness of the design to design basis transients.
- The integrity of the TRISO particle is an additional factor affecting fission product release to the environment. The layers encapsulating the fuel kernel, including a SiC layer, are designed to be structurally robust. This encapsulation of the fuel kernel provides a robust pressure barrier significantly mitigating release of fission products into the reactor coolant. The reliability of this encapsulating layer is targeted to be relatively high such that TRISO particle failures under design basis transients could be limited to fraction of the population on the order of $10^{-5}$. The failure fraction of TRISO particles is a performance metric for HTGR design concepts relevant to substantially mitigating radiological release to the environment. However, there have been no studies to evaluate the level of TRISO particle failures at which site-boundary dose would exceed regulatory limits for design basis events.

### 4.1.6. Summary of HTGR Source Term Reduction Factor Estimates

As shown in Figure 4-5, fission product release from TRISO fuel is very low with the exception of Ag for the temperatures tested in the INL AGR-1 safety tests. These results apply to situations when the SiC layer remains intact. The release of Cs has been found to increase when SiC failures have been identified. Transport of fission products out of TRISO fuel is thus significantly influenced by the fuel structures (principally SiC) that enclose the fuel particle.

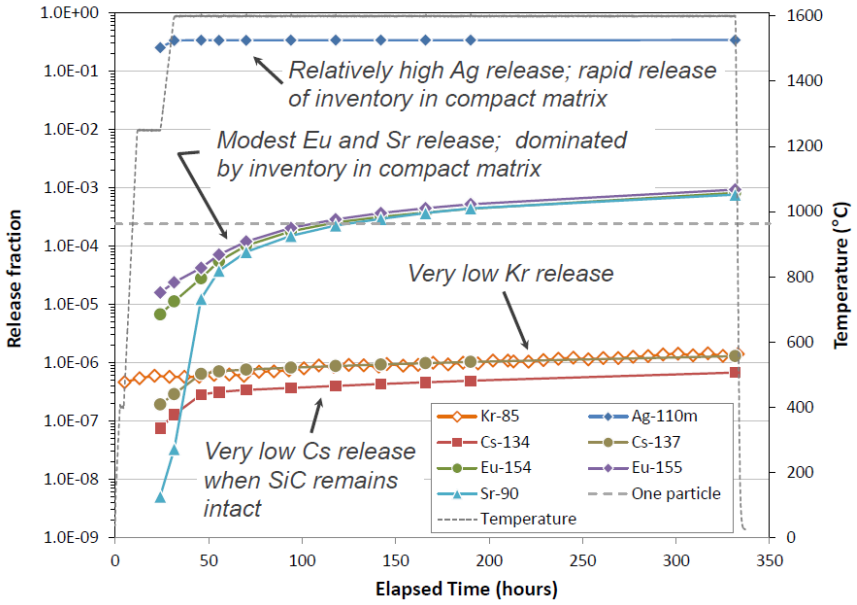


**Figure 4-5. Evaluation of Radionuclide Release Fractions from INL AGR-1 Safety Test Results for TRISO Fuel [36]**

Furthermore, in the range of temperatures tested with inert atmospheres, these fuel structures have been observed to maintain integrity sufficiently to prevent fission product release from fuel. Figure 4-6 illustrates the expected fraction of inherent and accident-induced TRISO fuel failures established as part of the NGNP fuel qualification [36]. The assumed fraction of TRISO particle failures is taken to be about $1\times10^{-4}$. This is used to identify the fraction of fuel that can release for when assuming an event scenario with largely intact TRISO fuel. A sensitivity study is also considered in which the failure of the SiC occurs across all the fuel pebbles due to high temperature accident conditions being realized (for example, due to an air ingress event).

When TRISO particles have not failed, these results indicate that it is reasonable to assume that there is negligible release from the TRISO pebble into the reactor vessel. When particles have failed, the fission product release is governed primarily by the release of fission products from the fuel matrix through the graphite fuel pebble. For simplicity, fuel matrix release of fission products similar to that of LWRs for in-vessel damage is assumed for this illustrative evaluation. There are current efforts underway to enhance the modeling of fission product release out of failed fuel particles and into the reactor vessel [22]. However, assuming LWR in-vessel damage release fractions is likely bounding for this scoping assessment.

| Condition | Design specifications for TRISO failure fraction | | AGR-1 95% confidence |
|---|---|---|---|
| | NGNP[a] | HTR-Modul[a] | |
| In-service | $\leq 2.0\times10^{-4}$ | $\leq 1.6\times10^{-4}$ | $\leq 1.1\times10^{-5}$ |
| 1600°C accident | $\leq 6.0\times10^{-4}$ | $\leq 6.6\times10^{-4}$ | $\leq 6.6\times10^{-5}$ [b] |

[a] Values for NGNP and HTR-Modul taken from NGNP Fuel Qualification White Paper, INL/EXT-10-18610, Idaho National Laboratory, 2010.
[b] Value obtained by combining statistics from 1600 and 1700°C AGR-1 safety tests (0 failures out of 45,550 particles)

**Figure 4-6. Results of TRISO Fuel Qualification as part of NGNP [36]**

As in the case of LWRs, it is assumed that there is relatively negligible hold-up of fission products within the reactor vessel. There is substantial deposition of activated carbon (in the form of graphite dust) in the reactor vessel. However, activated carbon is not considered for the purposes of performing an initial assessment of off-site consequences. As noted above this represents a relatively small fraction relative to the total core activity.

The retention of fission products in the reactor enclosure is assumed to be similar to that for LWRs. A total hold-up time of 20 hours of hold-up is assumed for fission products released into the reactor enclosure. After 20 hours, an unmitigated release is assumed. Since the reactor enclosure design is likely to evolve through further evaluation of accident scenarios, an additional sensitivity evaluation is considered in which leakage from an intact reactor enclosure occurs. To represent this, a reduction factor of 0.001 is assumed. This is the lower limit reduction factor applied in RASCAL [31].

Table 4-2 presents the HTGR source term reduction factor estimates for scenarios with significant retention of fission products within the TRISO particles.

**Table 4-2. Summary of HTGR Source Term Reduction Factor Complement Estimates – Significant Retention within TRISO Pebble (based on Reference [31] with adaptation to represent significant retention of fission products in TRISO pebble)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0.9999 | 0.05 | 0 | 0.999 | 0 |
| Halogens (I, Br) | 0.9999 | 0.65 | 0 | 0.999 | 0.6 |
| Alkali Metals (Cs, Rb) | 0.9999 | 0.75 | 0 | 0.999 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0.9999 | 0.95 | 0 | 0.999 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0.9999 | 0.98 | 0 | 0.999 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0.9999 | 0.9975 | 0 | 0.999 | 0.6 |
| Cerium Group (Ce, | 0.9999 | 0.9995 | 0 | 0.999 | 0.6 |

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Pu, Np) | | | | | |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0.9999 | 0.9998 | 0 | 0.999 | 0.6 |

Table 4-3 presents HTGR source term reduction factor estimates for a situation in which TRISO particles have largely failed. In this case, it is assumed that 50% of the TRISO particles in the core have experienced a failure. In other respects, the reduction factors for this sensitivity are the same as presented in Table 4-2, which provides an estimate for a situation with the majority of TRISO particles remaining intact.

**Table 4-3. Summary of HTGR Source Term Reduction Factor Complement Estimates – Insignificant Retention within TRISO Pebble (based on Reference [31] with adaptation to represent insignificant retention of fission products in TRISO pebble)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0.5 | 0.05 | 0 | 0.999 | 0 |
| Halogens (I, Br) | 0.5 | 0.65 | 0 | 0.999 | 0.6 |
| Alkali Metals (Cs, Rb) | 0.5 | 0.75 | 0 | 0.999 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0.5 | 0.95 | 0 | 0.999 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0.5 | 0.98 | 0 | 0.999 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0.5 | 0.9975 | 0 | 0.999 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0.5 | 0.9995 | 0 | 0.999 | 0.6 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0.5 | 0.9998 | 0 | 0.999 | 0.6 |

### *4.1.7. Sample HTGR Consequence Assessment*

Figure 4-7 presents the sample, illustrative evaluation of HTGR off-site consequences across a number of sensitivity cases, comparing off-site dose with a reference LWR dose. The dose results are presented normalized to the peak LWR dose.

The overall reduction of dose for an HTGR is attributable to a combination of lower power levels, leading to a reduced inventory of fission products that can be released from the fuel. In addition to this effect, retention of radionuclides in TRISO particles has a substantial effect. Overall, retention of fission products in TRISO particles could potentially reduce the off-site dose by nearly six orders of magnitude, including the effect of reduced operational power level.

To assess the role of TRISO particle retention of fission products, a sensitivity case assuming failure of 50% of the TRISO particles was performed. This insignificant retention sensitivity case exhibits enhanced off-site consequences. Doses are approximately three orders of magnitude higher relative to the sensitivity case assuming significant fission product retention in TRISO particles.

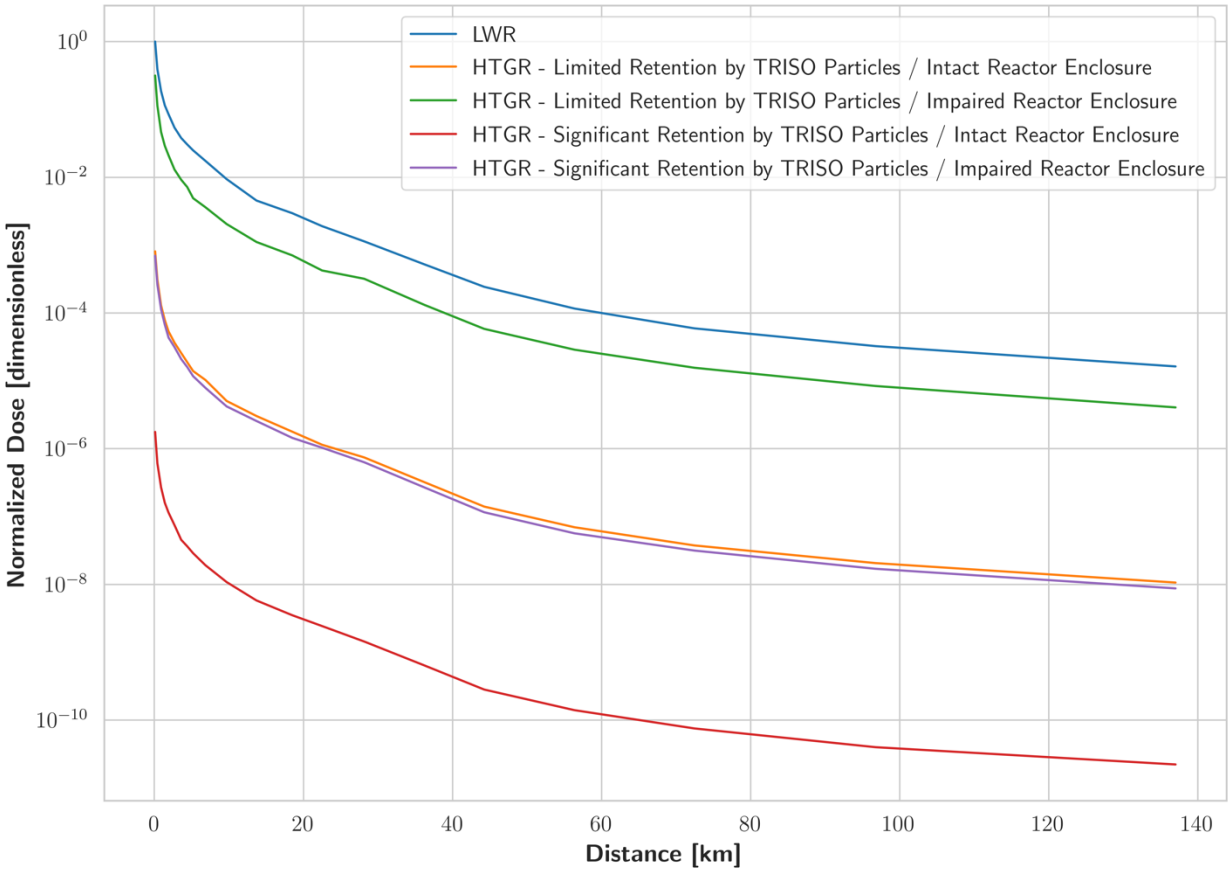


**Figure 4-7. Comparison of HTGR Off-Site Dose Evaluations with Reference LWR Evaluation**

The evaluation shown in Figure 4-7 assumes that the fission product inventory is rescaled relative to a typical LWR fission product inventory. To assess the implications of fission product retention measures independent of fission product inventory, an additional, illustrative sensitivity is performed assuming an LWR initial fission product inventory. The off-site dose is shown as a function of distance from the release point in Figure 4-8. In a situation with limited retention by TRISO particles, and an impaired reactor enclosure, there is a somewhat higher off-site consequence relative to the LWR. This is due to the limited additional fission product retention applied for the HTGR concept in the event that both TRISO fuel particles and the reactor enclosure have become impaired. This further emphasizes the important role that TRISO particles play in the overall strategy for mitigating releases to the environment.

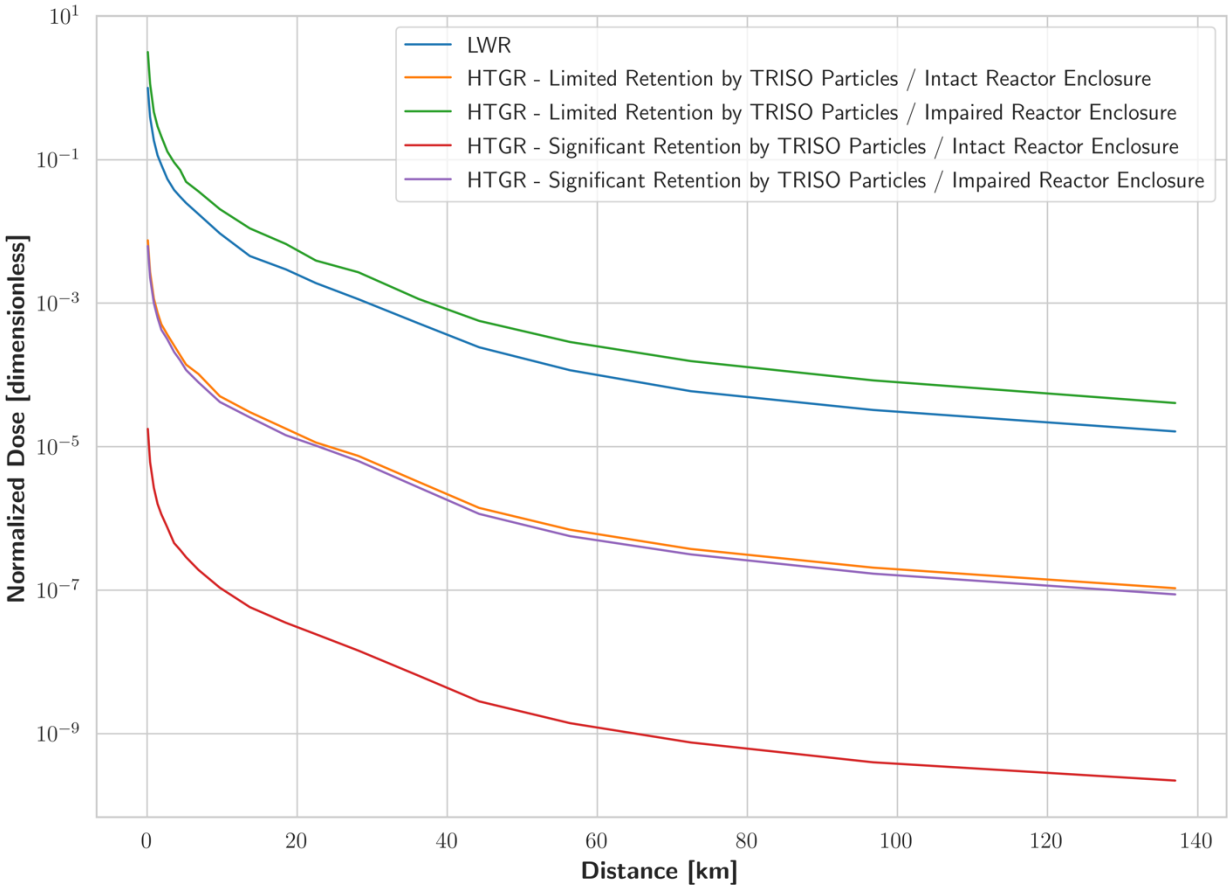


**Figure 4-8. Comparison of HTGR Off-Site Dose Evaluations with Reference LWR Evaluation assuming Initial HTGR Fission Product Inventory Equivalent to the LWR Fission Product Inventory**

## 4.2. SFR Release Mitigation Strategy Assessment

Experience in the U.S. with SFRs has been gained through a range of operating reactors.

- Experimental Breeder Reactor II (EBR-II) was operated from 1963 to 1994. It operated at a power level of 62.5 MW(t) using metal fuel. It was a pool-type reactor.
- The Fermi I reactor operated from 1963 to 1972. It operated at a power level of 200 MW(t). Metal fuel was used in the reactor from 1963 to 1966, with oxide fuel used from 1970 to 1972. The reactor heat removal system was configured in the form of a loop.
- The Fast Flux Test Facility (FFTF) was operated from 1980 to 1993. It operated at a power level of 400 MW(t) using metal oxide fuel. The reactor heat removal system was configured in the form of a loop.

Due to the operation of these reactors, some experience with actual fuel damage events was developed.

- The Fermi I incident was a flow blockage event that led to melting of two fuel assemblies and approximately 150 fuel pins. Fuel melting and relocation occurred. Only noble gases were found in the cover gas above the core. The other fission product releases were found to have been retained in the sodium.

- EBR-II experienced melting of a metal fuel element within an experimental capsule. A defect in the fuel led to insufficient cooling. This fuel was present in the reactor for 5 months, with the reactor going through a number of power cycles in this period. As a result, repeated periods of melting occurred. Releases that occurred from the melted fuel, with the exception of noble gases, were found within the sodium coolant. Noble gas releases were found in the cover gas.

It is important to note the role that fission product retention in the sodium could play. However, the solubility of radionuclides in the sodium across a range of accident conditions is not well characterized at present. This impact of fission product retention in sodium is a critical sensitivity to consider when evaluating the range of off-site consequences that could be realized.

### 4.2.1. Phenomenological Considerations

The relevant physical processes governing fission product release and transport are illustrated in Figure 4-9. These are largely similar to those introduced above for HTGRs. As in the case with HTGRs, the range of phenomena relevant to accident progression and source term evaluations share significant similarities with LWRs.

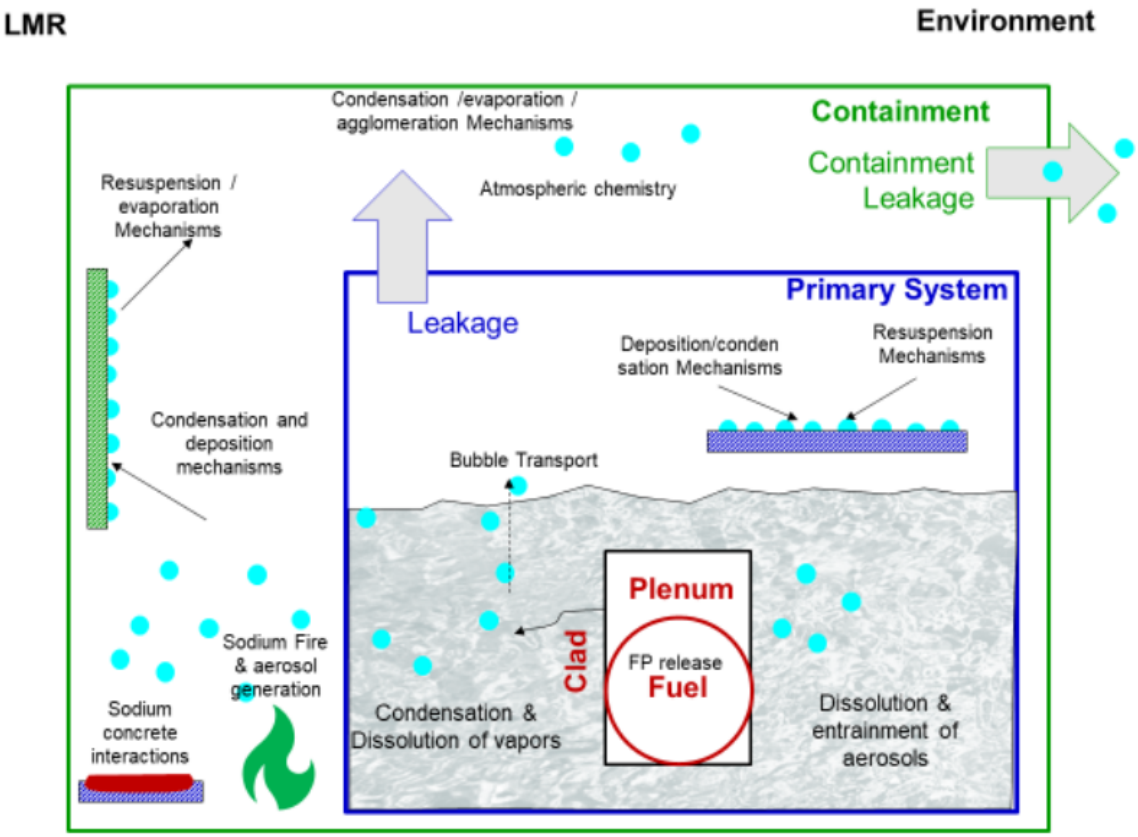


**Figure 4-9. Summary of Radionuclide Release and Transport Processes for Advanced Sodium Fast Reactors [22]**

### 4.2.2. Event Scenarios

The events considered for the SFR evaluate the impact of failures in the following safety functions

- Reactivity control
- Fission and decay heat removal

Under some conditions where reactor shutdown does not initiate, loss of core integrity in SFRs can occur prior to the core reaching a subcritical state. Thus, any impairment of core power removal must consider not only the progression of an accident at decay heat levels, but also accidents initiated prior to reactor shutdown.

The accident scenarios chosen for consideration follow those presented in past studies, such as Reference [17]. It is important to note that a current comprehensive study of accident progression in an SFR has not yet been performed. The most current studies, such as in Reference [17], have been focused on assessing the modeling and phenomenological knowledge gaps that exist and challenge the current capabilities to evaluate SFR accident source terms. The sample scenarios considered in this report are intended primarily to illustrate the framework for evaluating advanced reactor design release mitigation strategies using a scoping source term evaluation methodology.

Two scenarios are selected for consideration in this document. Each scenario involves a loss of flow but differ in terms of the availability of the reactivity control safety function.

1. Protected Loss of Flow (PLOF) – reactivity control safety function available
2. Unprotected Transient Overpower (UTOP) – reactivity control safety function not available

#### 4.2.2.1. Protected Loss of Flow Event Scenario

As discussed in Reference [17], the PLOF event scenario is characterized in terms of the following timeline.

- Loss of primary flow due to cessation of primary pumps with successful coast-down
- Loss of balance of plant heat removal due to secondary side trip
- Degraded function of the Direct Reactor Auxiliary Cooling System (DRACS) such that it operates at 1% of its design capacity
- Reactor trip occurs soon after the initiating loss of flow (about 2 s into the transient)

Note that in the PLOF scenarios considered in Reference [17], the operation of DRACS is assumed to be recovered at 72 hours into the event. This arrests the fuel temperature increase in the simulations reported in Reference [17] and ensures progression to severe core damage does not occur. The prolonged period of core heatup observed in this analysis is due to the relatively high thermal capacity of the working fluid (sodium) in an SFR.

#### 4.2.2.2. Unprotected Transient Overpower Event Scenario

As discussed in Reference [17], the UTOP event scenario is characterized in terms of the following timeline.

- Reactivity insertion of $0.075
- Reactor trip on high neutron flux fails (initiation signal around 2 s into the transient)

The UTOP event scenario bounds other types of unprotected event scenarios, such as the Unprotected Loss of Flow (ULOF) scenario. The ULOF scenario is similar to the PLOF scenario discussed above with the exception that reactor trip fails.

### 4.2.3. Initial Fission Product Inventory

A reference inventory for the SFR for illustrative purposes is extracted from existing work to assess potential source terms for an advanced SFR (see Reference [17]). The fission product inventory for the reference SFR is presented in Table 4-4, and compared to a typical inventory for the Peach Bottom BWR from the SOARCA study. The differences are influenced by the reactor power level as well as the neutron spectrum.

**Table 4-4. Representative SFR Fission Product Inventory**

| Radionuclide Class | Representative Element | Member Elements | Peach Bottom SOARCA Radioactive Mass [kg] | Reference SFR Radioactive Mass [kg] |
|---|---|---|---|---|
| Noble Gas | Xe | He, Ne, Ar, Kr, Xe, Rn, H, N | 531.7 | 213 |
| Alkali Metals | Cs | Li, Na, K, Rb, Cs, Fr, Cu | 323.0 | 129 |
| Alkaline Earths | Ba | Be, Mg, Ca, Sr, Ba, Ra, Es, Fm | 235.6 | 94 |
| Halogens | I | F, Cl, Br, I, At | 19.9 | 8 |
| Chalcogens | Te | O, S, Se, Te, Po | 49.1 | 20 |
| Platinoids | Ru | Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni | 342.8 | 137 |
| Early Transition Elements | Mo | V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W | 400.2 | 160 |
| Tetravalent | Ce | Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C | 1555.5 | 622 |
| Trivalents | La | Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf | 1793.7 | 717 |
| Uranium | U | U | 132794.0 | 53118 |
| More Volatile | Cd | Cd, Hg, | 6.6 | 3 |

| Radionuclide Class | Representative Element | Member Elements | Peach Bottom SOARCA Radioactive Mass [kg] | Reference SFR Radioactive Mass [kg] |
|---|---|---|---|---|
| Main Group | | Zn, As, Sb, Pb, Tl, Bi | | |
| Less Volatile Main Group | Sn | Ga, Ge, In, Sn, Ag | 9.6 | 4 |

### *4.2.4. Source Term Behavior in Given Accident Scenarios*

Advanced SFRs incorporate the following barriers that serve to mitigate the release of fission products to the environment across a range of event scenarios.

1. Retention of fission products dissolved in the fuel pin
2. Retention of fission products within the fuel cladding
3. Retention of fission products within quiescent sodium pools in the reactor system
4. Retention of fission products within the reactor enclosure
5. Deposition of fission products along leakage pathways to the environment

#### 4.2.4.1. Protected Loss of Flow Event Scenario

For a PLOF event scenario, the behavior of the above release mitigation measures can be characterized following the work of Powers et. al. [37]. Some additional discussion has been inserted based on the modeling capability gap analysis presented in Reference [17] for primarily illustrative purposes.

1. Retention of fission products within the fuel cladding
   Example results from SAS4A/SASSYS-1 simulations are presented in this section to better assess the role of accident progression in failing various fission product release mitigation strategies. These results were prepared for the mechanistic source term gap analysis study of Reference [17]. The work reported in Reference [17] has limited applicability to assessing consequences as a result of more severe core degradation because of inherent limitations of the legacy SAS4A/SASSYS-1 code system.

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**NOTE:** The SAS4A/SASSYS-1 code results cannot represent progression to core degradation because such models are not implemented in this code system. As discussed in Reference [22], which provides the NRC vision and strategy for non-LWR source term modeling, modifications have been identified that will close the limited number of SFR modeling gaps in the MELCOR code system. This will enable the MELCOR code to be used in the simulation of an entire accident scenario in an SFR, from initial core heatup through to core damage and progression to ex-vessel damage.

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The fission product release insights that can be gained from Reference [17] are primarily restricted to the performance of fuel pins. An illustration of the SFR fuel pin modeled in Reference [17] is shown in Figure 4-10.

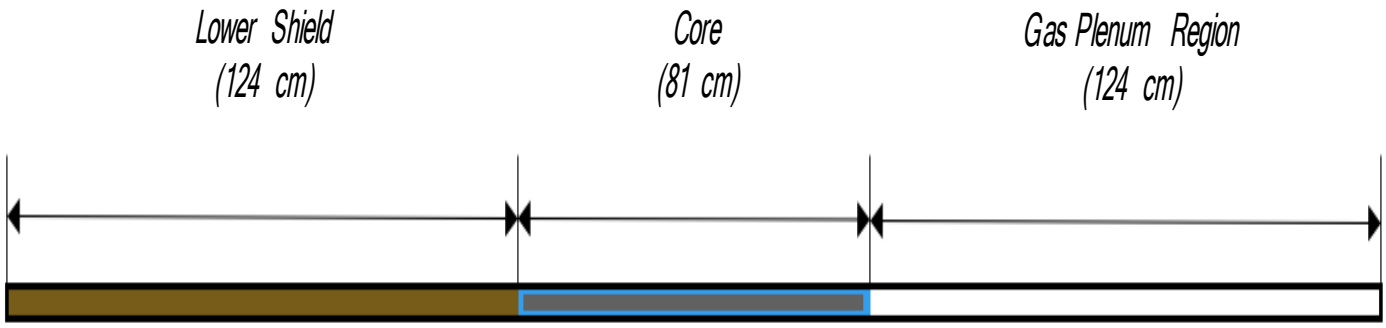


**Figure 4-10 – Illustration of SFR Reactor Fuel Pin [17]**

Figure 4-11 shows the primary system and peak cladding temperature excursions obtained from SAS4A/SASSYS-1 simulations performed for the gap assessment study in Reference [17]. The restoration of DRACS function at 72 hours in the simulation arrests the temperature excursion in the primary system and fuel cladding. As a result, source terms for this scenario are limited to releases as a result of fuel cladding damage.

The nature of fission product release for a clad failure event has been carefully considered by Powers et. al. in formulating research needs for the development of SFR accident source terms [37]. This work indicates the following:

- Metal fuel cladding failure leads to venting of primarily noble gases from the gap and plena in the fuel rods
- Upon failure of the fuel cladding, the releases will also include vapors of more volatile fission products such as cesium, iodine and tellurium
- As the fuel rod depressurizes, gas flows would likely entrain condensed particles or droplets
    - For oxide fuel, it is possible that small, solid fuel particles could be entrained in the gas flow into the primary system
    - For metal alloy fuels, droplets of bond sodium used to thermally bond the fuel and clad would likely be entrained in the gas flow; this bond sodium would be contaminated with dissolved radionuclides from operation

- The vented gases, entrained particles and droplets and vaporization of sodium coolant would generate a number of gas bubbles rising to the surface of the sodium pool in the primary system

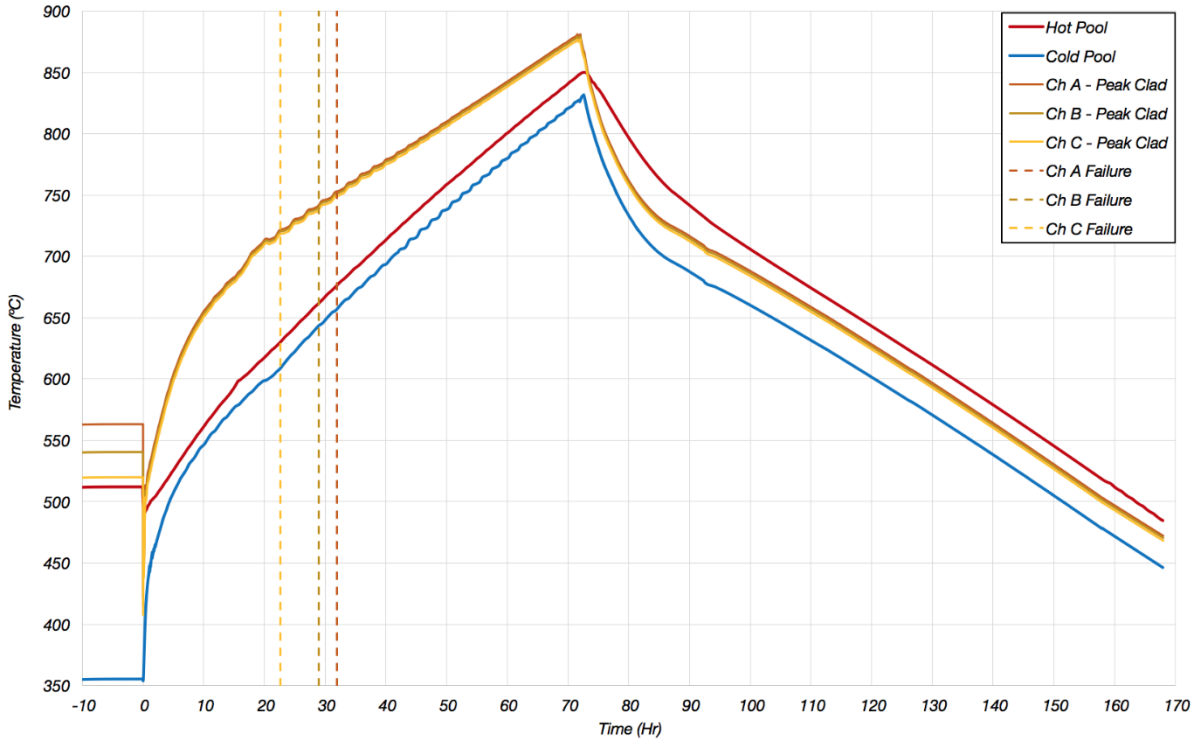


**Figure 4-11 – Primary System and Peak Cladding Temperature Excursions [17]**

The performance of the fuel cladding as a release mitigation measure is supported by the significant thermal inertia that exists in an advanced SFR design. The heatup of the sodium in the reactor system progresses over nearly 3 days before conditions arise that support cladding failure in the result shown in Figure 4-11. The specific timing will be sensitive to conditions of operation for fuel (e.g., the fuel burnup) since this determines the internal pressure in fuel pins. While the details of this result need to be re-evaluated using integral plant response codes, such

as MELCOR, the overall delay in progression to fission product release from fuel is a release mitigation strategy for an advanced SFR. This is demonstrated in the SAS4A/SASSYS-1 simulation shown in Reference [17] through the accident recovery observed with restoration of DRACS at 72 hours following the initial loss of primary system flow.

2. Retention of dissolved fission products in fuel

   In the absence of restoration of decay heat removal function, the fuel temperature increase would continue to progress. Fuel degradation would thus occur, the nature of which would differ between oxide and metal alloy fuels [37].

   - For oxide fuels, degradation would be similar to LWRs, with the exception that the metallic cladding in an advanced SFR would not chemically react with the sodium at high temperatures
     - Chemical interactions between the metal cladding (stainless steel) and the oxide fuel would occur and lead to liquefaction of the oxide fuel at temperatures well below its melting point
     - The low temperature liquefaction of the oxide fuel would be at a higher temperature than for LWR oxide fuel due to the absence of the low temperature monotectic interaction that occurs between clad and fuel in LWRs
   - For metal alloy fuels, degradation would be significantly different because of the distinctly different eutectic interactions that would occur at elevated temperatures
     - Prior to clad melting, chemical interactions between the Fe and Ni in the cladding and the Zr, Pu and U in the fuel would be promoted by exothermic formation of Laves phases and the heat of dilution between fuel and clad constituents
     - The combined clad-fuel chemical system would further support eutectic reactions that would liquefy fuel and cause cladding failures at relatively low temperatures (i.e., below the clad melting temperature)

   During the degradation of fuel and cladding, the high temperatures achieved would support diffusive transport of dissolved fission products. Radionuclides that have diffused to either the exposed surfaces of cladding or melt would then vaporize into the coolant.

   It is likely that the molten material would ultimately relocate into the lower plenum of the reactor vessel.

3. Retention of fission products within quiescent sodium pools and the reactor system

   Across a range of stages in the accident progression, fuel and fuel debris in the reactor will be submerged in sodium. Radionuclide vapor or particulates released from fuel into sodium can be dissolved or entrained in the sodium pool. Radionuclides dissolved in the sodium pool could deposit onto structural surfaces through precipitation or nucleate to form particles suspended in the pool. Thus, the radionuclides remaining in the pool could be either dissolved or suspended. Release to the cover gas above the pool would thus occur through vaporization or mechanical entrainment at the pool surface.

   Fission products emerging from the pool would then be subject to a number of transport processes similar to those occurring in LWRs. These fission products would emerge into an atmosphere considerably cooler than the sodium pool. Particle growth through vapor condensation and coagulation would tend to promote enhanced deposition of fission products

on structural surfaces in the reactor system. The deposition mechanisms of gravitational settling, inertial impaction, thermophoresis, diffusion and diffusiophoresis, also relevant in LWRs, would occur. Diffusiophoresis could be a significant mechanism in situations where substantial condensation of sodium vapor occurs on structural surfaces in the reactor system.

As in LWRs, fission products deposited on structural surfaces can re-evolve into the atmosphere due to

- Resuspension through sudden increases in gas flows or through other mechanical disturbances of the surfaces on which the fission products have deposited
- Revaporization of deposited fission products can occur as the temperature of the deposition surfaces increases due to convective heat transfer from gas flows out of the sodium pool and decay heat rejection from deposited fission products

4. Retention of fission products within the reactor enclosure

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**NOTE:** In the event that a bypass pathway between the reactor system and the environment exists, the reactor enclosure would not provide any additional release mitigation.

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As in the case of the reactor system, the transport processes that facilitate mitigation of releases to the environment are similar to those in LWRs. There are some notable deviations arising from the fact that fission products will be carried into the reactor enclosure in sodium vapor streams. The enclosure atmosphere will initially be comprised of air with some relative humidity. Since sodium vapor reacts with air and water, the exiting stream of contaminated sodium vapor will quickly form high number densities of aerosols, which will tend to rapidly agglomerate. Distinct from LWRs, the aerosol sizes in this case will be in excess of 20 μm. Fission product vapors will deposit on the surface of these large aerosol particles. Additionally, fission product particles will tend to coagulate with the aerosol particles. Such large aerosol particles will tend to rapidly settle to the containment floor due to gravity. Such rapid settling is distinct from LWRs where smaller aerosol particles are typical in containment.

Sodium fires could occur due to the reaction of sodium with air or water vapor in the reactor enclosure. In more severe cases, reactor enclosure boundary liners exposed to sodium could rupture. Concrete beneath the liner would react with sodium and core debris to produce off-gases, similar to what occurs when molten core interacts with concrete in an LWR. Sparging of these off-gases through fuel-containing debris would promote evolution of fission products into the atmosphere of the reactor enclosure. In addition, re-evolution of fission products could occur through revaporization and resuspension of fission products dissolved or suspended in sodium. Interactions with concrete will tend to enhance the fission product release from sodium pools due to the off-gases sparging through the sodium.

5. Retention of fission products through reactor enclosure leakage pathways to the environment

The leakage of fission products from the reactor enclosure is generally similar to that expected for LWRs. Attenuation of fission product leakage will occur for discharges through small cracks or small diameter pipes due to deposition [29] [30].

#### 4.2.4.2. Unprotected Transient Overpower Event Scenario

The considerations relevant to the assessment of advanced SFR release mitigation strategies for a PLOF event are generally applicable to their assessment of a UTOP event. A primary difference for this type of scenario is the timing of fission product release from the fuel. The progression to fuel

damage prior to the reactor reaching a subcritical state leads to much more energy being present in the fuel. This results in an earlier progression to fuel damage and subsequent accident progression.

The release of fission products from the fuel occurs at the onset of the accident. Radionuclides, particularly noble gases, will reach the cover gas very soon after accident initiation. Migration of radionuclides into the reactor enclosure will thus occur much earlier than the PLOF scenario. The initial inventory distributed throughout the sodium, cover gas and reactor enclosure regions will undergo radioactive decay. In the PLOF scenario, the decay of these short-lived fission products happens in the fuel after reactor shutdown because of the much more significant time to onset of fuel damage. Typically, the release of noble gases from the fuel into the reactor enclosure is more severe for this UTOP scenario relative to the PLOF transient—these fission products will experience radioactive decay during the first few hours after reactor shutdown. Radioactive decay serves as one means of removing fission products in the early time frame of this scenario.

Due to the more significant fuel melting, release of not just noble gases is expected for this scenario. While not all of the core is expected to melt for an unprotected scenario, it is assumed in the discussion below that core melting is equivalent to the extent expected for severe in-vessel core damage events in LWRs.

### *4.2.5. Summary of Dominant Factors Influencing Release Mitigation*

Fission product release characteristics for advanced SFRs have many features in common with LWRs. However, there are some notable areas where fission product release differs from LWRs.

- Fission product release across a range of the likely more dominant severe accident scenarios (protected transients) could be delayed by as much as 3 days due to the significant thermal capacitance of the sodium coolant. While this achieves significant coping time, it must be complemented by detailed strategies to restore decay heat removal functions. As found at Fukushima Daiichi, long-term accident management can be significantly complicated by how a plant responds to the unique common mode features of an external event. Even though a plant may be robust from the perspective of an internal events PRA, a significant compromise of safety function can be realized due to an external event. In addressing the insights from Fukushima Daiichi, the U.S. nuclear industry adoption of the FLEX strategy has been one means for nuclear power plants to establish a process by which to systematically restore long-term safety functions. Equivalent strategies for an SFR would be necessary for the delay of accident progression to be a relevant strategy to mitigate fission product release.
- Fission product release timing, however, is very sensitive to the operation of the reactivity control function. Adequately characterizing the potential for unprotected events to occur, establishing adequate reliability criteria and safety design robustness of shutdown systems, and developing programs to monitor and maintain shutdown system performance throughout operation is relatively critical for this design concept.

In addition to these considerations, currently available estimates of SFR source terms pre-date the significant enhancements to LWR source term modeling that have occurred in the past 30 years. As an example, the PRISM PRA [15] provides a number of source calculation results. Across a range of events, the magnitude of noble gas releases is generally consistent with that identified for LWRs. The timing of the release is generally delayed relative to LWRs for protected events, as noted above. The release of volatile fission products, however, is generally greater than that established for current state-of-the-art LWR source term evaluations. The magnitude of these volatile releases is similar to that identified for LWRs prior to the State-of-the-Art Reactor Consequence Analyses

(SOARCA) study. This study incorporated the significant enhancements to fission product release modeling in the past 30 years with experimental programs such as Phebus. The source term evaluations conducted for the PRISM PRA, thus, require re-evaluation in light of the significant enhancement in fission product modeling that has occurred in the past 30 years.

Mechanistic source term evaluations for advanced SFRs are already planned as part of the NRC vision and strategy for source term modeling for non-LWRs. This work would incorporate the fission product modeling enhancements that have been incorporated into MELCOR that led to the notable refinements in source term estimates found in the SOARCA study.

### *4.2.6. Summary of SFR Source Term Reduction Factor Estimates*

For the protected scenarios considered, fuel cladding failure is assumed to occur for all fuel pins based on analysis presented in Reference [17]. However, the availability of long-term cooling ensures that further progression of the event is arrested and more extensive fuel damage does not occur. For simplicity, focusing on an order of magnitude estimate of consequence, the subsequent release from the fuel matrix is assumed to be similar to that for an LWR experiencing only clad failures. This is acknowledged to be a highly simplified assumption. However, further refinement of this evaluation is being performed as part of the evolution of the MELCOR code system to provide mechanistic estimation of SFR source terms [22].

While there is also an indication that there could be enhanced removal of fission products in the reactor enclosure, as noted above, this simplified evaluation assumes that the fission product removal from the enclosure atmosphere is similar to that for LWRs. Without enclosure failure, it is assumed that fission product release to the environment is limited. As in the example of the HTGR above, the reduction factor is assumed to be 0.001. In events where reactor enclosure failure has occurred (e.g., possibly as the result of a sodium fire), some reduction of fission product release can occur due to deposition in the release pathway. For evaluation purposes, a reduction factor equivalent to that for deposition in a reactor enclosure bypass pathway is assumed. This value is 0.4 [31], with the exception of noble gases[4].

Table 4-5 presents the SFR source term reduction factor estimates for a protected scenario, assuming insignificant retention of fission products within the sodium coolant. Reduction factors are provided assuming an intact and an impaired reactor enclosure.

**Table 4-5. Summary of SFR Source Term Reduction Factor Complement Estimates for Protected Loss of Flow Scenario – Insignificant Retention within Sodium Coolant (based on Reference [31] with adaptation to represent insignificant retention of fission products in sodium coolant)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0.95 | 0 | 0.999 | 0 |
| Halogens (I, Br) | 0 | 0.95 | 0 | 0.999 | 0.6 |
| Alkali Metals (Cs, Rb) | 0 | 0.95 | 0 | 0.999 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0 | 1 | 0 | 0.999 | 0.6 |

[4] Noble gases will pass through a release pathway without any attenuation.

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Barium, Strontium (Ba, Sr) | 0 | 1 | 0 | 0.999 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 1 | 0 | 0.999 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0 | 1 | 0 | 0.999 | 0.6 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 1 | 0 | 0.999 | 0.6 |

In the case of an unprotected scenario, more significant fuel damage (fuel melting) is assumed to occur. The fuel cladding is assumed to provide negligible retention of fission products. Based on assessments in Reference [17], approximately 30% of the fuel melts in this scenario. The fission product reduction factors for the fuel matrix, for this unprotected scenario, are approximated by the LWR fuel matrix reduction factors, accounting for an approximate 30% additional reduction due to an assumed more limited fuel melting than encountered in an LWR event with severe in-vessel core damage. This is based on the assumption that fuel melting involves about 70% less material than the LWR scenario.

**Table 4-6. Summary of SFR Source Term Reduction Factor Complement Estimates for Unprotected Transient Overpower Scenario – Insignificant Retention within Sodium Coolant (based on Reference [31] with adaptation to represent insignificant retention of fission products in sodium coolant)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0.715 | 0 | 0.999 | 0 |
| Halogens (I, Br) | 0 | 0.895 | 0 | 0.999 | 0.6 |
| Alkali Metals (Cs, Rb) | 0 | 0.925 | 0 | 0.999 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0 | 0.985 | 0 | 0.999 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0 | 0.994 | 0 | 0.999 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0.99925 | 0 | 0.999 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0 | 0.99985 | 0 | 0.999 | 0.6 |
| Lanthanides (La, | 0 | 0.99994 | 0 | 0.999 | 0.6 |

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | | | | | |

The role of the sodium coolant in retaining the released fission products within the reactor vessel is a topic of additional uncertainty. As noted in Reference [18], past SFR events with fuel melting have found relatively significant retention of all radionuclide groups within the fuel and sodium coolant, with the exception of noble gases.

Table 4-7 presents the reduction factor estimates assumed for a sensitivity with all but the noble gases retained within the sodium coolant of the reactor vessel.

**Table 4-7. Summary of SFR Source Term Reduction Factor Complement Estimates for Protected Loss of Flow Scenario – Significant Retention within Sodium Coolant (based on Reference [31] with adaptation to represent significant retention of fission products in sodium coolant)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0.95 | 0 | 0.999 | 0 |
| Halogens (I, Br) | 0 | 0.95 | 1 | 0.999 | 0.6 |
| Alkali Metals (Cs, Rb) | 0 | 0.95 | 1 | 0.999 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0 | 1 | 1 | 0.999 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0 | 1 | 1 | 0.999 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 1 | 1 | 0.999 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0 | 1 | 1 | 0.999 | 0.6 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 1 | 1 | 0.999 | 0.6 |

Table 4-8 presents the similar sensitivity for an unprotected scenario. With the exception of the noble gases, all other classes of radionuclides are assumed to be retained within the sodium coolant.

**Table 4-8. Summary of SFR Source Term Reduction Factor Complement Estimates for Unprotected Transient Overpower Scenario – Significant Retention within Sodium Coolant (based on Reference [31] with adaptation to represent significant retention of fission products in sodium coolant)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor Enclosure | Impaired Reactor Enclosure |
|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0.715 | 0 | 0.999 | 0 |
| Halogens (I, Br) | 0 | 0.895 | 1 | 0.999 | 0.6 |
| Alkali Metals (Cs, Rb) | 0 | 0.925 | 1 | 0.999 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0 | 0.985 | 1 | 0.999 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0 | 0.994 | 1 | 0.999 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0.99925 | 1 | 0.999 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0 | 0.99985 | 1 | 0.999 | 0.6 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0.99994 | 1 | 0.999 | 0.6 |

### *4.2.7. Sample SFR Consequence Assessment*

Figure 4-12 presents an illustrative comparison of SFR off-site consequences against a reference LWR off-site consequence assessment. The consequence metric chosen for evaluation is the off-site dose. The doses presented are normalized to the peak off-site dose for the LWR. Overall, SFR consequences are generally lower than those for the reference LWR. Evident from these results is the crucial role potentially played by the sodium in retaining fission products. Assuming more significant retention of all but the noble gases results in a reduction in off-site consequences by nearly three orders of magnitude. This relatively large variation in potential off-site consequences reflects a critical knowledge gap. Fission product retention in sodium could ameliorate the extent of release to the environment. However, with limited understanding of radiochemistry for sodium systems, it is not known if other chemical processes could arise that would increase the volatility of some radionuclides.

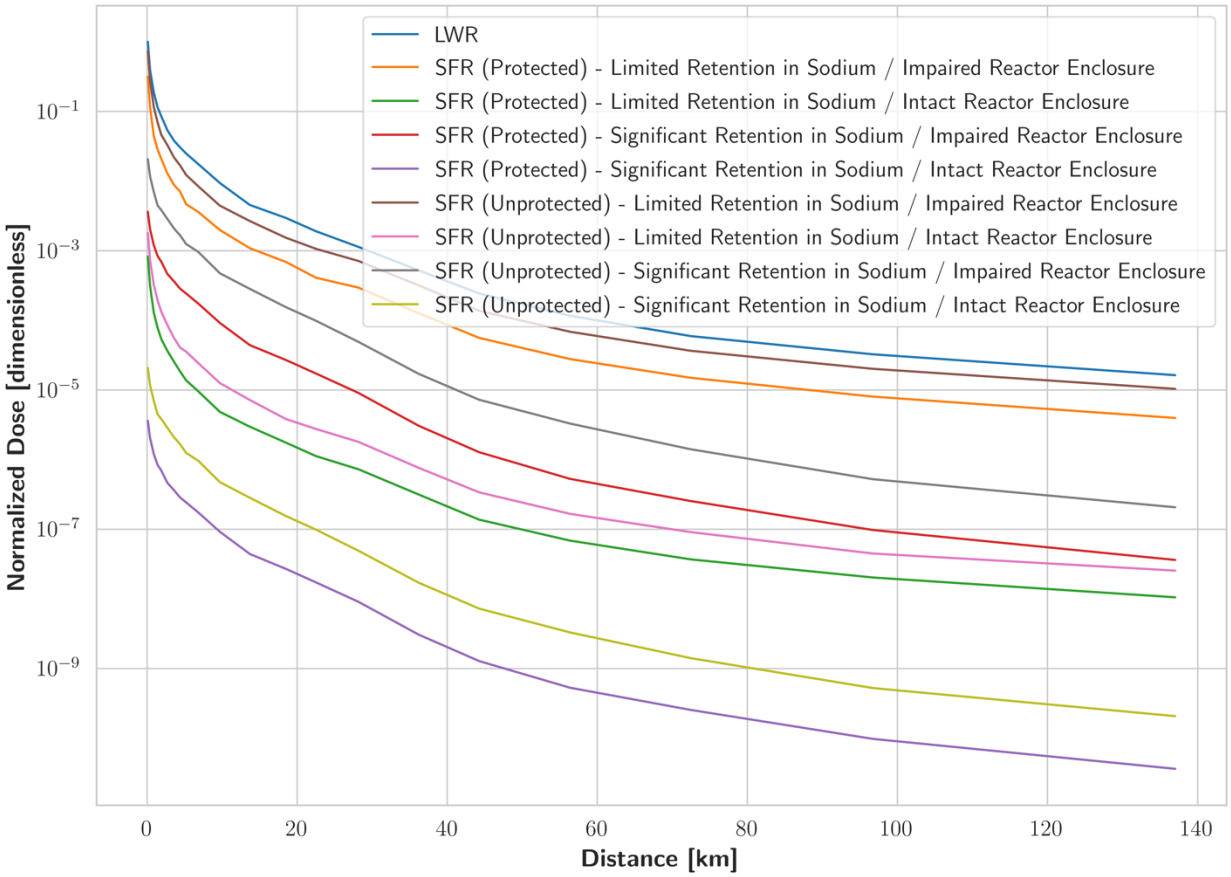


**Figure 4-12. Comparison of SFR Off-Site Dose Evaluations with Reference LWR Evaluation**

The evaluation shown in Figure 4-12 assumes that the fission product inventory is rescaled relative to a typical LWR fission product inventory. To assess the implications of fission product retention measures independent of fission product inventory, an additional, illustrative sensitivity is performed assuming an LWR initial fission product inventory. The off-site dose is shown as a function of distance from the release point in Figure 4-13. In a situation with limited retention of fission product in the sodium coolant, and an impaired reactor enclosure, there is a somewhat higher off-site consequence relative to the LWR. This is due to the limited additional fission product retention applied for the SFR concept in the event that both the sodium coolant and the reactor enclosure provide more limited mitigation of releases. This further emphasizes the important role that fission product retention in sodium coolant can play in an overall SFR strategy for mitigating releases to the environment.

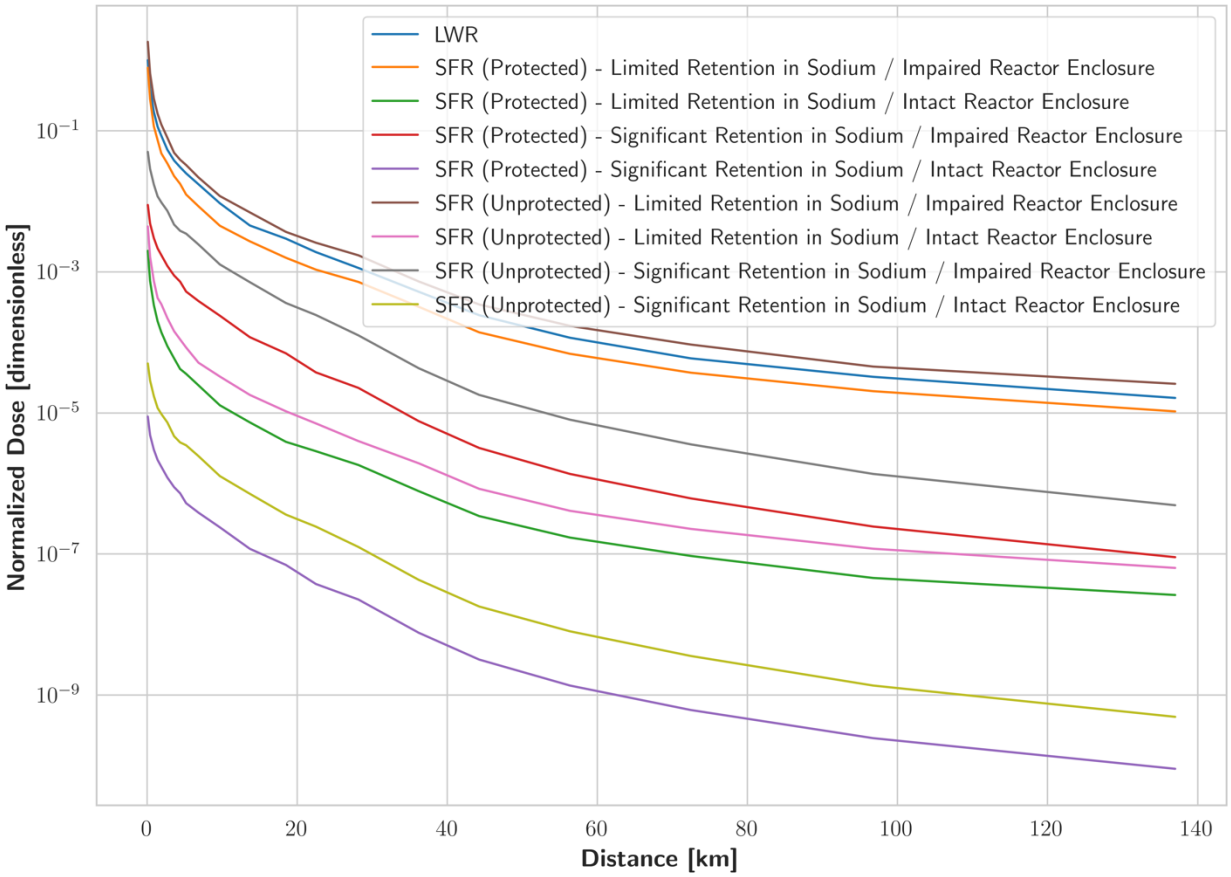


**Figure 4-13. Comparison of SFR Off-Site Dose Evaluations with Reference LWR Evaluation assuming Initial SFR Fission Product Inventory Equivalent to the LWR Fission Product Inventory**

## 4.3. MSR Source Term Assessment

The MSR considered in this document is a molten salt-fueled system. There are alternate systems that use molten salt as the coolant in conjunction with some type of solid fuel design, such as TRISO fuel. These are not considered in this document. Unlike sodium, these reactors generally do not exhibit phenomena driven by reaction of the coolant with water or air (e.g., sodium fires). MSR design concepts have the following features.

- Low pressure operation
- Relative to LWRs, there is a smaller volume of waste production due to more efficient utilization of fuel
- Passive cooling
- Adoption of intermediate loops to separate working fluids
- Support for a range of power cycles
- Higher outlet temperatures and greater thermal efficiencies relative to currently operating LWRs

In addition to these general features of MSR designs, they achieve a number of additional benefits relative to LWRs through the use of a different working fluid than water. Molten salts have a greater heat capacity than water (they also have a larger Prandtl number). As a result, they can store more energy and tend to transport energy more readily via convection relative to conduction –

momentum diffusivity dominates thermal diffusivity in molten salts. In addition, certain salts (e.g., fluoride salts) are chemically stable having low volatility at high temperature, do not react with air or water, are stable when subjected to radiation fields, exhibit good fission product retention, have high solubility for U/Th fluorides, and have reasonable neutronic properties.

### 4.3.1. Phenomenological Considerations

The relevant physical processes governing fission product release and transport are illustrated in Figure 4-14. These are largely similar to those introduced above for HTGRs and SFRs.

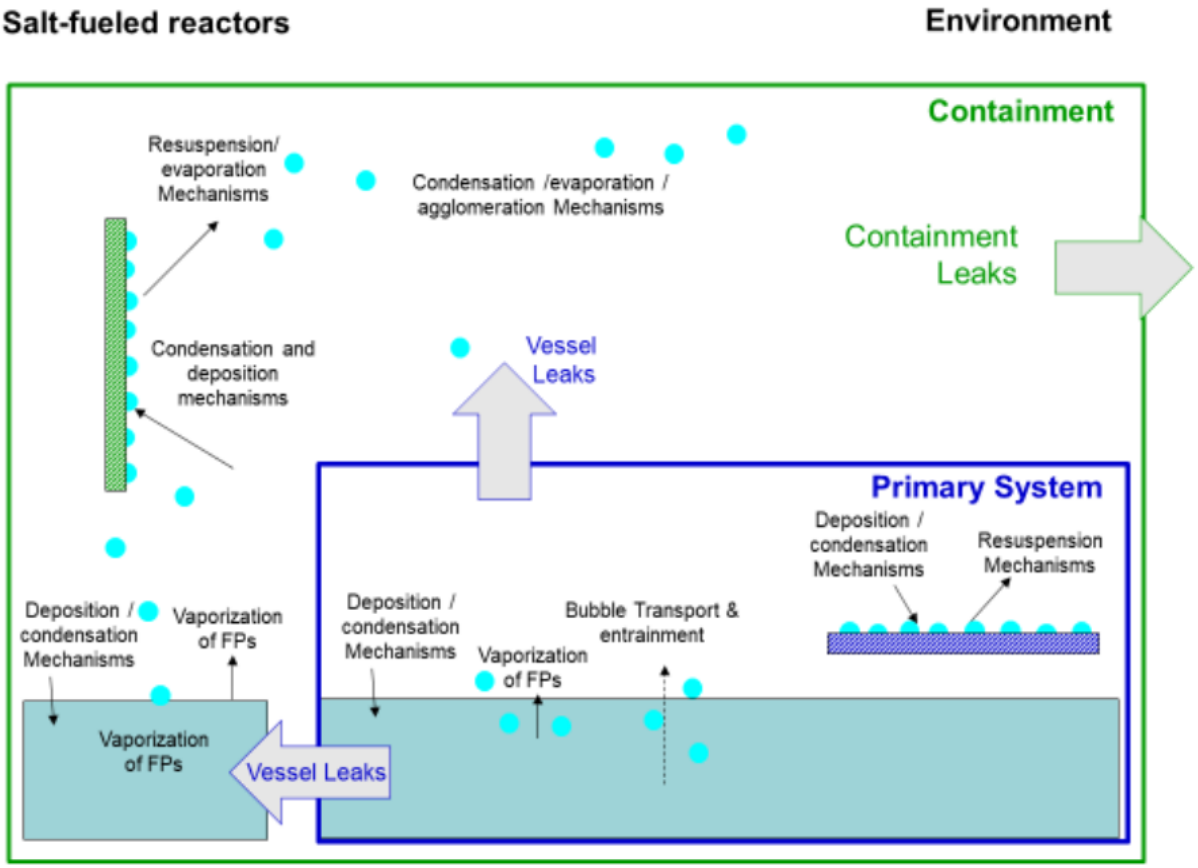


**Figure 4-14 – Summary of Radionuclide Release and Transport Processes for Molten Salt Reactors [22]**

### 4.3.2. Initiating Events

There are two event scenario classes considered in this example scoping source term evaluation of the MSR design concept.

1. A primary coolant leak (PCL) is a break in the primary system piping that discharges molten salt into the reactor cell; a range of molten salt drainage fractions could be realized depending on the nature of the break
   a. A small break results in drainage of about 5% of the molten salt inventory over the course of a day (the MSRE experienced a small fuel salt leak at a freeze flange)

b. A recirculation line break results in drainage of about 40% of the molten salt inventory into the reactor cell over the course of 15s
c. A drain line break results in discharge of 100% of the inventory into the reactor cell over the course of 370s
d. Coincident recirculation and drain line breaks result in drainage of 100% of the inventory into the reactor cell over the course of 280s

For these primary coolant leak scenarios, the response of the reactor cell is also sensitive to whether or not water ingress occurs. Water interacting with molten salt in the reactor cell will generate steam. This additional source of pressurization in the reactor cell could lead to its eventual failure. Over-pressurization of the reactor cell is intended to be relieved through the vapor-condensing system, which provides a scrubbed release pathway. In the event that the vapor-condensing system is unavailable, reactor cell overpressure failure will result in unscrubbed release of contamination to the environment. MSRE experienced numerous water leaks into the reactor cell from the containment cooling system.

2. A loss of heat removal (LOHR) will result in drainage of the molten salt to the drain tanks located in a separate compartment called the drain cell. With loss of heat removal function, the salt may eventually fail the drain tanks and drain to the drain tank cell floor. As in the case of the reactor cell, drain tank cell failure may result if water ingress occurs and provides a steam source to pressurize the cell to failure. As in the case of the reactor cell, over-pressurization of the drain tank cell is intended to be relieved through passage of gases through the vapor-condensing system. In the event that the vapor-condensing system is not available, drain tank cell failure will result in an unscrubbed release to the environment.

### *4.3.3. Initial Fission Product Inventory*

Table 4-9 provides a summary of a reference MSR fission product inventory in comparison to a typical LWR inventory extracted from the Peach Bottom SOARCA study. While there is a reduced inventory available for release, this reduction in inventory combined with dispersion in the environment would not be sufficient to eliminate the potential for off-site public health consequences.

**Table 4-9 – Representative MSR Fission Product Inventory**

| Radionuclide Class | Representative Element | Member Elements | Peach Bottom SOARCA Radioactive Mass [kg] | Reference MSR Radioactive Mass [kg] |
|---|---|---|---|---|
| Noble Gas | Xe | He, Ne, Ar, Kr, Xe, Rn, H, N | 531.7 | 5 |
| Alkali Metals | Cs | Li, Na, K, Rb, Cs, Fr, Cu | 323.0 | 3 |
| Alkaline Earths | Ba | Be, Mg, Ca, Sr, Ba, Ra, Es, Fm | 235.6 | 2 |

| Radionuclide Class | Representative Element | Member Elements | Peach Bottom SOARCA Radioactive Mass [kg] | Reference MSR Radioactive Mass [kg] |
|---|---|---|---|---|
| Halogens | I | F, Cl, Br, I, At | 19.9 | 0 |
| Chalcogens | Te | O, S, Se, Te, Po | 49.1 | 0 |
| Platinoids | Ru | Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni | 342.8 | 3 |
| Early Transition Elements | Mo | V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W | 400.2 | 4 |
| Tetravalent | Ce | Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C | 1555.5 | 16 |
| Trivalents | La | Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf | 1793.7 | 18 |
| Uranium | U | U | 132794.0 | 1328 |
| More Volatile Main Group | Cd | Cd, Hg, Zn, As, Sb, Pb, Tl, Bi | 6.6 | 0 |
| Less Volatile Main Group | Sn | Ga, Ge, In, Sn, Ag | 9.6 | 0 |

### *4.3.4. Source Term Behavior in Given Accident Scenarios*

Advanced MSRs incorporate the following barriers that serve to mitigate the release of fission products to the environment across a range of event scenarios. The different scenarios considered above share similarities and will be treated in common in the following discussion.

1. Retention of fission products dissolved in the molten fuel

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**NOTE:** The molten salt-fueled reactor does not possess a fuel cladding as in solid-fueled systems. Thus, this particular release mitigation barrier cannot be considered for this system.

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In either a primary coolant leakage or loss of heat removal event, the primary system will develop a breach that opens a pathway to the reactor cell or the drain tank cell, respectively.

Fission product gases (i.e., noble gases) will be able to escape into either the reactor cell or the drain tank cell. Fission products that have been dissolved in the fuel will only be able to escape into the reactor or drain tank cells through vaporization from the surface of the pool. As examined for the MSRE, there is evidence that many of these dissolved fission products can be stabilized in solution in the molten salt. There remains uncertainty regarding the solubility of these fission products at higher temperatures. Additional chemical reactions that would promote chemical forms that are not soluble in the molten salt are not well known at present. As noted in Section 3, these chemical reactions could lead to evolution of these fission product gases out of the molten salt. Since a key release mitigation measure for MSRs is the potential high solubility of many fission products in the molten salt, clearer understanding of the radiochemistry of proposed molten salts is needed.

2. Retention of fission products within frozen molten salt beds

   Freezing of molten salt in the reactor, drain tank, reactor cell or drain tank cell will ensure that molten salt temperatures are low enough to prevent any appreciable vaporization of fission products from the free surface of these salt beds. Since reactions of the molten salt with interfacing structures is not likely, there will be no additional means to sparge fission products from these beds as in the case of MCCI in water-moderated reactors.

3. Retention of fission products within the reactor or drain tank cell atmospheres

   Fission product gases leaving the molten salt will remain in the atmosphere of either the reactor or drain tank cells. Any discharge from these cells will transport these fission product gases to the environment.

   The presence of fission product vapors in these cells will be mitigated in a manner similar to that occurring for LWRs. Deposition of fission product vapors on surfaces will promote fission product retention. In the event that steam is present in the atmosphere, diffusiophoresis will be a dominant means of fission product deposition on surfaces.

4. Retention of fission products through leakage pathways to the environment

   All noble gases will pass through the vapor-condensing system. Fission product vapors passing through the vapor-condensing system will be substantially attenuated prior to release to the environment.

### *4.3.5. Summary of Dominant Release Mitigation Factors*

MSRs possess the following key release mitigation measures.

- High solubility of many fission products in the molten salt working fluid
- Scrubbing of fission product vapor that pass through the vapor-condensing system

These are the principal enhancements to the overall release mitigation strategy of an MSR. There remains uncertainty regarding the solubility of fission products in the molten salt due to the relatively limited understanding of the radiochemistry for these working fluids.

Should greater vaporization from the molten salt free surface be possible, additional consideration of the thermal state of molten salt beds on the floor of the reactor or drain tank cells would be needed. Higher temperatures of these beds would promote greater transport of fission products to the bed surfaces and as a result greater vaporization of fission products into the respective atmospheres. The current MSR design concepts are limited to relatively low operating powers. At these levels, it is reasonable to assume that molten salt beds on the floors of the reactor or drain

tank cells would freeze. At power levels in excess of 500 MW(t), additional consideration of the heatup of the reactor and drain tank cells would be required to ensure that there is an adequate heat transfer surface area to volume ratio to promote dissipation of decay heat to the environment. Unmitigated heatup of the molten salt beds would promote enhanced vaporization of fission products to the reactor or drain tank cell atmospheres.

### 4.3.6. Summary of MSR Source Term Reduction Factor Estimates

The MSR accident scenarios considered above assume that the molten salt progresses to a state where it is ultimately frozen. Long-term fission product release from this configuration is not expected for all but the noble gases, which will be released relatively soon after the initiation of the event that terminates operation of the off-gas system.

Since MSRs have fuel contained within the molten salt, there is no fuel structure reduction factor. This is always assumed to be inconsequential for MSRs as a result. The molten fuel serves as the fuel matrix. Release from the molten salt could largely be noble gases based on the potential for high solubility of other radionuclide classes within molten salts. In the first case, it is assumed that there is limited additional solubility of fission products in the salt. Releases are assumed to conservatively be bounded by reduction factors for LWRs experiencing severe in-vessel core damage. In the second case, all fission products but the noble gases are assumed to have high solubility in the molten salt. Solubility of fission products in molten salts will be significantly refined during effort to evolve MELCOR to performing mechanistic calculations of molten salt source terms [22].

When fission products are released from the molten salt, it is assumed that they are transported into either the reactor cell or drain cell with negligible retention in the reactor vessel or drain tanks. This is a simplifying assumption accounting for the fact that the off-gas system is assumed to have ceased operation as a result of the event.

Fission products that have entered containment are assumed to behave in a manner similar to that in LWRs. Two reactor or drain cell configurations are considered—intact and impaired. Additionally, a filtered release from the reactor/drain cell, through the vapor condensing system, is assumed as a third reactor enclosure configuration. For a filtered release, a reduction factor of 0.01 [31] is assumed, for all but the noble gases. For the intact reactor/drain cell, a reduction factor of 0.001 is assumed [31]. For an impaired containment, for all but the noble gases, a reduction factor of 0.4 is assumed to reflect some retention along the release pathway.

Table 4-10 presents the MSR source term reduction factor estimates assuming no retention of fission products other than noble gases within the salt. The reduction factor for noble gases is assumed to be 0.95, consistent with that for LWRs experiencing significant in-vessel core damage.

**Table 4-10. Summary of MSR Source Term Reduction Factor Complement Estimates – Insignificant Retention within Salt (based on Reference [31] with adaptation to represent insignificant retention of fission products in molten salt fluid)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor/Drain Cell | Filtered Release | Impaired Reactor/Drain Cell |
|---|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0.05 | 0 | 0.999 | 0 | 0 |
| Halogens (I, Br) | 0 | 0.65 | 0 | 0.999 | 0.99 | 0.6 |
| Alkali Metals (Cs, Rb) | 0 | 0.75 | 0 | 0.999 | 0.99 | 0.6 |

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor/Drain Cell | Filtered Release | Impaired Reactor/Drain Cell |
|---|---|---|---|---|---|---|
| Tellurium Group (Te, Sb, Se) | 0 | 0.95 | 0 | 0.999 | 0.99 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0 | 0.98 | 0 | 0.999 | 0.99 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0.9975 | 0 | 0.999 | 0.99 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0 | 0.9995 | 0 | 0.999 | 0.99 | 0.6 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0.9998 | 0 | 0.999 | 0.99 | 0.6 |

Table 4-11 presents the MSR source term reduction factor estimates assuming significant fission product retention within the salt for all radionuclide groups except the noble gases. The reduction factor for noble gases is assumed to be 0, with all noble gases released from the molten salt.

**Table 4-11. Summary of MSR Source Term Reduction Factor Complement Estimates – Significant Retention within Salt (based on Reference [31] with adaptation to represent insignificant retention of fission products in molten salt fluid)**

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor/Drain Cell | Filtered Release | Impaired Reactor/Drain Cell |
|---|---|---|---|---|---|---|
| Noble Gases (Kr, Xe) | 0 | 0 | 0 | 0.999 | 0 | 0 |
| Halogens (I, Br) | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |
| Alkali Metals (Cs, Rb) | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |
| Tellurium Group (Te, Sb, Se) | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |
| Barium, Strontium (Ba, Sr) | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |
| Cerium Group (Ce, Pu, Np) | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, | 0 | 1 | 0 | 0.999 | 0.99 | 0.6 |

| Radionuclide Group | Fuel Structure | Fuel Matrix | Reactor Vessel | Intact Reactor/Drain Cell | Filtered Release | Impaired Reactor/Drain Cell |
|---|---|---|---|---|---|---|
| Sm, Y, Cm, Am) | | | | | | |

### 4.3.7. Sample MSR Consequence Assessment

Figure 4-15 presents an illustrative comparison of MSR off-site consequences against a reference LWR off-site consequence assessment. The consequence metric chosen for evaluation is the off-site dose. The doses presented are normalized to the peak off-site dose for the LWR. Overall, MSR consequences are generally lower than those for the reference LWR. Evident from these results is the crucial role potentially played by the salt in retaining fission products. Assuming more significant retention of all but the noble gases results in a reduction in off-site consequences by nearly three orders of magnitude. This is a crucial consideration that must for evaluation of knowledge gaps in the modeling of accident progression and fission product transport for non-LWRs.

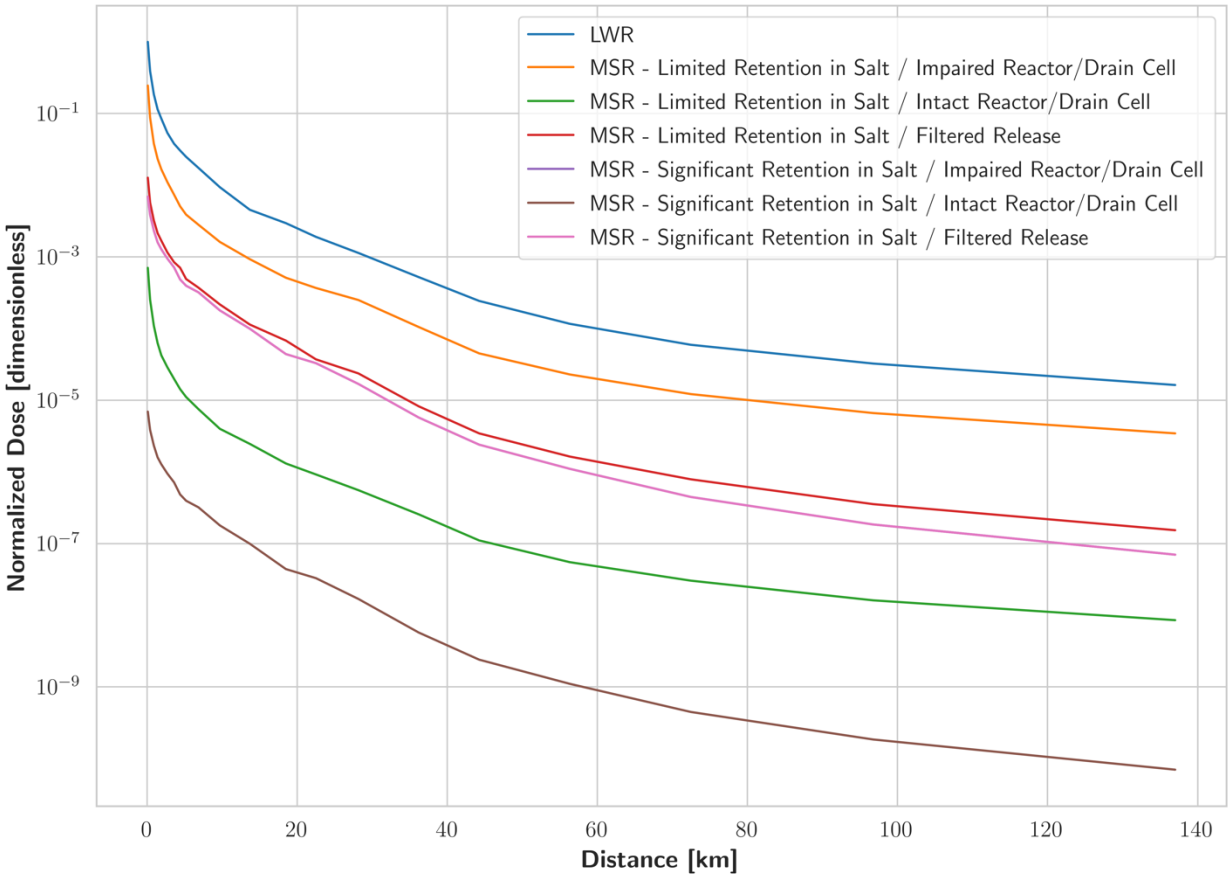


**Figure 4-15. Comparison of MSR Off-Site Dose Evaluations with Reference LWR Evaluation**

The evaluation shown in Figure 4-15 assumes that the fission product inventory is rescaled relative to a typical LWR fission product inventory. To assess the implications of fission product retention measures independent of fission product inventory, an additional, illustrative sensitivity is performed assuming an LWR initial fission product inventory. The off-site dose is shown as a function of

distance from the release point in Figure 4-16. In a situation with limited retention by the molten salt, and an impaired reactor enclosure, there is a somewhat higher off-site consequence relative to the LWR. This is due to the limited additional fission product retention applied for the MSR concept in the event that both the molten salt and the reactor enclosure provide limited retention of fission products. This further emphasizes the important role that fission product retention in molten salt can play in the overall strategy for mitigating releases to the environment for the MSR concept.

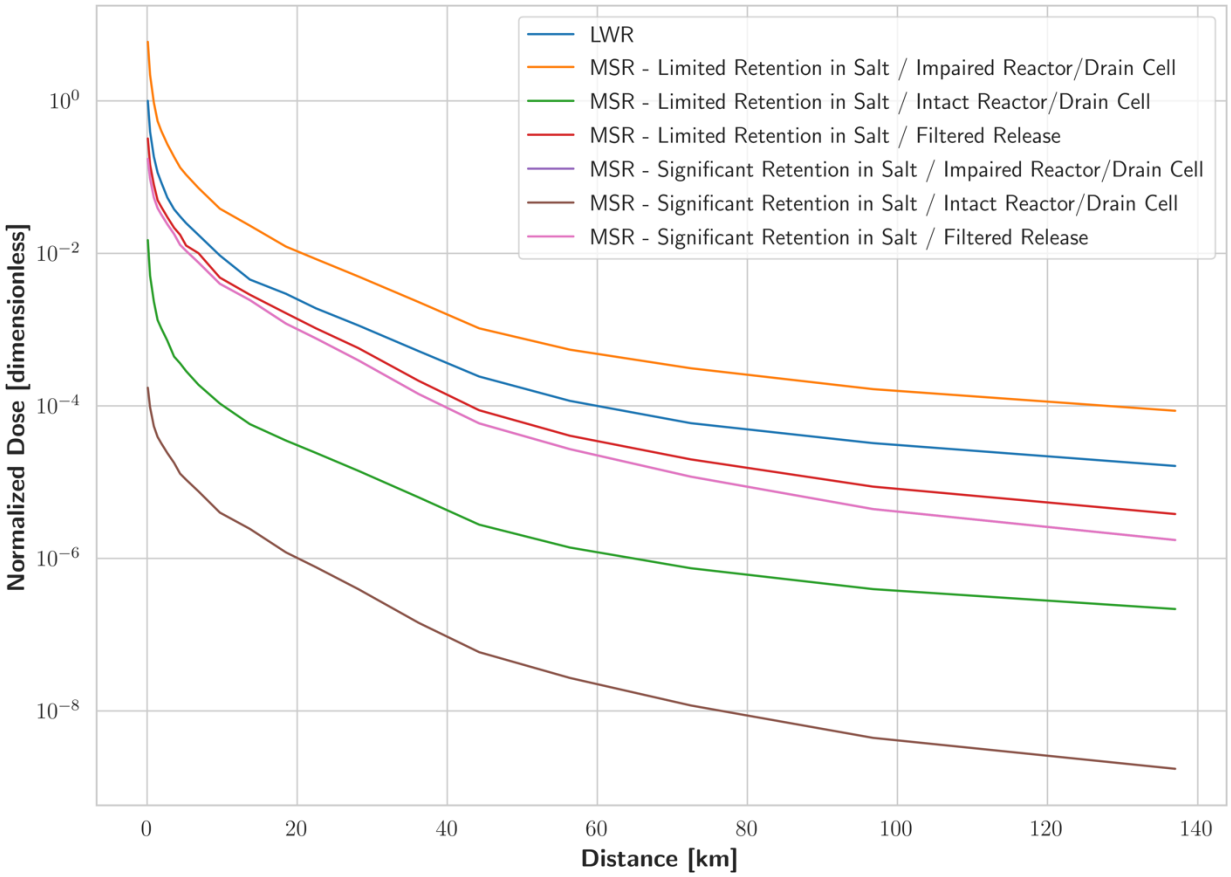


**Figure 4-16. Comparison of MSR Off-Site Dose Evaluations with Reference LWR Evaluation assuming Initial MSR Fission Product Inventory Equivalent to the LWR Fission Product Inventory**

# 5. SUMMARY

This report describes a structure to aid in evaluation of release mitigation strategies across a range of reactor technologies. The evaluation is conducted utilizing a scoping source term methodology to enable approximate assessment of off-site consequences. The scoping source term methodology is based on an estimated evaluation of the effectiveness of different fission product retention barriers. The off-site consequence assessment is performed using off-site dose as a function of distance from the plant as the consequence metric. The assessment performed for example reactor concepts utilizes previous studies of postulated accident sequences for each reactor concept. This simplified approach classifies release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance. It is not, however, intended to develop quantitative estimates of radiological release magnitudes and compositions to the environment based on a mechanistic evaluation of source term evolution. Rather, this approach is intended to identify the characteristics of a reactor design concept's release mitigation strategies that are most important to different classes of accident scenarios. The use of the scoping source term evaluation approach enables an assessment of the relative effect of different barriers that mitigate radiological release to the environment for non-LWRs.

To extend this evaluation framework, it is necessary that factors defining a chain of release attenuation stages, that form an overall release mitigation strategy, be established through mechanistic source term calculations. This has typically required the application of an integral plant analysis code such as MELCOR. At present, there is insufficient evidence to support a priori evaluation of the effectiveness of a release mitigation strategy for non-LWR concepts across the spectrum of events that could challenge the radiological containment function. While it is clear that these designs have significant margin to radiological release to the environment for the scenarios comprising the design basis, detailed studies have not yet been performed to assess the risk profile for these plants. Such studies would require extensive evaluation across a reasonably complete spectrum of accident scenarios that could lead to radiological release to the environment. The scoping source term approach could enable assessment of accident scenario characteristics having the most significant impact on the mitigation of radiological release for different non-LWR concepts.

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