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# NUCLEAR TECHNOLOGIES AND MATERIALS ADVANCED REACTOR CONCEPTS-20

# FAST MODULAR REACTOR SOURCE TERM METHODOLOGY


**Sponsored by the U.S. Department of Energy**
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| 2 | 2023/10/31 | ECN-114518: Added [[ ]] markings around proprietary information. |
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## TABLE OF CONTENTS

## LIST OF FIGURES

## LIST OF TABLES

# ACRONYMS

| Acronym | Definition |
|---|---|
| AEC | Atomic Energy Commission |
| ARC-20 | Advanced Reactor Concepts-20 |
| AMB | Active Magnetic Bearing |
| ASME | American Society of Mechanical Engineers |
| AST | Alternative Source Term |
| BWR | Boiling-Water Reactor |
| CCDF | Complimentary Cumulative Distribution Function |
| CEDE | Committed Effective Dose Equivalent |
| CFD | Computational Fluid Dynamics |
| CNS | Containment System |
| CV | Control Volume |
| DLOFC | Depressurized Loss of Forced Cooling |
| DOE | Department of Energy |
| DBA | Design Basis Accident |
| EAB | Exclusion Area Boundary |
| EDE | Effective Dose Equivalent |
| ESF | Engineered Safety Feature |
| FGR | Federal Guidance Report |
| FMR | Fast Modular Reactor |
| GA-EMS | General Atomics Electromagnetic Systems |
| GFR | Gas-cooled Fast Reactor |
| GWD/MTU | Giga Watt Days per Metric Ton Uranium |
| HALEU | High-Assay Low-Enriched Uranium |
| HP | High Pressure |
| HS | Heat Structure |
| HTGR | High Temperature Gas Reactor |
| LBE | Licensing Basis Event |
| LHS | Latin Hypercube Sampling |
| LMFBR | Liquid Metal Fast Breeder Reactor |
| LOCA | Loss of Coolant Accident |
| LP | Low Pressure |
| LPZ | Low-Population Zone |
| LWR | Light Water Reactor |
| MCS | Maintenance Cooling System |
| mSiC | monolithic SiC |

| Acronym | Definition |
|---|---|
| NRC | Nuclear Regulatory Commission |
| OD | Outer Diameter |
| PCS | Power Conversion System |
| PCU | Power Conversion Unit |
| PLS | Proportional Limit Stress |
| POF | Probability Of Failure |
| PWR | Pressurized Water Reactor |
| RVCS | Reactor Vessel Cooling System |
| RG | Regulatory Guide |
| RN | Radionuclide |
| rpm | revolutions per minute |
| SiC | Silicon Carbide |
| SME | Subject Matter Expert |
| SNL | Sandia National Laboratories |
| SOARCA | State-of-the-Art Reactor Consequence Analyses |
| SRM | Staff Requirements Memorandum |
| SSC | Structures, Systems, and Components |
| STCP | Source Term Code Package |
| TCG | Turbine-Compressor-Generator |
| TD | Theoretical Density |
| TEDE | Total Effective Dose Equivalent |
| TID | Technical Information Document |
| UDM | User-Defined Material |
| $UO_2$ | Uranium Dioxide |
| UTS | Ultimate Tensile Stress |
| UWM | University of Wisconsin Madison |
| $Zr_3Si_2$ | Zirconium Silicide |

## EXECUTIVE SUMMARY

This report presents the methodology used for radiation source term calculation and consequence analysis results of General Atomics Electromagnetic Systems (GA-EMS) 44-MWe Fast Modular Reactor (FMR). The analysis used MCNP 6.2 and ORIGEN 2.2 code to generate the reactor power and neutronics parameters and fission product inventory, respectively. MELCOR 2.2 and MACCS 4.2 codes were used for the accident analysis and radiation impact evaluation, respectively. An interfacing program was written to transfer data from MELCOR to MACCS.

In this report, the approach to develop mechanistic source terms is discussed, including information on radionuclide generation and transport in the core, primary circuit, and reactor building. The multiple barriers to radionuclide release are discussed, with particular emphasis on the role of the fuel systems as the primary barrier to radionuclide release. The data and models used in source term calculation are summarized, and the MELCOR modeling and sources of supporting data are reviewed. The data and models used in consequence analysis are summarized for the MACCS code along with interfacing methods with the MELCOR code.

A sample source term calculation was carried out for depressurized loss of forced cooling (DLOFC) accident. The sample calculation confirmed that the overall calculation procedure works properly, and the calculation results are consistent with data and models used for the analysis. Eventually, the methodology of estimating the source term and consequence by MELCOR-MACCS will provide results that can be used for the design improvement of the FMR system.

It is understood that the results of the source term calculation and consequence analysis will vary as there are uncertainties remaining in the data, models, computing tools, and the FMR design features and parameters. For the source term to be ultimately used for the licensing applications, it is recommended to continue and expand the source term calculation and consequence analysis as follows:

- Verify the data and models used for the source term calculation to be consistent with the FMR design.
- Evaluate other licensing basis events to identify the main characteristic and envelopes of the source term associated with the FMR design.
- Conduct the sensitivity analysis to the data and models to identify the major uncertainties significantly affecting the source term, including approximations used to model power distribution in the core.

# 1. INTRODUCTION

General Atomics Electromagnetic Systems (GA-EMS) is developing a helium-cooled Fast Modular Reactor (FMR).[1] The project has been selected by the U.S. Department of Energy (DOE) for Advanced Reactor Concepts-20 (ARC-20) program. The long-term goal is to design, license, and commercialize the FMR plant by the mid-2030s. To achieve this goal, early engagement with the Nuclear Regulatory Commission (NRC) is an important licensing strategy of the FMR project. As an effort to support the design and a part of the pre-application regulatory engagement plan, GA-EMS is developing source term methodology applicable to the FMR design.

The source term is a critical component of the overall assessment of the safety of a nuclear power plant, and it is used in the development of emergency response plans, evacuation plans, and other measures to protect public health and safety in the event of an accident. The NRC has used the source term in its policy and practices for licensing of nuclear reactors. Source term refers to the radioactive material released during a postulated nuclear reactor accident. Information of the postulated radiological release includes the containment leak rate and knowledge of the radioactive material composition and quantity, as well as the chemical and physical properties of the material within the containment.

The NRC's reactor site criteria in 10 CFR Part 100 require, for licensing purposes, that an accidental fission product release resulting from "substantial meltdown" of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated assuming that the containment remains intact but leaks at its maximum allowable leak rate.

As described in 10 CFR 52.47 (a)(2), the standard plant is expected to reflect extremely low probability accidents. An applicant shall perform an evaluation of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences.

The evaluation must determine that:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE),
- An individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product

[1] H. Choi et al., “The Fast Modular Reactor (FMR) - Development Plan of a New 50 MWe Gas-cooled Fast Reactor”, *Tran. Am. Nucl. Soc*. 124, 454–456, 2021.

release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

This report is intended to serve as the basis for interactions with the NRC on the source term methodology of the FMR. It should be noted that the FMR design is still evolving. As such, GA-EMS expects that NRC's review of this report focuses on the source term calculation and consequence analysis approaches rather than data and models. This report presents the source term calculation and consequence analysis approaches as follows:

- The approach to develop mechanistic source terms is discussed, including information on radionuclide generation and transport in the core, primary circuit, and reactor building.
- The multiple barriers to radionuclide release are discussed, with particular emphasis on the role of the fuel systems as the primary barrier to radionuclide release.
- The models and codes used in source term development are summarized, and the MELCOR modeling and sources of supporting data are reviewed.
- The models and codes used in consequence analysis are summarized for the MACCS with interfacing methods.
- A sample accident analysis is carried out to estimate the source term and consequences.

The NRC has issued multiple regulatory documents that are related to light water reactor (LWR) source term methodologies. These documents provide guidance on how to estimate the amount and type of radioactive material that can be released in the event of an accident specific to LWRs. However, they also provide the bases for the regulation of other types of reactors including High Temperature Gas Reactors (HTGRs) such as the FMR. The NRC states that the non-LWR applicants can use modern analysis tools to demonstrate quantitatively the safety features of new reactor designs.[2]

## 1.1. Federal Regulations

*10 CFR Part 50*: Domestic Licensing of Production and Utilization Facilities - Section 50.34, "Contents of Applications; Technical Information," requires that each applicant for a construction permit or operating license provides an analysis and evaluation of the design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the *risk to public health and safety* resulting from operation of the facility.

*10 CFR Part 100*: Reactor Site Criteria - Section 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release

[2] "Nuclear Power Reactor Source Term", https://www.nrc.gov/reactors/new-reactors/advanced/nuclear-power-reactor-source-term.html, 2023.

assumed in these evaluations should be based upon a major accident involving *substantial meltdown of the core* with subsequent release of appreciable quantities of fission products.

*10 CFR Part 51*: Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions - This regulation establishes the environmental protection requirements for nuclear power plants, including radiological considerations.

*10 CFR Part 52*: Licenses, Certifications, and Approvals for Nuclear Power Plants - This regulation establishes the requirements for obtaining licenses, certifications, and approvals for nuclear power plants, including radiological evaluations and source terms. 10 CFR Part 52.47 Section (a) (iv) states that the safety analysis report must include information of the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.

## 1.2. Regulatory Requirements

### 1.2.1. TID-14844

In 1962, the Atomic Energy Commission (AEC) published a technical information document (TID) TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites.” It requires an assumption of “maximum credible accident” resulting in release of 100% noble gasses, 50% halogens and 1% solids in the fission product inventory of the reactor core to the reactor building. This is roughly 15% of all activity present in the core.

### 1.2.2. SECY-93-092

In the NRC staff requirements memorandum (SRM) to SECY-93-092 in July 1993, the NRC approved that source terms for non-LWRs are based upon a mechanistic analysis and that the acceptability of an applicant's analysis will rely on the NRC staff's assurance that the following conditions are met:

- The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis.
- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including the specific consideration of containment design.
- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.
- The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

SECY-93-092 describes a mechanistic source term as the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences. It is calculated by best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, and into the environs.

### 1.2.3. NUREG-1465

In 1995, the NRC published NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants.” It provided a guideline for realistic assessment of source term including uncertainties instead of the deterministic bounding value in TID-14844, as well as the credit to engineered safety features (ESFs) for fission product removal. The calculation of mechanistic source terms can be conducted by Source Term Code Package (STCP) as well as an advanced severe accident system code like MELCOR.

### 1.2.4. Regulatory Guide 1.183

In 2000, the NRC published Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents (DBAs) at Nuclear Power Reactors.” RG 1.183 provides assumptions and methods that are acceptable to the NRC for performing design basis radiological analyses using an Alternative Source Term (AST).

RG 1.183 expects an acceptable AST to have the following attributes:

- The AST must be based on <u>major accidents</u>, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.
- The AST must be expressed in terms of <u>times and rates of appearance of radioactive fission products</u> released into containment, the <u>types and quantities</u> of the radioactive species released, and the <u>chemical forms of iodine</u> released.
- The AST must not be based upon a single accident scenario but instead must represent a <u>spectrum of credible severe accident events</u>. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered.
- The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.
- The AST must be peer-reviewed by appropriately qualified subject matter experts (SMEs). The peer-review comments and their resolution should be part of the documentation supporting the AST.

Note that the release fractions of RG 1.183 are similar to those of NUREG-1465 except that only the gap release and early in-vessel phases are considered. This is due to the design basis source term criteria in 10 CFR 50.67, which considers the total effective dose only for the first two hours of the accident since that is the worst. The release fractions match those of NUREG-1465, but the onset of each phase is explicitly defined in RG 1.183.

## 2. DESIGN FEATURES OF FMR

The FMR is a gas-cooled fast reactor (GFR), operating at system temperature range of 509 °C to 800 °C. It is a grid-capable power source with a gross electric output of 44 MW. The reactor core uses helium coolant and uranium dioxide ($UO_2$) fuel pellets encapsulated in a silicon carbide (SiC) composite cladding, arranged in a triangular pitch and forming a hexagonal fuel assembly. The reactor core is an annular shape surrounded by solid reflector blocks of zirconium silicide ($Zr_3Si_2$)[3] and graphite that preserve neutrons. $Zr_3Si_2$ is a heavy reflector specifically developed for the GFR.

Helium is chemically inert and will not cause any chemical or nuclear reaction. The use of conventional $UO_2$ as the fuel in combination with accident tolerant $SiC_f$/SiC composite cladding offers advanced safety characteristics such as high temperature operation and passively safe core design that minimize the likelihood of accidents. The helium coolant is intrinsically safe as it does not change its material phase, does not react with other materials or burn in air. The major systems and components are deployed underground and protected from the external impact as illustrated in Figure 1.

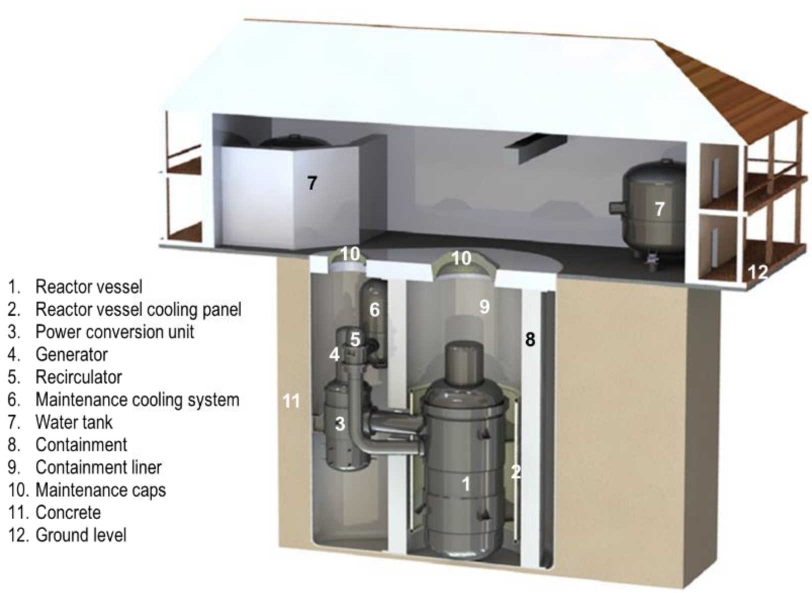


**Figure 1. FMR nuclear island components**

[3] G. M. Jacobsen, et al., "Fabrication and Characterization of Zirconium Silicide for Application to Gas-Cooled Fast Reactors," *Nuclear Technology* **208**, 27-36, 2022.

The Power Conversion System (PCS) consists of a turbine, compressors, heat exchangers, and a generator, that converts the thermal energy generated by the reactor into electricity. The turbine-compressor-generator (TCG) are mounted in vertical configuration. The generator is in a separate, connected vessel at the top of the turbomachine. The magnetic coupling of the turbine and generator isolates the helium in the generator from the primary coolant.

One of the advanced design features of the FMR is its ability to passively remove decay heat from the core and vessel, regardless of whether helium is present. This is achieved through the implementation of a gravity-driven reactor vessel cooling system (RVCS). RVCS is always in operation and continues passively removing the heat from the reactor vessel by natural circulation of water circulating in the RVCS loop. Unlike traditional gas-cooled reactors, which are typically packed with solid graphite, the FMR does not rely on conduction-cooldown. Instead, the passive safety of the core is primarily enhanced by the radiation heat transfer mechanism.

For a rodded core like the FMR, the radiation heat transfer is the dominant heat transfer mechanism from the fuel rods to the surrounding solid structures, rather than conduction or convection. Other design features, such as the large thermal margin, low power density, and annular core configuration, further enhance the passive safety of the core.

Heat from the reactor vessel is transferred to the cooling panel of the RVCS through radiation. This system ensures that any decay heat generated by the core can be safely and efficiently removed, without the need for active cooling systems or other complex mechanisms. As a result, the FMR is able to offer exceptional levels of safety and reliability, making it an attractive option and a significant advancement in nuclear power generation technology.

### 2.1. Reactor System

The reactor system is inside the RPV which is typically divided into upper and lower parts connected by a welded flange as illustrated in Figure 2. The lower part contains the fuel assemblies, reflectors, core support, core barrel, flow path to the power conversion unit (PCU), flow path to the maintenance cooling system (MCS), and a lower plenum. The upper part contains an upper plenum, thermal shield, control rod guides, and control rod drive mechanism. The outer diameter (OD) and height of the RPV are [[redacted]] and [[redacted]], respectively.

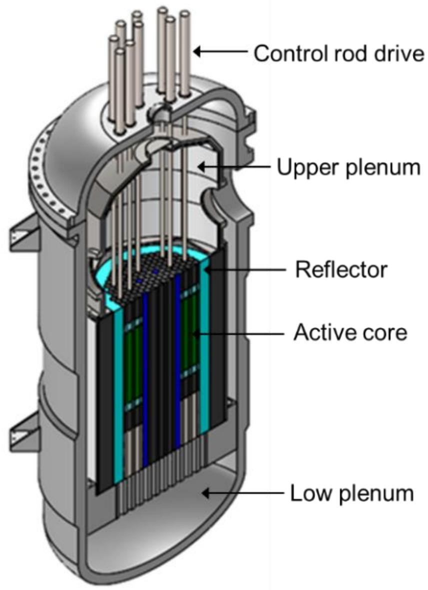


**Figure 2. Schematic reactor system layout**

### 2.1.1. Fuel Rod and Assembly

The FMR fuel is made of $UO_2$. The fuel is in the form of a sintered pellet with a theoretical density (TD) of 95%. The fissile ($^{235}U$) content of the fuel is 19.75%, categorized as High-Assay Low-Enriched Uranium (HALEU). The fuel rod was designed to have a large fuel gap of 100-200 μm and a large gas plenum to accommodate fuel swelling and fission gas accumulation, respectively, from the high-burnup operation. The fuel rods are fabricated from cylindrical tubes made of $SiC_f$/SiC composite.[4] The fuel rods are arranged in a triangular pitch, forming a hexagonal fuel assembly. Each fuel assembly contains 120 fuel rods and a central support tube for handling as illustrated in Figure 3.

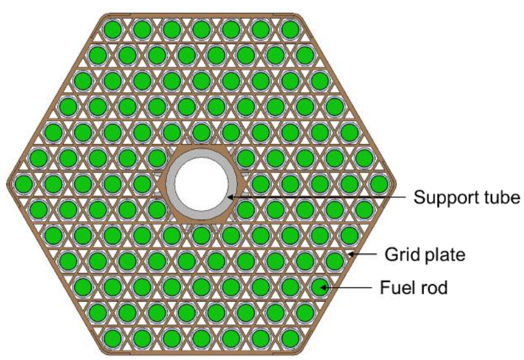


**Figure 3. Plan view of a fuel assembly**

[4] C. P. Deck et al., "Overview of General Atomics SiGA™ SiC-SiC Composite Development for Accident Tolerant Fuel," *Tran. Am. Nucl. Soc*. **120**, 371 (2019).

### 2.1.2. Reactor Core Configuration

The active core is configured in an annular shape with an OD of [[redacted]] and active height of [[redacted]]. Figure 4 shows the fuel loading pattern, consisting of three zones in radial direction with equal volume. For the fueling scheme, the fresh fuels are loaded in the middle zone 2, once burned fuels in the inner zone 1, and twice burned in the outer zone 3. This loading pattern provides acceptable neutronics and thermal performance in terms of the cycle length and total peaking factor. The reactor is anticipated to operate ~16 years between refueling, accumulating a cycle burnup of ~33 GWd/t. Therefore, the fuel residence time is ~48 years and the average discharge burnup is ~100 GWd/t.

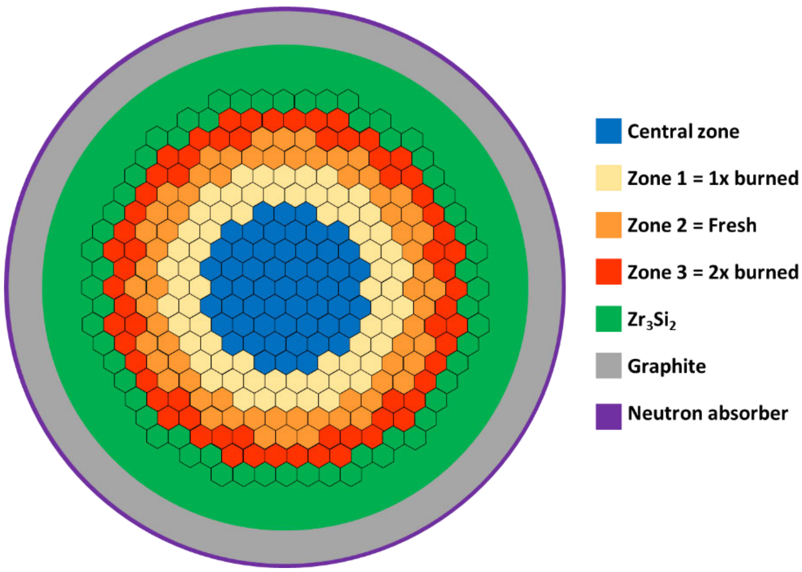


**Figure 4. Core configuration and fuel loading pattern**

### 2.1.3. Reflectors

The reactor core consisting of hexagonal fuel assemblies is surrounded on all sides by reflector blocks to minimize neutron leakage by returning neutrons back to the reactor core. The use of reflector materials, having a high neutron scattering cross-section and low absorption cross-section such as $Zr_3Si_2$ and graphite, increases the neutronic efficiency. Another aspect of using solid reflectors is to provide a heat flow path from the reactor core to the RPV to passively cool down the core during the emergency core cooling condition.

Above the active core, both the $Zr_3Si_2$ and graphite top reflectors in a hexagonal block form are deployed. The axial thickness of these reflector blocks is approximately the same as radial thickness of the radial reflector. These top reflector blocks have multiple holes for the control rod guide tube and coolant flow. Below the active core, primary reflector pellets are deployed inside the fuel rod with similar axial dimensions to the top reflector blocks.

## 2.2. Power Conversion System

Direct Brayton cycle provides the fast grid-responding capability and high thermal efficiency during normal operation. The high-speed generator, along with a permanent magnet rotor and the helium turbo-compressor, has a small rotational inertia. The high-efficiency, solid-state converter accepts a wide range of input frequencies and voltages and outputs the line frequency.

For the turbine, the mass flow and cooling flow at the first stage are 67.4 kg/s and 0.7 kg/s, respectively. The temperature and pressure are 800ºC and 7 MPa at the inlet and 539ºC and 3.33 MPa at the outlet of the six-stage turbine, respectively, with inlet Mach number of [[redacted]] and [[redacted]] reaction. The maximum tip speed of turbine blade is [[redacted]] at the last stage. The design point turbine isentropic efficiency is 93.6%, or 92.13 MW turbine output, with a windage loss of 1 MW.

The compressor aerodynamic design utilized [[redacted]] reaction with the same hub diameter for both high-pressure (HP) and low-pressure (LP) modules. The compressor design was conducted with a constant hub-tip ratio of [[redacted]]. The inlet Mach number for HP and LP compressor modules are [[redacted]] and [[redacted]], respectively, at the best design point efficiency. The number of stages is 7 for both HP and LP, resulting in aerodynamic part length of [[redacted]] and [[redacted]] for HP and LP module, respectively. The maximum tip speed is [[redacted]] and [[redacted]] for HP and LP module, respectively.

Figure 5 shows cross section of turbine and compressor modules with diaphragm couplings along with helium flow. Both turbine and compressors were designed with independent set of radial and axial active magnetic bearings (AMBs). Entire turbomachinery is connected to the permanent magnet generator using a diaphragm coupling. Both turbine and compressors were designed to have the first bending mode greater than 14,000 revolutions per minute (rpm).

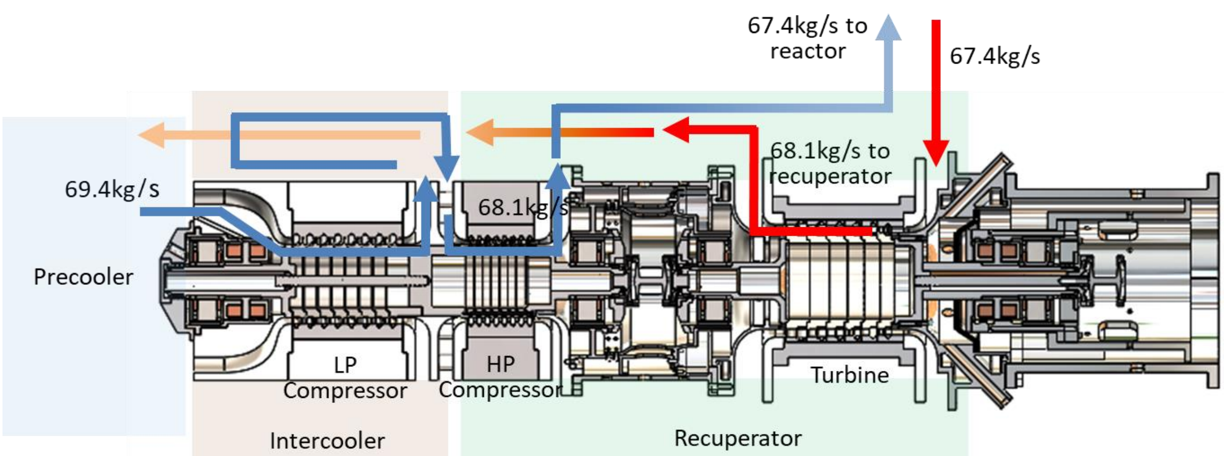


**Figure 5. Turbomachinery layout**

### 2.3. Containment

The Containment System (CNS) is the collection of boundaries that separates the containment atmosphere from the outside environment. The CNS is designed to contain the release of radioactivity following postulated accident conditions. The CNS will be a pressurized environment with no built-in mechanism for depressurization after high energy line breaks inside containment. Included in the CNS is the reactor building which will be constructed over the below grade CNS.

The primary functions of the CNS are as follows:

- Contains the release of radioactivity following postulated accident.
- Provides containment isolation feature so that fluid lines penetrating the primary containment boundary are isolated in the event of an accident.
- Provides periodic leak rate testing feature for the containment, containment penetrations and isolation barriers.

The CNS is a high-strength, low-leakage, pressure-retaining structure surrounding the reactor and its primary cooling system to control the release of radioactivity to the environment. The operating environment (or requirements) of the CNS are as follows:

- The CNS shall be designed to internal temperatures from 25°C to 45°C.
- The CNS shall be designed to internal pressures of up to .7 MPa.
- The CNS shall be designed to the total neutron fluence of reactor operating at 100 MWth power, 100% duty cycle for the design life duration.

## 3. SOURCE TERM CALCULATION APPROACHES

NRC proposed a calculational framework for evaluation of HTGRs with a network of computer programs/codes, models, and data.[5] The calculation procedure of the FMR source term is the same as this framework except that the MCNP6.2[6] and ORIGEN2.2[7] are used for reactor physics and fission product inventory calculations, respectively, as pictured in Figure 6. MELCOR[8] is the main computational tool that calculates the fission products transport inside the reactor pressure

[5] "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," 2020.
[6] C. J. Werner, Ed., "MCNP User's Manual – Code Version 6.2," LA-UR-17-29981, Los Alamos National Laboratory, 2017.
[7] "ORIGEN 2.2 Isotope Generation and Depletion Code," CCC-371/ORIGEN 2.2, Oak Ridge National Laboratory, 2002.
[8] L. L. Humphries, "MELCOR Computer Code Manuals," SAND2021-0241 O, Sandia National Laboratories, 2021.

boundary and the containment, followed by the MACCS[9] simulation of the radiation source dispersion from the containment into the environment.

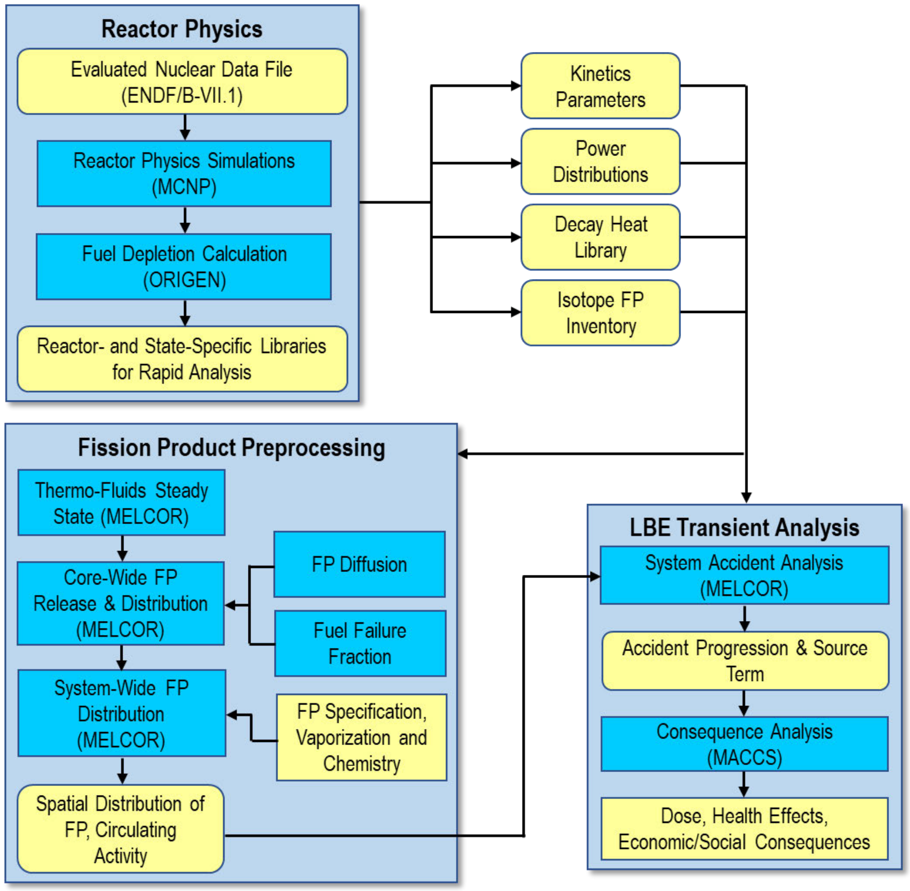


**Figure 6. FMR evaluation model modified from NRC's HTGR evaluation model**

The MELCOR source term calculation involves the complex physical and chemical processes that affects fission product release, fuel-coolant interactions, core degradation, possible hydrogen generation and combustion, and containment failure. It also considers the impact of mitigating actions, such as the use of spray or venting system, on the release of radioactive material. A set of assumptions are also used for the behaviors of the fuel, fission products, reactor coolant system and containment during operation and accident.

The methodology of MELCOR source term calculation is as follows:

1) Develop MCNP model to provide MELCOR with the power distribution and kinetics parameters.

[9] K. Clavier et al., "MACCS User Guide – Version 4.2," SAND2023-01315, Sandia National Laboratories, 2023.

2) Develop ORIGEN model to provide MELCOR with the decay heat and core radionuclide inventories.
3) Build MELCOR full plant model including the Radio-Nuclide (RN) package model.
4) Select accident scenario for the source term calculation.
5) Perform MELCOR calculations for the selected scenario to estimate accident progression and radionuclide release.
6) Conduct MACCS calculation for the consequence analysis of the radiation source term.

MELCOR calculates the timing and magnitude of fission product radionuclide release from failed fuel for selected design basis accidents. For each accident, key parameters such as the onset time of the fission product release from the gap, duration in the gap and early in-vessel release, and the in-vessel release fractions for each major radionuclide group are calculated. The modeling assumptions of the source term calculation are as follows:

- A series of equipment failures results in core heat up and fuel damage,
- Activity is released from the fuel over a time period and homogenously mixes in containment atmosphere,
- Leakage from containment is direct to the environment, with no removal mechanisms in reactor building (scrubbing, partitioning, deposition, etc.).

### 3.1. Fission Products Barriers

There are multiple barriers incorporated by the FMR design that prevent fission product release as illustrated in Figure 7 and described below:

– *1st Barrier* is the most important system consisting of the fuel pellet and cladding. This is a robust physical barrier capable of withstanding the high temperatures and pressures expected during accident conditions.

– *2nd Barrier* is the vessel system that forms a barrier between the reactor coolant and the surrounding system of the containment building. The reactor vessel, PCU vessel, cross vessels and associated subcomponents act to retain the helium coolant and any entrained radioactive materials. This barrier is designed to withstand high pressures and temperatures during the normal operation and accident conditions.

– *3rd Barrier* is the containment that is the ultimate physical boundary designed to provide additional safety margin and defense-in-depth. The containment building prevents the release of fission products into the environment, even in the event of a severe accident with failure of the 1st and 2nd barriers.

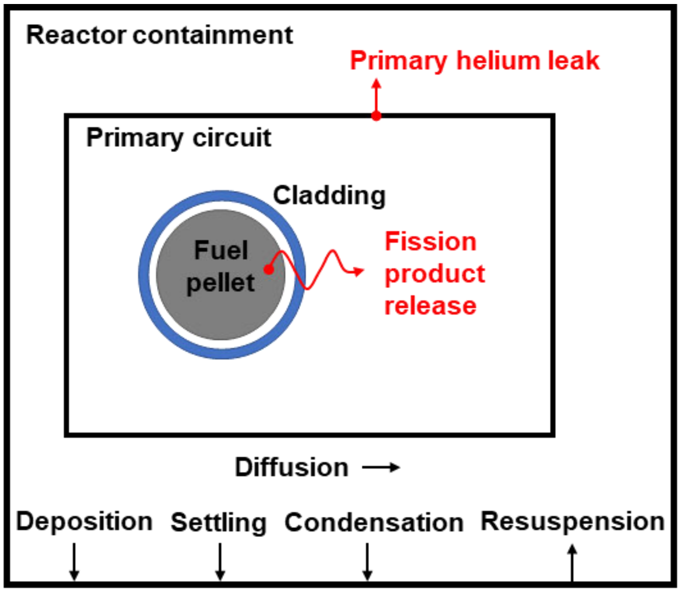


**Figure 7. FMR radionuclide retention system**

The first barrier is the most crucial protection against the release of fission products. The second barrier is designed to complement and supplement the fuel barrier. The third barrier is designed as an independent barrier to aid in meeting the plant design requirements, providing additional margins, and providing defense-in-depth. The overall source term will depend on multiple parameters including the fuel design and manufacturing, fuel performance during operation, and accident conditions.

### 3.2. Fission Product Release Mechanism

Most of the fission products which are formed in the fuel remain within the fuel pellet. Some of those fission products, however, become mobile during normal operation and accident events. Their migration from the fuel to the reactor coolant is slowed and inhibited by the first barrier, namely the fuel pellet and cladding. After being released from the fuel rod, fission products are mixed with the circulating helium coolant. Most of released fission products are entrained in the flowing helium and circulating during normal operation. During the accident accompanying with the breach of primary pressure boundary, these circulating fission products are released to the containment.

The fission product release and source term are determined by several factors that interact from each other. These factors are listed as follows:

- The inventory of fission products and other radionuclides in the core, which are affected by the amount of fuel loading in the reactor, fuel burnup and irradiation history. This inventory can be calculated by analytical methods.

- The fraction of radionuclides released from the fuel, and the physical and chemical forms of released radioactive materials, with some radionuclides being more volatile and easier to be released, while others may be retained in the fuel and primary coolant system.
- The progression of core damage, which is characterized by the causes of the accidents, the severity and extent of core damage, as well as the duration of the accident.
- The retention of radionuclides by the primary coolant system, which depends on the design features of the vessel system including leak rate, plateout, and liftoff.
- The performance of containment, that is to prevent the release of radioactive material to the environment.

The behavior of radionuclides during normal operation is a complex process that depends on several factors, including fuel quality, coolant chemistry, and reactor design. After some period of normal operation, steady-state distributions of most of the species of radionuclides generated by fission will be established in the primary circuit.[10]

The majority of the xenon and krypton formed in the fuel pin will be released from the fuel pin during a cladding breach. It can be assumed that any xenon and krypton released into the primary circuit will reach the containment during design basis events with potential radiological consequences. Once in containment, the noble gases will mix with the existing gas in the containment and escape to the environment according to the containment design basis leak rate.

### 3.2.1. Fission Gas Release from $UO_2$

The fission gases, e.g., xenon and krypton, are generated from both the direct fission of uranium and through the decay of other fission products. During operation, fission gas bubbles are created, causing the fuel to swell as the burnup level increases. Small bubbles of noble gases form, agglomerate and migrate through interconnected porosity in the fuel matrix. The fission gas bubbles travel through these new pathways and eventually reach to the fission gas plenum, resulting in internal gas pressure increase.[11]

Various numerical models are available in the fuel performance and system performance analysis codes to calculate the fission gas release. As an example, a fission gas release model is implemented in BISON code SIFGRS model which originates from Pastore et al.[12] It is a physics-based model that governs a series of gas behaviors in fuels, including intra-granular gas migration, intra- and inter- granular bubble growth and coalescence, and fission gas release.

[10] W. Moe et al., “Mechanistic Source Terms White Paper”, INL/EXT-10-17997, Idaho National Laboratory, July 2010

[11] R. J. White, M. O. Tucker, “A New Fission-Gas Release Model,” *J. Nucl. Mater.*, **118**, 1 (1983)

[12] G. Pastore et al., “Physics-based modelling of fission gas swelling and release in $UO_2$ applied to integral fuel rod analysis,” *Nuclear Engineering and Design* **256**, 75-86, 2013.

Another example is FASTGRASS that is a rate theory fuel performance code with a focus on fission gas behavior.[13] The rate theory model implemented in FASTGRASS is generic and can be used for any type of solid-state fuel materials after appropriate parameterization. The model has mainly been parameterized for $UO_2$ in LWRs. The FASTGRASS model for LWR $UO_2$ has been verified and validated through a series of LWR experiments.

In the FASTGRASS, as an example, the equation for transport of a gas to the grain boundaries in the presence of intragranular precipitation, gas bubble nucleation, and irradiation-induced re-solution can be written as:

$$\frac{\partial C_g}{\partial t} = D_g \nabla^2 C_g - dh/dt + bNC_b - (dN/dt)_n + K \qquad \text{Eq. 1}$$

where $(dN/dt)_n$ is gas bubble nucleation rate, $dh/dt$ is the rate of gas precipitation into bubbles, and $bNC_b$ is the effect of irradiation-induced re-solution.

MELCOR has its own radionuclides release models in RN Package that is further discussed in Section 3.3.

### 3.2.2. Fuel Failure Criteria

There are multiple fuel failure criteria being used for the conventional nuclear fuel such as melting temperature, stress-strain, internal pressure, etc.[14] A distinct feature of the FMR fuel rod from the conventional fuel is the $SiC_f$/SiC composite cladding that has a very high temperature and radiation resistance while much less ductile when compared with the metal cladding. The failure criteria are still under development. Nonetheless, the conventional criteria can be used as described below.

*Maximum Hoop Stress and Strain*:

For multi-layer $SiC_f$/SiC composite cladding, impermeability is provided by a monolithic SiC (mSiC) layer, while the structural integrity is maintained by the composite layer. If the composite layer exceeds its Ultimate Tensile Stress (UTS), it is considered failed. Stone et al.[15] constructed a stress-strain curve shown in Figure 8 through a statistical approach using measured data from open literature[16] and measurements of tubular materials.[17] The best estimate of hoop strain limit

[13] J. Rest and S. Zawadzki, "FASTGRASS: A mechanistic model for the prediction of Xe, I, Cs, Te, Ba, and Sr release from nuclear fuel under normal and severe-accident conditions," NUREG/CR-5840, ANL-92/3, Argonne National Laboratory, 1992.

[14] "Standard Review Plan Section 4.2: Fuel System Design," NUREG-0800, U.S. Nuclear Regulatory Commission, 2007.

[15] J. G. Stone et al., "Stress analysis and probabilistic assessment of multi-layer SiC-based accident tolerant nuclear fuel cladding," *J. Nuclear Materials* **466**, 682–697, 2015.

[16] Y. Katoh et al., "Continuous SiC fiber, CVI SiC matrix composites for nuclear applications: Properties and irradiation effects," *J. Nuclear Materials* **448**, 448–476, 2014.

[17] G. M. Jacobsen et al., "Investigation of the C-ring test for measuring hoop tensile strength of nuclear grade ceramic composites," *J. Nuclear Materials* **452**, 125–132, 2014.

was determined to be 0.62% based on the composite layer UTS,[18] beyond which the impermeability of the $SiC_f/SiC$ composite would be compromised.

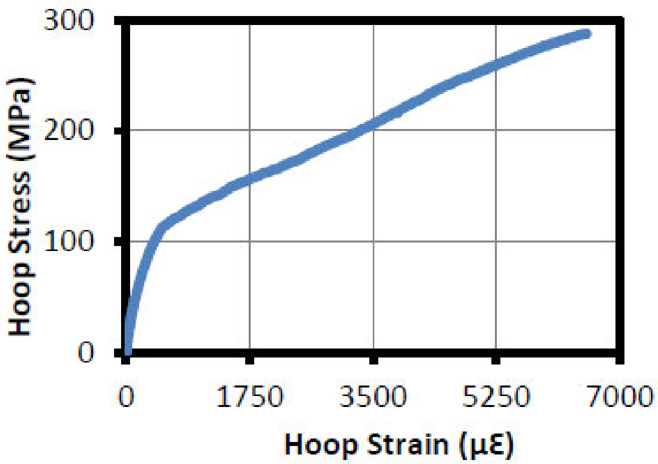


**Figure 8. Typical hoop stress-strain plot for GA-EMS cladding**

The UTS for composite material varies greatly with architecture/manufacturer. Typical values range from 230-270 MPa for axial UTS and 200-340 MPa for hoop UTS. Weibull moduli of 7-12 are typically observed. A more thorough investigation of typical statistical variation can be found in the literature.[19,20,21] GA-EMS determined the stress allowable for composite materials per American Society of Mechanical Engineers (ASME) Section III Division 5. Specifically, the material allowable is determined for $SiC_f/SiC$ composite material in following steps:

1) Determine the stress classification and probability of failure of the component,
2) Obtain Weibull parameters (from material data if necessary),
3) Determine correction factors to be applied to Weibull parameters, and
4) Determine $S_{gm}$ equation and obtain $S_{gm}$ for desired stress classification and probability of failure (POF).

[18] G. Jacobsen, H. Chiger, "General Atomics Silicon Carbide Cladding," GA-A28712 Revision 1, General Atomics, 2017.
[19] T. Koyanagi, Y. Katoh, G. Singh, M. Snead, "SiC/SiC Cladding Materials Properties Handbook," ORNL/TM-2017/385, Oak Ridge National Laboratory, 2017.
[20] G. M. Jacobsen, K. Shapovalov, E. Song, C. P. Deck, "Mechanical Behavior of Nuclear Grade SiC-SiC Tubing at Operating and Accident Temperatures," *12th Pacific Rim Conference on Ceramic and Glass Technology*, Waikoloa, HI, May 21-26, 2017.
[21] G. Singh, S. Gonczy, C. Deck, E. Lara-Curzio, Y. Katoh, "Interlaboratory round robin study on axial tensile properties of SiC-SiC CMC tubular test specimens," *Int J Appl Ceram Technol*. **15**, 1334– 1349, 2018.

Table 1 lists the calculated allowables for $SiC_f/SiC$ composite by using UTS as the characteristic strength, $S_c$.[22] The estimated stress allowable is approximately 1/3 UTS for POF of $10^{-4}$.

**Table 1. Allowable stress values for SiC based on UTS**

| Direction | Young's modulus (GPa) | UTS (MPa) | Standard deviation | Weibull modulus | $S_g$(MPa) | | |
|---|---|---|---|---|---|---|---|
| | | | | | POF $10^{-4}$ | POF $10^{-3}$ | POF $10^{-2}$ |
| Hoop | 228 | 359 | 23 | 13.6 | 117 | 154 | 201 |
| Axial | 222 | 288 | 15 | 12.1 | 82 | 111 | 150 |

However, the hermeticity of the $SiC_f/SiC$ cladding is determined by the proportional limit stress (PLS) of the SiC composite layer. The PLS in the axial direction have been measured in the 80-100 MPa range and for the hoop direction 100-160 MPa. These properties are dependent on fiber architecture and manufacturer. It was observed that the effects of irradiation and temperature on the PLS are similar to those of UTS. Table 2 lists the calculated allowables for $SiC_f/SiC$ composite by using PLS, beyond which the impermeability of the SiCf/SiC composite would be compromised. Note that the number of data used in this calculation is less than that used for UTS-based calculation.

**Table 2. Allowable stress values for SiC based on PLS**

| Direction | Young's modulus (GPa) | PLS (MPa) | Standard deviation | Weibull modulus | $S_g$(MPa) | | |
|---|---|---|---|---|---|---|---|
| | | | | | POF $10^{-4}$ | POF $10^{-3}$ | POF $10^{-2}$ |
| Hoop | 207 | 153 | 9 | 13.6 | 49.9 | 65.4 | 85.8 |
| Axial | 171 | 98 | 8 | 12.1 | 27.8 | 37.7 | 51.2 |

*Maximum Pressure Difference*:

The pressure difference is the difference between the pressure inside and outside the fuel rod. The maximum pressure difference criterion for fuel rods is based on the differential pressure that the fuel cladding can withstand before failing. This criterion is critical in preventing fuel rod failure during a loss of coolant accident (LOCA), where the coolant flow to the reactor core is interrupted. During a LOCA, the fuel rods experience a sudden drop in coolant pressure, which can cause the cladding to rupture. The maximum pressure difference criterion for fuel rods is set to 25.6 MPa

[22] E. Chin, "Silicon Carbide Composite Stress Allowable Per American Society of Mechanical Engineers Section III Division 5," 30599300R0028, General Atomics, 2023.

based on joint burst strength.[23] If the pressure difference exceeds this limit, the cladding may rupture, causing fuel failure.

*Maximum Temperature*:

The maximum temperature criterion for fuel rods is typically based on the melting point of the fuel and cladding materials. The fuel and cladding materials must be able to withstand the high temperatures inside a nuclear reactor without melting or degrading. The melting temperature of unirradiated $UO_2$ is 2840°C determined experimentally by Brassfield et al.[24]

The SiC does not melt but begins to sublimate near 2700°C. However, $SiC_f$/SiC composite cladding begins degradation of its mechanical strength near 1800°C.[25] The degradation of mechanical strength doesn't mean that the $SiC_f$/SiC composite cladding collapses at temperatures above 1800ºC and releases fission gases. However, in order to build and verify the source term calculation method, it was assumed that the fuel will release all fission gases at 1800ºC in this study.

### 3.2.3. Core Damage Progression

The amount of fission products effectively counted as the source term depends on the progression of fuel and core damage that is affected by manufacturing faults, accident management/mitigation activities, cleanup activities, etc. As an example, if an emergency ventilation system is available, it will reduce the source term. It also has a significant impact on the residence time of the air in the containment and therefore has an impact on the radionuclides inventory deposited on the surfaces inside the containment. The emergency ventilation rate may also affect the retention factor of charcoal filters for iodine removal for accidents with high humidity in the exhaust air.

Regarding the impact on the environment and the radiological consequences, the release point from the containment, the release mode (single puff, intermittent or continuous release) and the energy content of the release are also very important. As the FMR design specifications mature for the fuel, primary heat transport system, and containment, these additional factors will be implemented in the source term calculation.

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[23] H. E. Khalifa, C. P. Deck, O. Gutierrez, G. M. Jacobsen, C. A. Back, "Fabrication and characterization of joined silicon carbide cylindrical components for nuclear applications," *J. of Nuclear Materials* **457**, 227-240, 2015.
[24] H. C. Brassfield, J. F. White, L. Sjodahl, J. T. Bittel, "Recommended Property and Reactor Kinetics Data for Use in Evaluating a Light-Water-Coolant Reactor Loss-of-Coolant Incident Involving Zircaloy-4 or 304-SS-Clad $UO_2$," Report GEMP-482, General Electric Company, 1968.
[25] K. Shapovalov et al., "C-ring Testing of Nuclear Grade Silicon Carbide Composites at Temperatures up to 1900ºC," *J. Nuclear Materials* 522, 184–191, 2019.

### 3.3. MELCOR Methodology for Radionuclide Release

The approach of MELCOR source term evaluation within the NRC framework is illustrated in Figure 9. The process of estimating the source term is similar for both LWRs and non-LWRs, using the same computer code systems with the primary difference being the approach that is used to analyze various accident events. For the FMR, the analytical evaluations process starts with the development of radionuclide inventories using the ORIGEN code. The next step is to perform severe accident analysis with MELCOR using these inventories. MELCOR computes radionuclide release to the site and the environment as a radionuclide source term.

The consequence of the radionuclide source term is then assessed using software tools such as MACCS, RADTRAD[26] and RASCAL.[27]

- The MACCS is primarily used for off-site consequence analysis such as dose to the public. It accounts for atmospheric transport, dispersion, and deposition of radionuclides, allowing for the assessment of both health and economic consequences.
- The RADTRAD is used for on-site calculations, such as control room dose calculations. It uses a combination of realistic system behavior and atmospheric dispersion characteristics to model radionuclides as they move from the primary containment to elsewhere on-site.
- The RASCAL software is a response tool used to make recommendations regarding emergency response decisions.

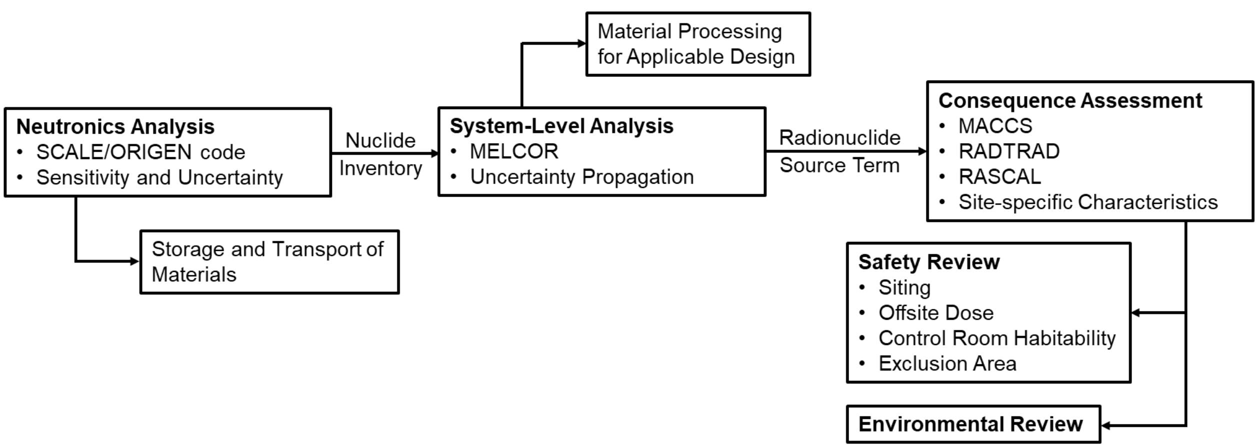


**Figure 9. MELCOR source term calculation approach within the NRC framework**

[26] W. C. Arcieri, D. L. Mlynarczyk, L. Larsen, "SNAP/RADTRAD 4.0: Description of Models and Methods," NUREG/CR-7220, U.S. Nuclear Regulatory Commission, 2016.
[27] J. V. Ramsdell, Jr., G. F. Athey, J. P. Rishel, "RASCAL 4.3: Description of Models and Methods," NUREG-1940, Supplement 1, U.S. Nuclear Regulatory Commission, 2015.

### 3.3.1. Fuel Gap Fission Products Release

Source term calculations start with determining the fission product inventory. The fission product inventory can be categorized based on their locations such as the fuel pellet, fuel gap/plenum, and reactor coolant system. ORIGEN provides total fission product inventory of the FMR fuel as a function of fuel burnup to the MELCOR simulations.

The fission product inventory in the fuel plenum is determined by a fuel-to-gap release model. Various mechanisms affect the release of fission products from the fuel pellet to the gap as discussed in Section 3.2.1. Several options are currently available for the release of radionuclides from the core components; the CORSOR, CORSOR-M, CORSOR-BOOTH, or modified ORNL-BOOTH model may be specified on input record RN1_FP00. The CORSOR-BOOTH and generalized release models are summarized as follows:

*CORSOR-Booth Model*

The CORSOR-Booth model considers mass transport limitations to radionuclide releases and uses the Booth model[28] for diffusion with empirical diffusion coefficients for cesium releases. Release fractions for other classes of radionuclides are calculated relative to that for cesium. The classical or effective diffusion coefficient for cesium in the fuel matrix is given by:

$$D = D_0 exp(-Q/\,RT) \qquad \text{Eq. 2}$$

where *R* is the universal gas constant, *T* is the temperature, *Q* is the activation energy, and the pre-exponential factor $D_0$ is a function of the fuel burnup. For the fuel burnup greater than 30 GWd/MTU, the model increases the $D_0$ by a factor of five.

The cesium release fraction (*f*) at time *t* is calculated from an approximate solution of Fick's law for fuel grains of spherical geometry,

$$f = 6\sqrt{\frac{D't}{\pi}} - 3D't \quad for\ D't < 1/\pi^2 \qquad \text{Eq. 3}$$

$$f = 1 - \frac{6}{\pi^2}\exp(-\pi^2 D't) \quad for\ D't > 1/\pi^2 \qquad \text{Eq. 4}$$

where $D't = Dt/a^2$ (dimensionless) and *a* = equivalent sphere radius for the fuel grain.

*Generalized Release Model*

The generalized release model is an alternate release model that can be easily customized by the user to allow both diffusion and burst component. The cumulative burst fission product release fraction is described by the following equation:

[28] A. H. Booth, "A Method of Calculating Fission Gas Diffusion from $UO_2$ Fuel and Its Application to the X-2 Loop Test," AECL-496, CRDC-721, Atomic Energy of Canada, Ltd., 1957.

$$FB_{j,i} = a\_burst_j(c_0 + c_1 * T_i + c_2 * T_i^2 + c_3 * T_i^3) \quad \text{Eq. 5}$$

where $T_i$ = fuel temperature during the time step *i*, $c_n$ = user-provided coefficients, and $a\_burst_j$ = user-provided class *j* coefficient.

A cumulative diffusive fission product release fraction is described by the following equation:

$$FD_{j,i} = b\_diff_j(FD_{j,i-1} + (1 - FB_{j,i-1} - FD_{j,i-1}) * [1 - e^{-kd_{j,i} * \Delta t_i}]) \quad \text{Eq. 6}$$

where $FD_{j,i}$ = cumulative fraction of diffusive fission product released up to time step *i*, $b\_diff_j$ = user-provided class-dependent coefficient, $FB_{j,i}$ = cumulative fraction of burst fission product released up to time step *i*, and $kd_{j,i}$ = release rate of fission product class *j* calculated at temperature $T_i$ of the time step *i*.

The total cumulative fission product release fraction at time step *i* for fission product *j* is determined by:

$$F_{j,i} = d\_total_j(FB_{j,i} + FD_{j,i}) \quad \text{Eq. 7}$$

where $d\_total_j$ = user-provided class-dependent multiplier.

### 3.3.2. Core Fission Products Release

In MELCOR simulations, the radionuclides residing in the COR package fuel are assumed to be in elemental form, with only radioactive mass (no associated molecular mass). When released from fuel, the total class masses are converted to compound form, resulting in an increase in mass from the added nonradioactive material.

The core can release both radioactive and nonradioactive materials. To calculate the release of radioactive materials from the core, MELCOR uses release models and aerosol transport models that are specific to the different aerosol transport mechanism and phenomena. For fuel material, the default release models are based on the release of radionuclides from $UO_2$. The release models for other core components, such as fuel rod cladding, are based on empirical data and modeling studies.

The calculation of the release of radioactive materials from the core involves several steps:

- The release models are used to determine the amount and rate of release of radioactive materials from each component of the core.
- The resulting data is then used to calculate the total amount of radioactive materials released from the core over time, that is influenced by multiple factors such as the type and severity of the accident, the composition of the materials involved, and the environmental conditions.

To apply the release models to core materials other than fuel, such as the fuel rod cladding, the user must change the default values of the core material release multipliers contained in sensitivity coefficient array 7100. For these other core materials, the mapping scheme determines the apportioning of the core masses among the RN classes, and the entire masses are considered nonradioactive.

### 3.3.3. Fission Products Aerosol Transport

MELCOR RN package predicts the behavior of aerosols during an accident. The principal aerosol quantities of interest are the mass and composition of aerosol particles and their distribution throughout the reactor coolant system and containment. Fission products may be aerosolized as they are released from fuel early in an accident and later expelled from the reactor coolant system. Other events and processes that occur late in the accident such as core-concrete interactions, direct containment heating, deflagrations, and resuspension may also generate aerosols. High structural temperatures may also result in aerosolization of nonradioactive materials.

The calculation of aerosol agglomeration and deposition processes is based on evaluation of the size distribution of each type of aerosol mass, or component, as a function of time. This analysis is based on the mass contained within each size bin or section, with each section having its own chemical composition defined by the masses of the various components present in that section. A section represents a particular group of aerosol sizes, and a component refers to a specific type of aerosol material.

*Agglomeration*

Agglomeration modeling involves the prediction of the size distribution and settling velocities of aerosols generated during a severe accident. Agglomeration occurs when two aerosol particles collide and merge to form a larger particle. The sectional method used in MELCOR treats four agglomeration processes: Brownian diffusion, differential gravitational settling, and turbulent agglomeration by shear and inertial forces. The model assumes that simultaneous agglomeration of three or more particles is negligible.

The rate of agglomeration is influenced by a variety of factors, including the size and composition of the aerosols, the temperature and pressure of the containment atmosphere, and the concentration of other species present. MELCOR models these factors using a set of equations that describe the interactions between aerosols of different sizes and compositions. The full dependence of the agglomeration coefficients β ($m^3$/s) upon the aerosol and atmosphere properties as implemented in MELCOR is given in the following equations as presented in the MELCOR reference manual.

$$\beta = 2\pi\left(D_i + D_j\right)(\gamma_i d_i + \gamma_j d_j)/F \qquad \text{Eq. 8}$$

$$D_i = \frac{kT}{3\pi d_i \mu \chi_i} C_i \qquad \text{Eq. 9}$$

$$C_i = 1 + Kn_i[c_m + 0.4 \exp\left(-1.1/Kn_i\right)] \qquad \text{Eq. 10}$$

$$F = \frac{d_i + d_j}{d_i + d_j + 2g_{ij}} + \frac{8(D_i + D_j)}{v_{ij}(d_i + d_j)c_s} \quad \text{Eq. 11}$$

$$g_{ij} = (g_i^2 + g_j^2)^{1/2} \quad \text{Eq. 12}$$

$$v_{ij} = (v_i^2 + v_j^2)^{1/2} \quad \text{Eq. 13}$$

$$g = \frac{1}{3d_i I_i}[(d_i + I_i)^3 - (d_i^2 + I_i^2)^{\frac{3}{2}}] - d_i \quad \text{Eq. 14}$$

$$I = \frac{8D_i}{\pi v_i} \quad \text{Eq. 15}$$

$$v = (\frac{8kT}{\pi m_i})^{1/2} \quad \text{Eq. 16}$$

$$Kn_i = 2\lambda/d_i \quad \text{Eq. 17}$$

$$\lambda = \frac{\mu}{\rho_g}(1.89\text{x}10^{-4}\, M_{w,gj}/T)^{1/2} \quad \text{Eq. 18}$$

$$\rho_g = 1.21\text{x}10^{-4}\, P\, M_{w,gj}/T \quad \text{Eq. 19}$$

Variables $d_i$ and $d_j$ are the diameters of the two interacting particles, with $d_i > d_j$. The collision efficiency for gravitational agglomeration is represented by $\varepsilon_g$, with a specific value calculated in the code. The magnitude of the Brownian kernel increases with increasing values of the size ratio $d_i/d_j$. Aerosol particles are not always assumed to be spherical, particularly when liquid is present. In such cases, the agglomeration and dynamic shape factors are used to adjust the effective aerosol densities.

*Deposition, Settling, and Fallout*

The MELCOR code simulates different processes for aerosol deposition onto heat structure (HS). Calculations for modeling aerosol behavior consider five key parameters obtained from the HS package, including geometric orientation, surface area in the atmosphere, surface heat flux, mass transfer coefficient, and water condensation mass flux.

The orientation of a heat structure surface is crucial for aerosol deposition, as the code only calculates deposition kernels for ceilings, floors, and walls. The default treatment for rectangular heat structures is to consider the upper surface with an inclination less than 45 degrees as a floor, and the lower surface as a ceiling. Other structures, such as vertical cylinders and spheres, are treated as walls.

MELCOR applies gravitational settling and Brownian diffusion kernels to flowthrough areas in addition to heat structure and pool surfaces. Additionally, aerosols can agglomerate and become larger than the user-specified maximum diameter. These large aerosols are assumed to immediately deposit onto horizontal heat structure surfaces, or to settle from one control volume to another through flowthrough areas defined as part of Radionuclide (RN) input.

The term "fallout" is used exclusively for this immediate deposition or settling of large aerosols. All control volumes must have at least one upward-facing deposition surface or flowthrough area defined to receive fallout aerosols generated by this mechanism.

The code considers the condensation of aerosols on cooler surfaces within the containment building, which can remove radioactive materials from the gas phase and deposit them on surfaces. Gravitational deposition is effective only for upward-facing surfaces (i.e., floors) and flowthroughs to lower control volumes. For downward-facing surfaces (i.e., ceilings), this mechanism works to oppose other deposition processes. The gravitational deposition velocity is given by:

$$v_{grav} = \frac{d_p^2 \rho_p g C_m}{18 \mu \chi} \quad \text{Eq. 20}$$

$$C_m = 1 + \frac{2\lambda}{d_p}\left[F_{slip} + 0.4\exp\left(-1.1 d_p/2\lambda\right)\right] \quad \text{Eq. 21}$$

where $v_{grav}$ = the downward terminal velocity (m/s), $d_p$ = the particle diameter (m), $\rho_p$ = the particle density (kg/m$^3$), $g$ = acceleration of gravity (9.8 m/s$^2$), $C_m$ = the particle mobility or Cunningham slip correction factor (that reduces Stokes drag force to account for non-continuum effects), $\mu$ = viscosity of air at 298 K [~1.8x10$^{-5}$ (N-s/m$^2$)], $\chi$ = dynamic shape facto, $\lambda$ = mean free path of air at 298 K [~0.069x10$^{-6}$(m)], and $F_{slip}$ = slip factor specified on Input Record RN1=MS00 (default = 1.257).

This model assumes that the aerosol particle Reynolds number, based on particle diameter and net deposition velocity, is much less than 1, which physically means that the inertial effects of the flow may be neglected.

Deposition can also result from diffusion of aerosols in a concentration gradient from a higher to a lower concentration region. The diffusive deposition velocity is given by:

$$v_{diff} = \frac{\sigma T C_m}{3\pi \mu \chi \mathrm{d}_p \Delta} \quad \text{Eq. 22}$$

where $v_{diff}$ = diffusion deposition velocity (m/s), $\sigma$ = Boltzmann constant [1.38x10$^{-23}$ (J/s-m$^2$K$^4$)], $T$ = atmosphere temperature (K), and $\Delta$ = diffusion boundary layer thickness (RN1_MS01, default value of 10$^{-5}$ m).

*Resuspension*

Radioactive materials can be released from the fuel, coolant, or structural components of the plant, and can be transported in the form of aerosols. Once these aerosols settle on surfaces in the containment, they can become a source of secondary release if they are resuspended into the air. Resuspension is the process by which particles that have settled on a surface are lifted back into the air. In a severe accident, resuspension can occur when particles that have settled on surfaces in the containment are disturbed and become airborne again.

The resuspension model works by determining whether the deposited aerosols are released from a given surface. For dry surfaces, the particles remain attached to the surface until the gas flow past the surface is sufficient to aerosolize or resuspend the deposit. Once the gas flow is strong enough, all particles larger than a certain critical diameter are resuspended. This critical diameter varies depending on the specific conditions, such as the type of surface and the properties of the aerosol particles. When aerosols are released into the air, they are subject to various forces, such as gravity, Brownian motion, and air flow.

One important factor of the resuspension model is the critical diameter, which is the size of the particles that can be resuspended. The critical diameter depends on the properties of the surface, such as its roughness and porosity, and the properties of the aerosol particles, such as their size, density, and shape.

MELCOR stores the cumulative particle size distribution of deposited aerosol by section and resuspends all particles in a section for which the lower section boundary particle diameter is above the critical diameter. This approach assumes that the resuspended aerosol does not form agglomerates and is the same particle size distribution as the aerosol that deposited on the surface.

The critical diameter depends on various factors, such as Reynolds number of the flow, the roughness of the surface, and the properties of the aerosol particles. The default MELCOR resuspension model computes the critical diameter for resuspension inside a pipe as

$$D_{crit} = \frac{4\text{x}10^{-5}}{\pi\tau_{wall}} \qquad \text{Eq. 213}$$

where $\tau_{wall}$ is the wall shear stress (N/m$^2$) and the critical diameter is in units of meters.

The wall shear stress $\tau_{wall}$ can be expressed as

$$\tau_{wall} = \frac{1}{2} f \rho U^2 \qquad \text{Eq. 24}$$

where $f$ = friction factor, $\rho$ = gas density (kg/m$^3$), and $U$ = gas velocity along the surface (m/s).

The friction factor is calculated using Blasius formula:

$$f = \frac{0.0791}{Re^{0.25}} \qquad \text{Eq. 225}$$

where $Re$ is the flow Reynold's number.

## 3.4. Software Used for Source Term Calculation

### 3.4.1. MCNP Version 6.2

MCNP is a widely used radiation transport code that utilizes Monte Carlo methods to simulate the transport of particles through matter. One of the major strengths of MCNP is its ability to model transport of a wide range of particles, including neutrons, photons, electrons, and many other elementary particles with continuous-energy cross-section data. MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori, that results in a highly detailed and flexible representation of the materials and geometries. The MCNP calculates the criticality, power distribution, reaction rates, etc. For the FMR source term calculation, the MCNP provides the power distribution, fuel temperature coefficients, coolant temperature coefficients, coolant void reactivity, and control rod worth, that are required for the transient and accident analyses.

### 3.4.2. ORIGEN 2.2

ORIGEN is a computer code used for depletion and radioactive decay calculations, providing various nuclear material characteristics such as the buildup, decay, and processing of radioactive materials. The nuclide number density changes are described by nonhomogeneous first-order ordinary differential equations which are then solved by employing the matrix exponential method.[29] The ORIGEN uses multiple nuclear data libraries for the Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR) and Liquid Metal Fast Breeder Reactor (LMFBR). Two executable files are available for the thermal and fast reactor systems. For the source term calculation, the ORIGEN provides the isotopic concentrations, thermal power, and radioactivity.

### 3.4.3. MELCOR Version 2.2 (build 18019)

MELCOR is widely used for severe accident analysis including estimation of severe accident source terms and their sensitivities and uncertainties in various applications.  It is used to model the progression of severe accidents through modeling of the major systems and their coupled interactions in the plant. Specific calculations relevant to the source term analysis are as follows:

- thermal-hydraulic response of the primary coolant system and containment,
- core heating, fuel and cladding degradation, and core material melting and relocation,
- reactor vessel heating, thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor cavity,
- core-concrete attack and aerosol generation,
- fission product release and transport, and

[29] A. G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," *Nuclear Technology* **62**, 335-352, 1983.

- impact of Engineered Safety Features (ESFs) on thermal-hydraulic and radionuclide behavior.

### 3.4.4. MACCS Version 4.2

MACCS executes a sequence of mathematical and statistical calculations to estimate radionuclides immediately after release from the containment, movement of the material as it disperses downwind of the plant, deposition of the radioactive material, and the effects of the airborne and deposited material on humans and the environment. Coupled with the MELCOR code, MACCS performs public health and environmental consequence analysis, including the near-term health effects, chronic health effects and economic consequences. Seven different exposure pathways are included in MACCS along with accounting for emergency response actions such as sheltering-in-place and evacuation.

## 4. FMR MODELING FOR SOURCE TERM CALCULATION

The MELCOR code is composed of an executive driver and a number of major modules, called packages, to model the major systems of a nuclear power plant and their generally coupled interactions. The modeling is general and flexible, making use of a "control volume" approach in describing the plant system. For source term analysis, the following packages are relevant:

- *Executive (EXEC)*: controls execution of MELGEN and MELCOR and passes information between packages,
- *Core (COR)*: models the thermal response of the reactor core and lower plenum structures,
- *Control Volume Hydrodynamics (CVH) and Flow (FL)*: model the thermal hydraulic behavior of fluids, using flow paths to transfer mass and energy between control volumes,
- *Heat Structure (HS)*: calculates heat conduction through solid structures and energy transfer at surface boundaries,
- *Control Function (CF)*: allows the user to define functions of MELCOR variables, which can be used for reactor control logic, valve movement, or pump control, or to create a new variable to add to the plot file,
- *Non-condensable Gas (NCG)*: treats gases as ideal gases,
- *Material Properties (MP)*: includes material properties used by other packages,
- *Decay Heat (DCH)*: models decay heat from fission products,
- *Tabular Function (TF)*: allows the user to create one-dimensional tables that can be used to define material properties, create a decay heat curve, provide heat transfer coefficients to the HS package, or define mass and energy sinks,

- *RadioNuclide (RN)*: calculates the release and transport behavior of fission product vapors and aerosols.

A consequence analysis methodology was developed for the released source term from the FMR given a specific accident scenario. This methodology uses the MELCOR systems code to predict the response of the FMR fuel to the accident and any fission product release to the containment, coupled with the MACCS code to predict the off-site environmental impact. Interface programs have been written to convert fission product inventory and decay heat data from the ORIGEN code to the required format of the MELCOR and MACCS. The off-site health, dose and economic impacts are calculated by MACCS at multiple distances for off-site dose (e.g., 1, 10, 50 and 1000 miles).

## 4.1. MELCOR Model of FMR

MELCOR 2.2[30] has options to define new materials using User-Defined Material (UDM) that was used to model SiC cladding, supporting structure within the core, and $Zr_3Si_2$ reflector of the FMR. The FMR plant model was built based on most up-to-date FMR design parameters.[31,32] While the FMR is a GFR, the active core configuration is geometrically similar to a typical PWR, i.e., a cylindrical rod array of $UO_2$ fuel rods with the major difference being the cladding material (SiC), reflector regions and the coolant material (helium). Thus, using PWR reactor type in MELCOR is conceptually more suitable for the FMR core cell model. The non-LWR materials are specified using the UDM function in MELCOR.

### 4.1.1. Reactor Vessel Control Volumes

The FMR reactor system is modeled as a reactor vessel connected with a coolant source and a sink. As shown in Figure 10 as initial FMR MELCOR modeling, the reactor vessel is divided into several control volumes, including the downcomer channel, the lower plenum, six radial rings each with six axial levels of control volumes representing the core, the upper plenum, and a cavity volume. The control volumes (CV) in the core and the downcomer have name structure CV1XY where X is the radial ring and Y is the axial level of the control volume.

The coolant source and sink are CV210 and CV200, respectively. The RPV is connected to the PCU at these points. By adding a source and sink in the MELCOR model, the various systems that comprise the PCU is simplified for ease of simulation. Subsequently, the PCS loop model was further developed, as shown in Figure 11, that includes more control volumes and flow paths

[30] L. L. Humphries et al., “MELCOR Computer Code Manuals Vol. 1: Primer and Users’ Guide Version 2.2.18019,” SAND2021-0252, Sandia National Laboratories, 2021.
[31] H. Choi et al., “The Fast Modular Reactor (FMR) – Development Plan of a New 50 MWe Gas- Cooled Fast Reactor” *Trans. Am. Nucl. Soc.* **124**, 454-456, 2021.
[32] H. Choi, D. Leer, M. Virgen, O. Gutierrez, J. Bolin, “Preliminary Neutronics Design and Analysis of the Fast Modular Reactor,” *Nucl. Sci. and Eng*. **197**, 1758-1768, 2023.

of the turbine, compressor, recuperator, and pre-cooler with corresponding boundary conditions to properly simulate the momentum and energy transfer.

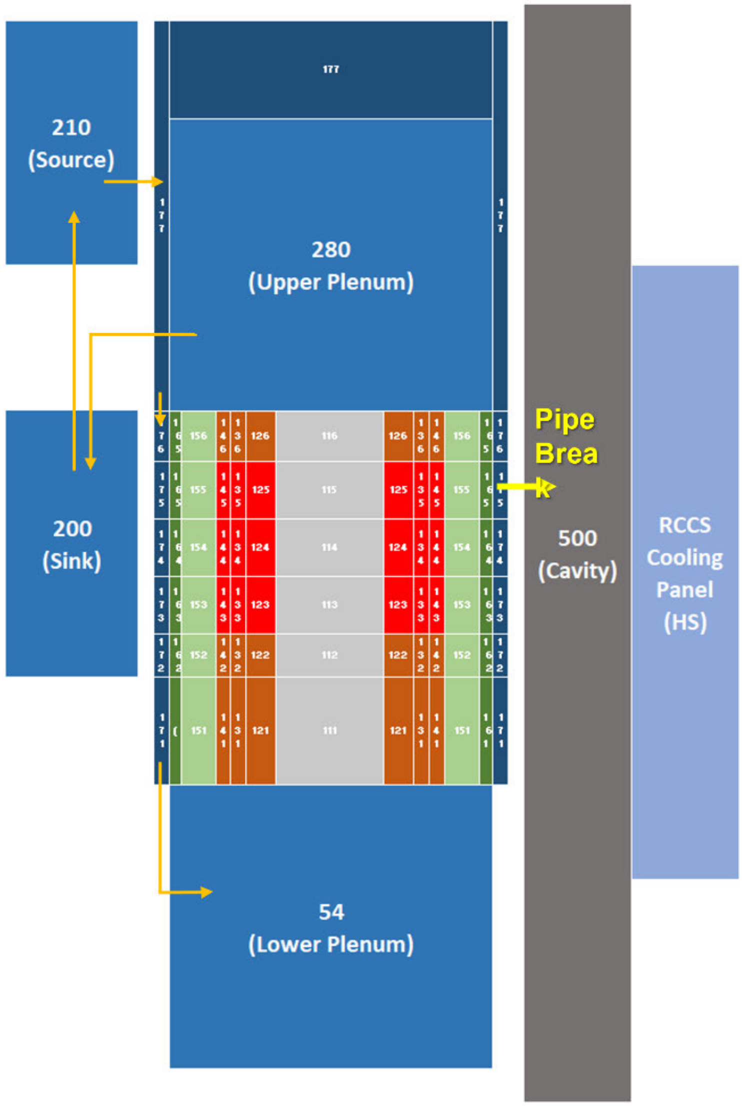


**Figure 10. MELCOR model of FMR control volume**

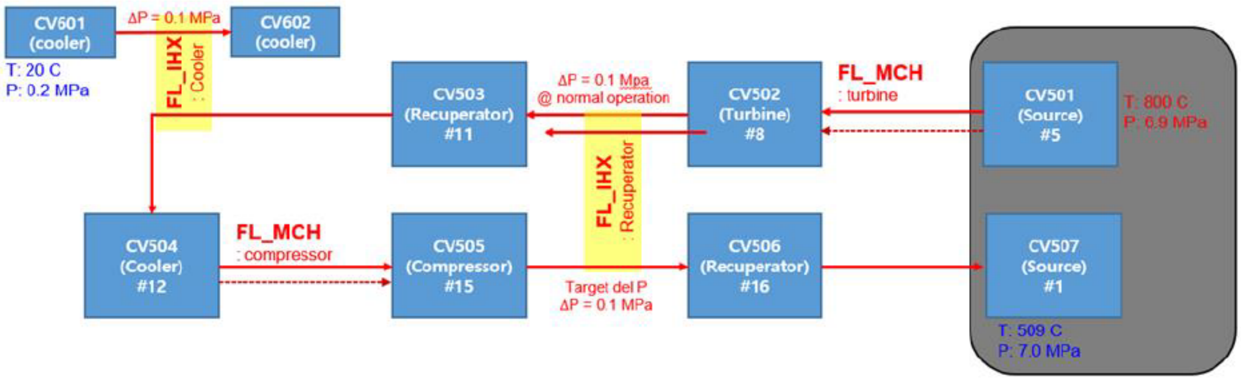


**Figure 11. PCS loop model**

Figure 12 shows the refined core cell structures. In the refined FMR plant model, the reactor core is divided into 6 radial zones and 24 axial cells to simulate details of the fuel rod design such as the lower/upper end cap and lower/upper plenum and the reflector regions. The geometrical dimensions and corresponding masses of each component match the current FMR design specifications.

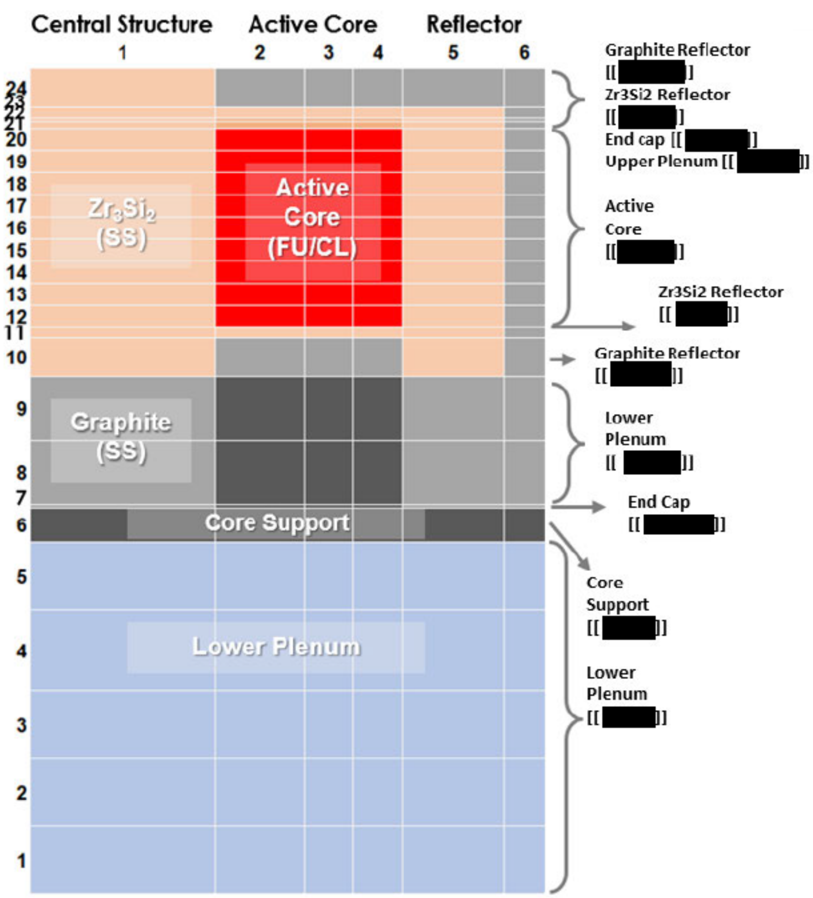


**Figure 12. MELCOR radial and axial nodes of the FMR core cells**

The thermal-hydraulic parameters of the FMR MELCOR model are consistent with the FMR design parameters: the inlet and outlet helium temperatures of 509 °C and 800 °C, respectively, the frictional pressure drop through the PCS budgeted to be 0.2 MPa, with total reactor helium flow of 66 kg/s, core inlet mass flow of [[redacted]], central structure inlet mass flow of [[redacted]], and reflector inlet zone mass flow of [[redacted]].

### 4.1.2. Heat Structures

Figure 13 shows the heat structure nodes of the FMR model. Three sets of heat structures are defined in the reactor model: the core barrel, the reactor pressure vessel, and the upper boundary of the core. These are denoted as HS3200X, HS3300X, and HS3400Y, respectively, where X refers to the axial level and Y refers to the radial ring of each specific heat structure.

The core barrel, which contains the entire core and the $B_4C$ shield, is a region of stainless-steel between the graphite reflector and the downcomer. The geometry of the core barrel heat structures is 'CYLINDRICAL'. Their initial temperature is set based on the steady-state calculation. The elevation parameter of each heat structure is given by the lowest point of that axial level defined in the CVH package input. The orientation of each core barrel heat structure is vertical.

There is no internal power source defined for any heat structure in this model. Three temperature nodes were defined radially for HS3200X: the first at [[redacted]], the second at [[redacted]] and the third at [[redacted]]. These represent the $B_4C$ shield and inner and outer walls of the stainless-steel wall for a total thickness of [[redacted]]. The left and right boundary conditions are the same for all core barrel heat structures HS32009-HS32023. The boundary condition on both sides is 'CalcCoefHS', meaning that a convective boundary condition is applied with the heat transfer coefficients calculated by the HS package. The flow is set as EXTERNAL on the inner (left) boundary.

Similar to the core barrel heat structures, the RPV geometry is also cylindrical with a vertical orientation. There is no internal power source defined for any heat structure in this model. There are two temperature nodes input values defined at [[redacted]] and [[redacted]] in radius for the inner and outer walls with a total thickness of [[redacted]].

The final set of heat structures in the reactor model are the upper boundary heat structures, HS3400Y, where Y ranges from 1 to 6, each value indicating the radial ring that a heat structure occupies. The innermost ring contains the central structure, followed by three rings of fuel, fifth ring for the $Zr_3Si_2$ reflector, and the last ring for the graphite reflector. The upper boundary heat structures is mandatory for the MELCOR core package input for typical PWR design. Thus, the thickness of HS3400Y is 1mm, which has negligible thermal inertia. Instead, FMR design has the top portion of the core barrel between the inner upper plenum and the outer upper plenum. This is modeled using HS34007 (Fig. 13). It has a consistent surface area, thickness as the design to match the thermal inertia.

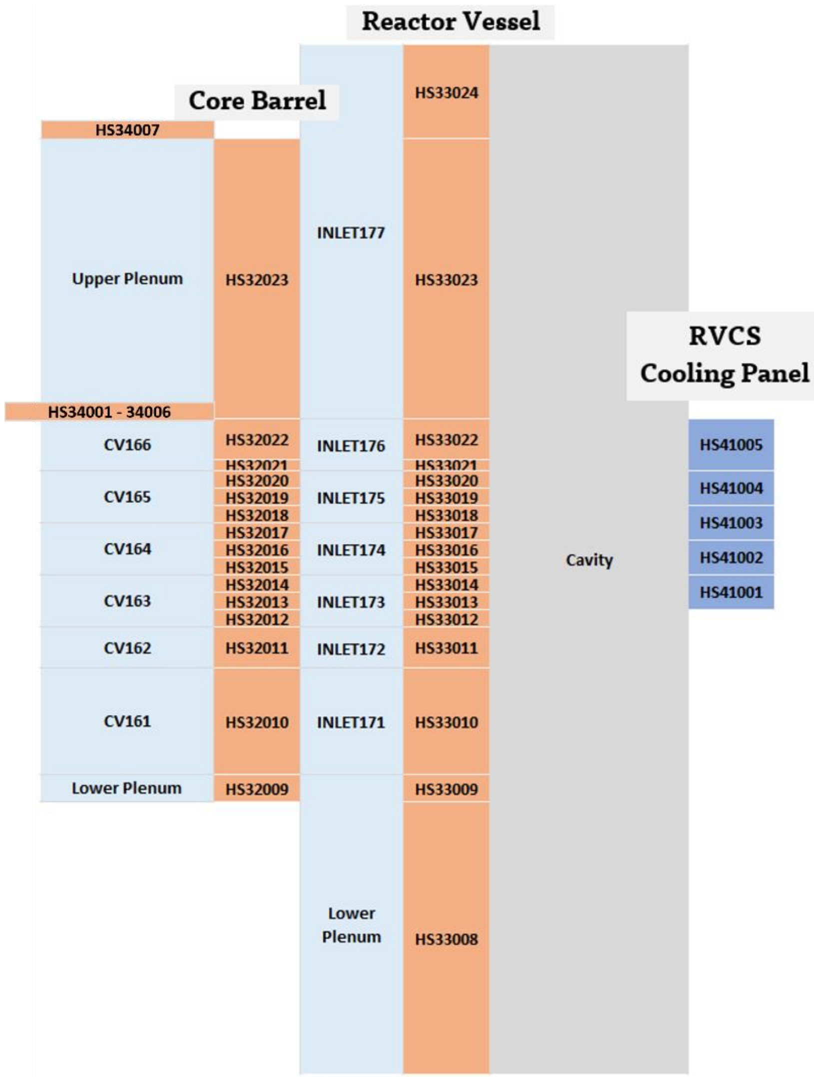


**Figure 13. MELCOR heat structure nodes of the FMR core**

### 4.1.3. Flow Paths

Helium flows from the source CV to the first inlet CV, named CV177, down through the rest of the downcomers and into the lower plenum. Going up through each radial ring of the core, the helium merges in the upper plenum (CV280) and exits to the sink. Apart from the axial flow paths in the core region, there are additional flow paths that connect each control volume radially to the adjacent control volume(s) representing circulation between the assemblies. These flow paths are defined to connect the midplane of the control volumes they link and have forward/reverse loss coefficients and choked flow forward/reverse discharge coefficients specified. These coefficients of axial FLs consider the effect of entrance/exit and the spacer grid for the loss coefficients.

The connections between control volumes are defined with flow paths. To simulate the failure of the reactor vessel, a valve opens the flow path connecting the reactor vessel and the containment building (Fig. 10, yellow arrow). Likewise, for simulation of the failure of containment building, a valve opens the flow paths connecting the reactor building and the environment. Those valves are controlled by control functions which define the time to open of the flow path.

Each flow path connects two control volumes, one referred to as the from-volume and the other as the to-volume defining the direction of positive flow. An arbitrary number of flow paths may be connected to or from each control volume. Parallel paths connecting the same two volumes are allowed. Mass and energy are advected through the flow paths, from one volume to another, in response to solutions of the momentum equation.

*1) Fuel Regions*

The flow area within the fueled regions of the core was calculated based on the design specifications. The flow area of coolant channels is obtained by substituting the area of fuel rods and central guide tube, $A_{flow}$. To still maintain conditions that simulate the actual flow of coolant within the reactor, the hydraulic diameter ($d_H$) is calculated based on the individual coolant channel size.

$$A_{flow} = A_{assembly} - A_{fuel\ rod} - A_{guide\ tube} \quad \text{Eq. 236}$$

$$d_H = \frac{4A_{wetted}}{P_{wetted}} \quad \text{Eq. 247}$$

*2) Central Structure Region*

According to design specifications, [[redacted]] of mass flow goes through the central structures. Based on the FMR design on the central structure assemblies and the iterative calculation, the flow area was determined as [[redacted]]. The hydraulic diameter is [[redacted]] considering the gap dimension between assemblies.

*3) Reflector Regions*

Based on design specifications, [[redacted]] of mass flow goes through the $Zr_3Si_2$ and graphite reflector region. The flow areas are [[redacted]] and [[redacted]], respectively. The hydraulic diameter is [[redacted]] same as central structure region since the gap dimension is identical.

*4) Helium Inlet Channels*

The total flow area through the helium inlet channel is calculated to be [[redacted]]. The hydraulic diameter for the helium inlet channels is [[redacted]], calculated as the hydraulic diameter of an annulus.

$$d_H = D_{out} - D_{in} \quad \text{Eq. 258}$$

### 4.2. Containment Model

To simulate the depressurization of the primary system due to a pipe break, an additional flow path is modeled from the helium inlet pipe (CV 175) to discharge into the containment. The containment is divided into two rooms: one for the reactor and another for the PCS with one more control volume for the control building above the containment, as shown in Figure 14. The flow path to the environment was added from the control building to the environment to provide the needed information for any MACCS dose evaluations.

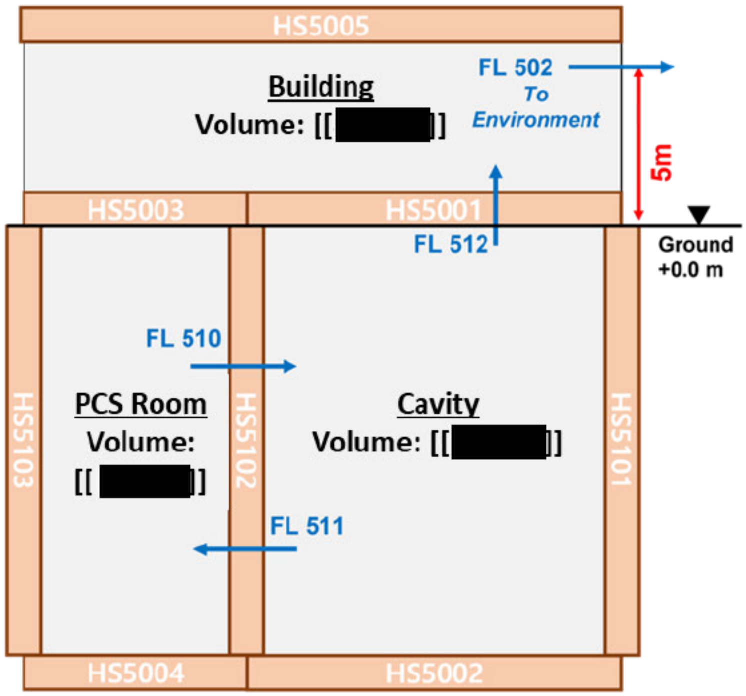


**Figure 14. Control volumes and heat structures for containment node**

### 4.3. Radionuclide Inventory

Initial RN inventories and distributions must be specified for the core. Initial distribution of RN masses in the core is specified using the input RN_FPN. The power distribution value is used for RN_FPN. Data for the time-dependent, isotope-wise fission product inventory following decay was generated by the ORIGEN. Here, the inventory for 24 years of irradiation (i.e., average fuel irradiation time) is utilized. One thing to note is that the use of RN inventory at 24-year of irradiation for the whole core is a conservative assumption. The ORIGEN outputs the masses, activity, and decay heat power of each isotope. The total decay heat is also calculated by ORIGEN and utilized in MELCOR.

MELCOR does not track specific isotopes but instead groups of elements in radionuclide classes. A Python script is used to sum the isotope masses and decay heats from ORIGEN output for each of these radionuclide classes. Table 3 summarizes MELCOR radionuclide classification. The developed Python script also calculates the distribution of iodine, cesium, and molybdenum across the CS, MO, CSI, and CSM classes based on the references, which define fission product inventories and decay heat during a severe accident.[33,34,35]

**Table 3. MELCOR/MACCS radionuclide classification**

| Number | Chemical group | Header element | Elements |
|---|---|---|---|
| 1 | Noble gases | XE | Xe, Kr, Rn, He, Ne, Ar, H, N |
| 2 | Alkali metals | CS | Cs, Rb, Li, Na, K, Fr, Cu |
| 3 | Alkaline earths | BA | Ba, Sr, Be, Mg, Ca, Ra, Es, Fm |
| 4 | Halogens | I2 | I, Br, F, Cl, At |
| 5 | Chalcogens | TE | Te, Se, S, O, Po |
| 6 | Platinoids | RU | Ru, Pd, Rh, Ni, Re, Os, Ir, Pt, Au |
| 7 | Transition metals | MO | Mo, Tc, Nb, Fe, Cr, Mn, V, Co, Ta, W |
| 8 | Tetravalents | CE | Ce, Zr, Th, Np, Ti, Hf, Pa, Pu, C |
| 9 | Trivalents | LA | La, Pm, Sm, Y, Pr, Nd, Al, Sc, Ac, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf |
| 10 | Uranium | UO2 | U |
| 11 | More volatile main group metals | CD | Cd, Hg, Pb, Zn, As, Sb, Tl, Bi |
| 12 | Less volatile main group metals | AG | Sn, Ag, In, Ga, Ge |
| 13 | Boron | BO2 | B, Si, P |
| 14 | Water | H2O | 'WT' |
| 15 | Concrete | CON | 'CC' |
| 16 | Cesium iodide | CSI | Taken from Classes 2 and 4 |
| 17 | Cesium molybdate | CSM | Taken from Classes 2 and 7 |

### 4.3.1. CSI Class

For the CSI class, the total mass of iodine class in the core is assumed to be present as cesium iodide class. The corresponding mass of cesium required to make up the compound was calculated using the ratio defined in the MELCOR best practice State-of-the-Art Reactor

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33 P. W. Humrickhouse, B. J. Merrill, "Status Report on Modifications to MELCOR for modeling of Accident Tolerant Fuel (ATF)," INL/EXT-17-43162, Idaho National Laboratory, 2017.
34 J. Cardoni, "Radionuclide Inventory and Decay Heat Quantification Methodology for Severe Accident Simulations," SAND2014-17667, Sandia National Laboratories, 2014.
35 L. Kmetyk, L. Smith, "Summary of MELCOR 1.8.2 calculations for three LOCA sequences (AG, S2D, and S3D) at the Surry Plant," NUREG/CR-6107, SAND-93-2042; Sandia National Laboratories, 1994.

Consequence Analyses (SOARCA).[36] Figure 15 shows reclassification of the CSI class. The cesium radionuclide mass calculated from ORIGEN code was subtracted from the total cesium element mass and added to the CSI class.

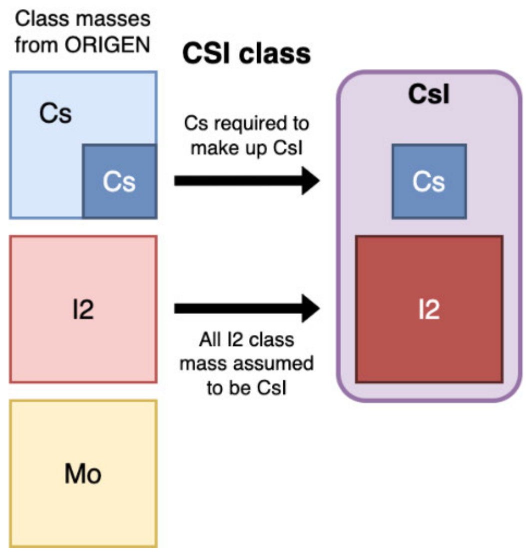


**Figure 15. Reclassification of CSI class elements**

Release of radionuclides can occur from the fuel pellets as well as from the fuel-clad gap. It is assumed that the gaps in each radial ring can communicate axially between core cells, so when the cladding temperature in any core cell exceeds the cladding failure temperature specified by the user, the entire gap inventory in that ring is released. The initial gap inventory is specified by RN1_GAP by RN class. The gap inventories for Xe, CS, BA, CsI, and TE class are 3%, 5%, 0.0001%, 5%, and 0.01%, respectively.

### 4.3.2. CS Class

Figure 16 depicts reclassification of CS class from CSI class and pure cesium. Since iodine in the CSI class contains entire amount of iodine present in the reactor, 5% of iodine from the CSI class will be present in the gap. In other words, 95% of iodine is assumed to be present in the CSI class while 5% is in the gap as CS class.

[36] "MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," NUREG/CR-7008, U.S. Nuclear Regulatory Commission, 2014.

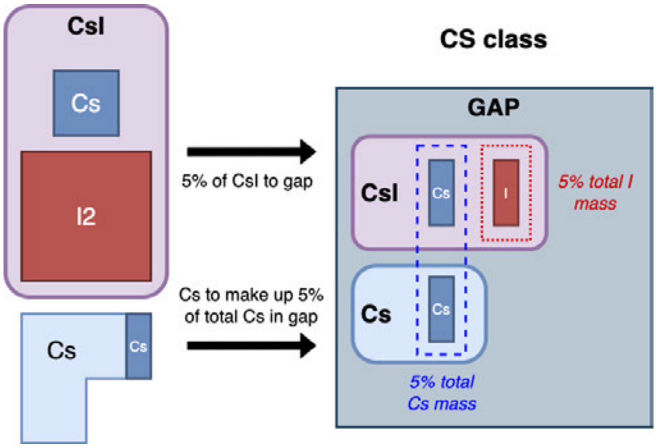


**Figure 16. Reclassification of CS class**

### 4.3.3. CSM and MO Classes

As shown in Figure 17, the remaining cesium mass left after the CSI and CS class contributions is assumed to be present in the CSM class. The ratio between cesium and molybdenum is 0.73479/0.26521 in the CSM class. The molybdenum mass is subtracted from the total amount based on the ratio given and added to the CSM class. The remaining molybdenum mass is then allocated to the MO class as shown in Figure 18.

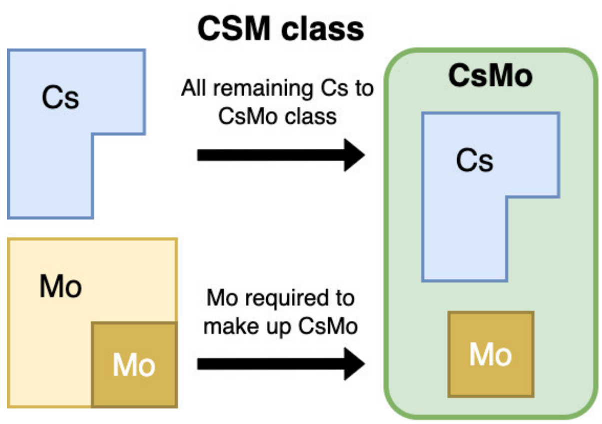


**Figure 17. Reclassification of CSM class**

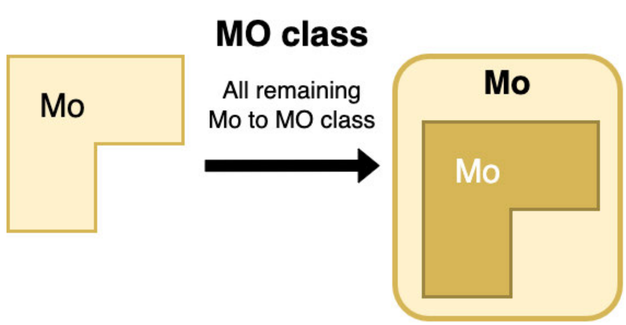


**Figure 18. Reclassification of MO class**

### 4.4. MELCOR Fission Product Transport

Given fuel rod failure and fission product release into the primary system, these materials would likely experience lower temperatures in the primary system and subsequently into the containment. This will cause the condensable fission product materials to form aerosols at submicron sizes. MELCOR has the capability to estimate the RN transport of these condensed aerosols as well as the non-condensable RN gases. The calculation of aerosol agglomeration and deposition processes is based on the MAEROS model,[37] which is part of the MELCOR code.

The MAEROS model is a multi-sectional, multi-component aerosol dynamics model that evaluates the size distribution of each type of aerosol, and the composition as a function of time. The lower and upper bound of aerosol diameter is specified by user input (RN1_ASP). In the current simulations, the lower bound is 1 µm and the upper bound is 50 µm.

Aerosols can directly deposit onto heat structure and pool surfaces through several processes, including gravitational settling, diffusion to surfaces due to temperature differences (thermophoresis), and compositional concentration differences (diffusiophoresis). As a default, all heat structure surfaces are automatically designated as deposition surfaces for aerosols using information from the HS package and can specify the surface orientation.

Aerosol deposition on a certain surface can also be disabled through input on the RN1_DS. In the FMR model, all the HSs are specified as possible aerosol deposition surfaces with appropriate orientation. Furthermore, the number of sections in the aerosol calculation, components, and material classes are specified in the input RN1_DIM. In the current calculation, it is assumed as 10 sections, 2 components, and 17 classes, which are again the recommended default values.

Aerosols can settle from one control volume to another through flow-path areas. Such areas correspond to open flow paths between the control volumes, through which aerosols and radionuclide gases are also transported. The appropriate settling areas are specified in the RN1_SET input which match the flow area of respected flow-paths. Aerosols are not transported through these areas if the settling area is blocked by a liquid pool, but that is likely a rare case for the FMR design.

Aerosols can agglomerate, resulting in a larger size than the user-defined maximum diameter. These aerosols are assumed to immediately deposit onto horizontal heat structures or to settle on the appropriate surfaces. Even though condensation, evaporation and hygroscopic behavior is one of the most important transport mechanisms, it is not considered as important mechanism in FMR model since water does not normally exist except as steam for the water ingression scenarios. For the same reason, the pool scrubbing model is not considered.

[37] F. Gelbard, “MAEROS User Manual,” NUREG/CR-1391, SAND-80-0822, Sandia National Laboratories, 1982.

### 4.5. MELCOR-MACCS Coupled Simulation

MACCS[38] is a fully integrated, engineering-level severe accident code developed by SNL for the NRC to analyze the offsite consequences of a hypothetical release of radioactive material to the environment. The code performs a probabilistic calculation of the atmospheric transport and deposition of radionuclide releases given the site weather data, the exposure as a result of inhalation, ingestion, and external irradiation and emergency response, land contamination and long-term remediation in a probabilistic approach. The consequence analysis includes for short- and long-term exposure and health impact, dose to the environment and the economic impacts from the protective and remediation actions.

The key input variable for each MACCS calculation is the accident plume segments that are released from the containment following an accident simulation, i.e., number of plumes, release duration, sensible heat, flow rate, and the radioactivity of the isotopes in the plume. Figure 19 shows the schematic diagram of MELCOR-MACCS code system:

- Users can operate MACCS through the Windows application WinMACCS for user data input, set options and post-processing of MACCS calculation output.
- MELMACCS converts MELCOR output data (plot, ".ptf" files) to MACCS input files, that determines the accident plume source term.

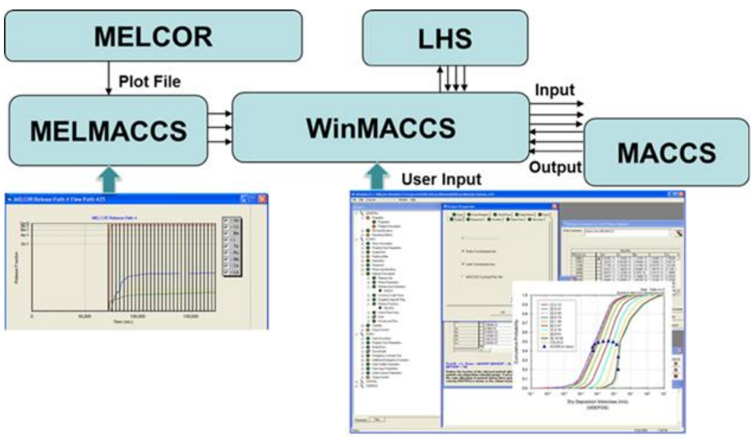


**Figure 19. MELCOR-MACCS calculation flow**

The MACCS code basically includes following three physical modeling modules:

- *ATMOS* simulates the movement, dispersal and settling of the accident plume,

[38] USNRC, "MACCS (MELCOR Accident Consequence Code System) (NUREG/BR-0527)," https://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0527/index.html, 2023.

- *EARLY* simulates short-term mitigation actions and health consequences like evacuation, and
- *CHRONC* simulates chronic longer-term effects and mitigation actions like latent cancers.

The ATMOS module simulates plume transport based on the plume qualities with user definitions. This ATMOS module uses either the Gaussian or the HYSPLIT dispersion model[39] along with a hybrid gravitational or user-data based deposition model to estimate dispersion of the plum from the source. Users can define plume rise and meander models and the long-range dispersion model.

The EARLY module simulates short-term exposure to human and mitigation or evacuation actions. The user can define evacuation timing, duration, faction, and other information on immediate response in accident.

The longer-term impact is simulated by the CHRONC module. The user can define the cost of relocating people, temporarily or permanently, the amount of farmland contamination and the effectiveness and cost of decontamination operations. Additional environmental inputs are generated from different auxiliary programs for the population and land use around the plant (SecPop) and conversion of radiation ingestion into health consequences (COMIDA2).

In this study, MACCS calculates total fatalities, total dose exposure and total economic costs. The MACCS reports results as probability distributions, i.e., complimentary cumulative distribution functions (CCDFs), with metrics like the mean, median ($50^{th}$ percentile) and peak trial results.

It should be noted that the FMR plant site has not been determined. As such, the input and approaches of the radiation consequence analysis rely on MACCS default options as follows:

- A default set of meteorological data, accident site, dose coefficient file, food ingestion file (i.e., Surry plant site and weather) and the default MACCS simulation options are used.
- For this probabilistic approach, the Latin Hypercube Sampling (LHS) method[40] is used, i.e., 1000 run samples with weather, as the uncertain variable affecting the results.

## 5. SAMPLE ANALYSIS OF DEPRESSURIZED LOSS OF FORCED COOLING

The Depressurized Loss of Forced Cooling (DLOFC) accident was selected as an example scenario to demonstrate the MELCOR-MACCS coupled simulation of the FMR plant and evaluate the radiation source term and its dose consequence. This accident is initiated by an assumed

[39] "HYSPLIT," https://www.arl.noaa.gov/hysplit/, Air Resources Laboratory, 2023.
[40] "Latin Hypercube Sampling: Simple Definition," https://www.statisticshowto.com/latin-hypercube-sampling/, 2023.

instantaneous breach in the FMR cross-duct vessel followed by rapid blowdown of the helium coolant into the reactor containment with reactor shutdown. The normal PCS and MCS are assumed failed and the redundant passive RVCS removes the decay heat from the core during the accident progression.

### 5.1. MELCOR Simulation

The simulation starts at -20,000 sec to attain steady-state condition for FMR before accident initiation. The accident is then initiated at 0 sec with the pump trip and the reactor trip simultaneously, even though the reactor trip may be a few seconds delayed in a real situation. The compressor flow linearly decreases with a linear flow rate decrease over a few seconds. Then, we assumed the containment failure at 10 hours on purpose to develop the MELCOR-MACCS analysis methodology. In this simulation, the RVCS cooling panel keeps the water temperature constant during this entire accident calculation.

#### 5.1.1. Heat Transfer Modeling

In case of severe accidents (like a depressurized loss-of-flow accident), the radial heat transfer path becomes more important – especially, the dominant role of radiation heat transfer in transferring heat generated from fuel core cells to the outer RPV wall, and then eventually to the RVCS cold wall – since the natural circulation cooling by helium gas is not efficient under the depressurized condition.

The radiation heat transfer between the core barrel and the RPV wall is simulated through a structure-to-structure radiation model (HS_RD) with the view factors calculated through Monte Carlo method (MCNP) and with an emissivity of 0.8. The heat transfer from the RPV wall to the RVCS cooling panel and the concrete walls is simulated through the enclosure radiation model (HS_RAD) with an emissivity of 0.8.

The view factor for the radiation heat transfer inside the core is another important parameter used to determine the core cooling behavior. In MELCOR, the values of ‘FCELR’ (radial view factor) and ‘FCELA’ (axial view factor) are needed to properly obtain the heat transfer between the core cells containing the fuel rod assemblies (the dominant heat transfer path during a depressurized accident). View factors for both radial and axial radiation heat transfer were calculated within entire core region which are consistent with the correlations noted in the MELCOR reference manual.

The radial and axial view factors are summarized in Table 4 and Table 5, respectively. The radial view factors of the fuel are given for the entire fuel length (axial level 6-22), while the axial view factors of the fuel are given for the active fuel height (axial level 12-20).

**Table 4. Summary of radial radiative view factors in core region**

| Axial Level | Zone 1>2 | Zone 2>3 | Zone 3>4 | Zone 4>5 |
|---|---|---|---|---|
| **6-22** | **0.054** | **0.051** | **0.060** | **0.115** |
| 23-24 | 0.164 | 0.258 | 0.305 | 0.352 |

Note: FCELR view factor is from core cell (i,j) to core cell (i+1,j).

**Table 5. Summary of axial radiative view factors in core region**

| Axial Level | Zone 1>2 | Zone 2>3 | Zone 3>4 | Zone 4>5 |
|---|---|---|---|---|
| 6 | 0.036 | 0.042 | 0.042 | 0.041 |
| 7 | 0.282 | 0.345 | 0.341 | 0.339 |
| 8-9 | 0.005 | 0.023 | 0.022 | 0.022 |
| 10 | 0.042 | 0.037 | 0.036 | 0.036 |
| 11 | 0.092 | 0.117 | 0.115 | 0.114 |
| **12-20** | **0.044** | **0.063** | **0.061** | **0.061** |
| 21 | 0.150 | 0.181 | 0.177 | 0.177 |
| 22 | 0.292 | 0.345 | 0.340 | 0.340 |
| 23 | 0.093 | 0.118 | 0.116 | 0.115 |
| 24 | 0.036 | 0.037 | 0.036 | 0.036 |

Note: FCELA view factor is from core cell (i,j) to core cell (i+1,j+1…n ).

The conduction heat transfer path, simulated through the conductance model (COR_HTR) from Zone 5 to Zone 6 and from Zone 6 to core barrel, includes the gas gap between the reflector assemblies and the gas gap between the graphite and the core barrel. Currently, the gap dimensions are assumed as 1 cm of the gap between Zone 5 and Zone 6, and 1 mm of the gap between Zone 6 and the core barrel. The radiation heat transfer is not considered when conduction is the only heat transfer path with full contact between the reflectors and between the graphite reflector and the core barrel.

### 5.1.2. Accident Progression

As the pipe break occurs, the reactor is rapidly depressurized to below 0.7 MPa, approaching to an equilibrium state within ~200 seconds as shown in Figure 20. Then, the pressure becomes atmospheric pressure after 10 hours as containment fails. In the current model, the reactor free volume is [[redacted]] including the PCS free volume of approximately [[redacted]]. The total containment system volume is [[redacted]]. Even though the reactor was depressurized, there are two natural circulation flow patterns that develop through the core due to the temperature differences between the core (hot) and the downcomer (cold) and between the active core (hot) and the reflector (cold).

The blue plot (FL-MFLOW.54) in Figure 21 represents the natural circulation from the core outlet through the PCS to the downcomer and into the core. The pink and purple plot (FL-MFLOW.144 and 154) shows the natural circulation inside the core representing the downward flow in the reflector zone. Figure 22 shows the temperature profile of core at 12 hours after the pipe break. The maximum temperature occurs at the Zone 2 on axial level 16, which is the center part of the active fuel.

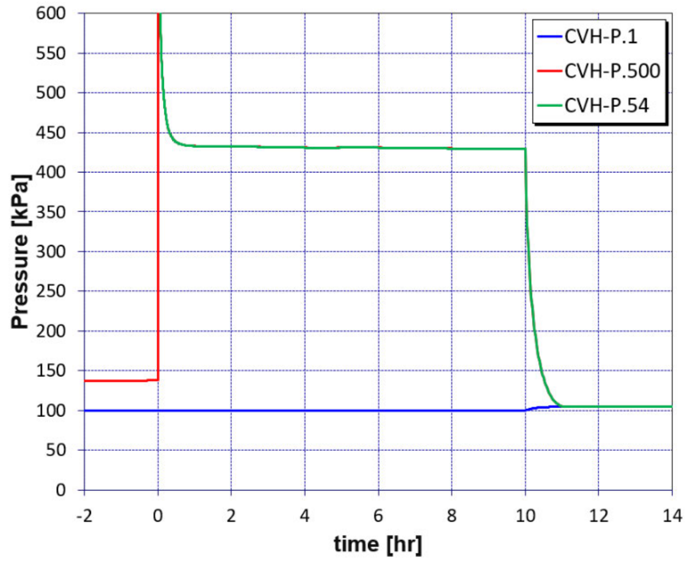


**Figure 20. Pressure history of lower plenum (green), containment (red), and environment (blue) under DLOFC**

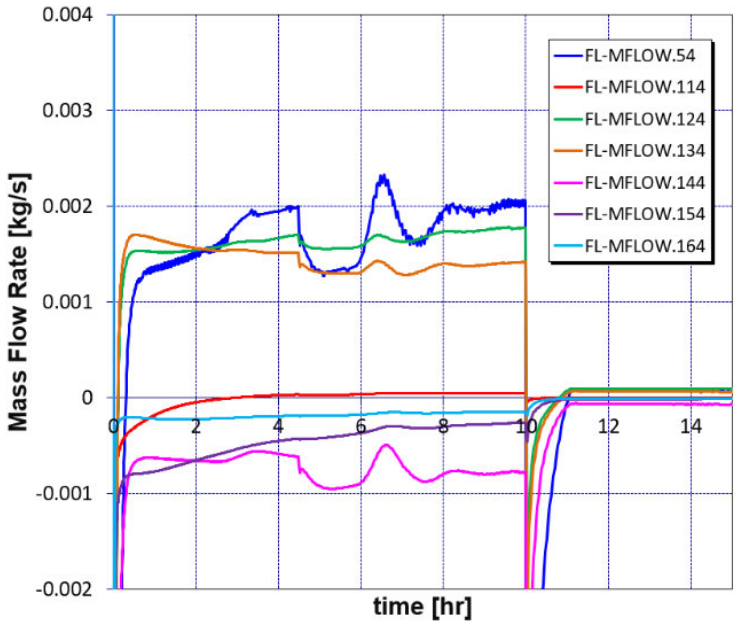


**Figure 21. Mass flow rate of reactor during DLOFC: (FL54) from downcomer to lower plenum, (FL1x4) from CV1x3 to CV1x4 in each ring x, and (FL300) leakage to containment**

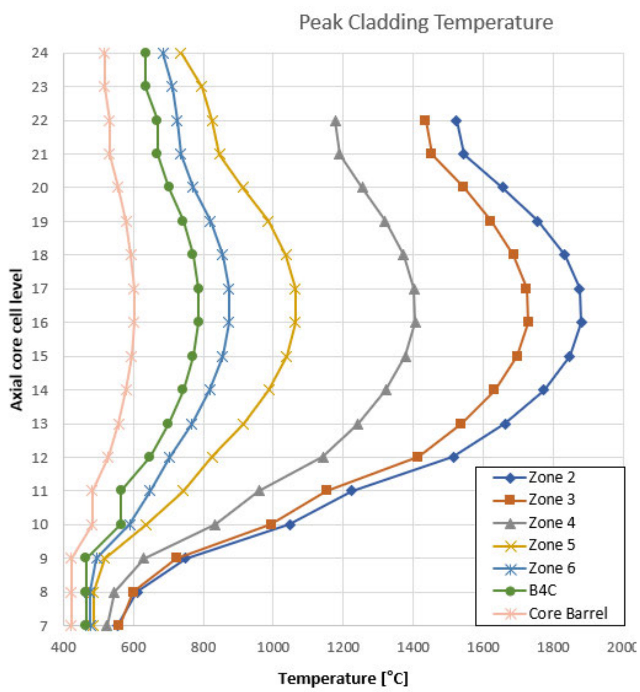


**Figure 22. Cladding, reflector, and core barrel temperature profile at 12-hour during DLOFC**

The recommended design limit of the $SiC_f/SiC$ composite cladding is 1800°C based on its degradation of mechanical strength, as discussed in Section 3.2.2. This design limit is used as the onset of fission product gap release in his study to generate the source term. Due to the blowdown of the helium from the RPV into the containment, the pressure of the containment is elevated and remains at 0.43 MPa.

The MELCOR information required by MACCS is defined in the flow path package. In this analysis, the flow path from the reactor building into the environment is considered as the leakage flow path for MACCS input. This provides the input data for MACCS using the defined flow path. In the current example, the leakage is defined as follows:

- The leakage is a horizontal flow to the environment with a flow path size (100 mm) that rapidly releases the radionuclide aerosols and containment gases.
- The flow path from the containment to reactor building actually determines the flow rate out to the environment since it has smaller size (50 mm), which is a containment failure point.

Total released RN mass history from the core is shown in Figure 23. The changes of RN inventory are summarized in Table 6.

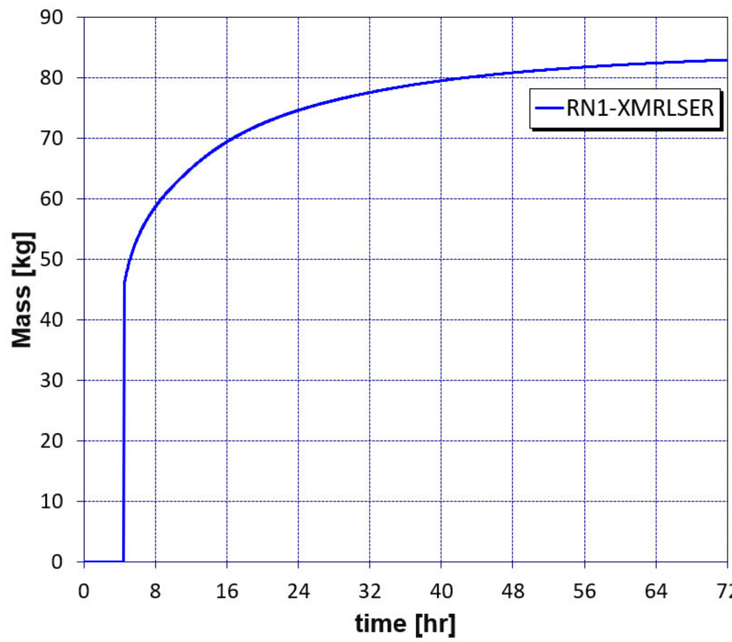


**Figure 23. Total released RN mass history from the core**

**Table 6. Released RN mass to the containment and environment**

| Number | Chemical group | Total initial inventory mass [kg] | Released mass to containment [kg] | Released mass to environment [kg] |
| --- | --- | --- | --- | --- |
| 1 | XE | 124.7 | 35.962 | 18.697 |
| 2 | CS | 15.3 | 0.182 | 0.056 |
| 3 | BA | 67.7 | 0.003 | 0.0007 |
| 4 | I2 | 0 | 0 | 0 |
| 5 | TE | 16.0 | 0.026 | 0.007 |
| 6 | RU | 83.7 | ~ 0 | ~ 0 |
| 7 | MO | 83.6 | 0.527 | 0.180 |
| 8 | CE | 668.6 | ~ 0 | ~ 0 |
| 9 | LA | 216.0 | 0.00004 | ~ 0 |
| 10 | UO2 | 16,407 | 0.003 | 0.0009 |
| 11 | CD | 3.3 | 0.007 | 0.002 |
| 12 | AG | 5.2 | 0.012 | 0.004 |
| 13 | BO2 | Default | 0 | 0 |
| 14 | H2O | Default | 0 | 0 |
| 15 | CON | Default | 0 | 0 |
| 16 | CSI | 16.282 | 0.008 | 0.003 |
| 17 | CSM | 105.11 | 0 | 0 |

### 5.2. MACCS Simulation

MACCS receives the radionuclide-specific release fractions from MELCOR to perform a consequence analysis. This data supplied to MACCS allows it to determine the proportion of mass and activity of each isotope within each radionuclide class in MELCOR.

#### 5.2.1. MACCS Environmental Impact

Except for the plume release mass from the containment, most input parameters were set as default in MACCS. The key variable to specify in the MACCS calculation is the plume segments: their number, their release duration, the sensible heat, the flow rate, and the activity of the isotopes in the plume. Weather data for the Surry reactor site was randomized using a standard LHS method to probabilistically model the plume for 1000 trials.

MACCS divides isotopes in aerosol form into 10 groups based on particle size (i.e., 0.1-0.186, 0.186-0.347, 0.347-0.645, 0.645-1.2, 1.2-2.24, 2.24-4.16, 4.16-7.75, 7.75-14.4, 14.4-26.9, 26.9-50 $\mu$m). Each of the particle sizes is handled separately. As part of converting MELCOR output files to MACCS inputs, MELMACCS estimates the composition of the plume based on the inventory of the core at the time of release and the released mass of each MELCOR chemical class.

After a user-specified time (i.e., one week), MACCS completes the emergency phase simulation by the EARLY system and starts intermediate (1 year) and long-term phase (49 years) simulation by CHRONC system. Broadly, this could be treated as lingering consequences of the accident after plant conditions have stabilized. Variables include factors like the cost of relocating people temporarily and permanently, the amount of farmland contamination and the effectiveness and cost of decontamination operations. Together, these models simulate the consequences of radionuclide releases.

In this example, there are no early fatalities. The long-term summary metrics are estimated: latent health effect (i.e., total cancer fatalities), population dose (i.e., total dose for 50 years) and economic consequence within the range of 10 mi (16.1 km), 50 mi (80.5 km) and 1000 mi (1609 km) from the accident. In addition, the projected peak dose can be estimated over in the short term (two hours) and over the long term (over thirty years).

### 5.2.2. Results of Consequence Analysis

The results of the sample DLOFC simulations were obtained for the cancer fatality, long-term dose, total economics costs, and the projected peak dose as given in Table 7. The "Non-zero consequence probability" indicates whether the consequence will occur. That is, if dose exposure was found, the value is 1, regardless of any weather effects. This indicates that the radiological impact is well simulated during a MACCS calculation.

The MACCS analysis calculates TEDE. The long-term dose is TEDE dose for the total remaining population in that specified area for 50-year period whereas the peak dose is for an individual at the specified location for 2 hours.

In the case of economics analyses, while the overall consequences have a non-zero probability of 1, some of the contributing factors (e.g., need to decontaminate farmland) have a non-zero probability of less than 1. This suggests that there might be no need to decontaminate the farmland or to relocate large groups of people for a long-term period.

For the projected peak dose, the mean TEDE is dose to an individual located at 0.2 km from the FMR containment over worst 2-hour period. This is a result from 1000 trials with different weather patterns. Note that the peak value of TEDE is [[redacted]] but the probability of occurrence is [[redacted]].

Assuming that the DLOFC is the bounding severe accident, these MACCS long-term dose consequence results can be compared to the NRC regulatory requirements to determine the exclusion area boundary (EAB) and the LPZ distances following the requirements of CFR 50.34(a)(1) and 52.17(a)(1).

**Table 7. Consequence analysis results of DLOFC**

| Consequence | Distance (km) | Non-zero consequence probability | Mean | 50th percentile | Peak consequence | Peak probability |
|---|---|---|---|---|---|---|
| Total cancer fatalities | 0 - 1609 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 0 - 80.5 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 0 - 16.1 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| Total long-term dose (person-Sievert) | 0 - 1609 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 0 - 80.5 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| Total economic costs (USD) | 0 - 1609 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 0 - 80.5 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| Projected peak dose after 2 hours (Sievert) | 0 - 0.2 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 0.2 - 0.5 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 0.5 - 1.2 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |
| | 1.2 - 1.6 | 1 | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] | [[[redacted]]] |

Table 8 shows the inventory of the key radionuclides released to the environment after the accident. These values were calculated for the 50-year period. In the MELCOR model, these radionuclides are released as a source term within 72 hours. The TEDE is calculated as the sum of the effective dose equivalent (EDE) for external exposures and the committed effective dose equivalent (CEDE) for internal exposures following the exposure-to-committed effective dose equivalent factors for inhalation of radioactive material in Federal Guidance Report (FGR) Number 11.[41,42]

**Table 8. Total release mass and activity of major isotopes**

| Isotope | Activity (Bq) | Mass (g) |
|---|---|---|
| Sr-90 | [[[redacted]]] | [[[redacted]]] |
| I-131 | [[[redacted]]] | [[[redacted]]] |
| Te-132 | [[[redacted]]] | [[[redacted]]] |
| Cs-134 | [[[redacted]]] | [[[redacted]]] |
| Cs-137 | [[[redacted]]] | [[[redacted]]] |

41 K. L. Compton, A. Hathaway, E. Dickson, “Use of MACCS Dose Coefficient Files to Compute Total Effective Dose Equivalent,” U.S. Nuclear Regulatory Commission, 2021.
42 K. F. Eckerman, A. B. Wolbarst, A. C. B. Richardson, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report No. 11, U.S. Environmental Protection Agency, 1988.

## 6. CONCLUSION

This report of the source term calculation methodology describes the method for estimating the release of fission product from the FMR power plant into the environment. It addresses the regulatory requirements, computer codes, accident analysis, and consequence analysis by the coupled simulations of the MELCOR and MACCS codes. The outcomes of this source term calculation of the FMR plant are as follows:

- The FMR system model was built by MELCOR using the up-to-date design specifications of the FMR.
- Interface programs and procedures have been established to conduct the coupled simulation of the MELCOR and MACCS for the source term and consequence analysis.
- The sample accident analysis confirmed that the overall calculation procedure is correct, and the calculation results are consistent with data and models used for the analysis.
- The source term and consequence analysis by MELCOR-MACCS will provide the bases for determining the design parameters relevant to the reactor, plant, and environmental safety.

It should be noted again that the objective of the sample calculation in this report is to verify the overall calculation procedure not to evaluate the performance of the design. Several assumptions were used to intentionally generate the source term and propagate it into the environment along with default data and options of the MELCOR and MACCS. Examples of assumptions used in the sample calculation are as follows:

- The shutdown cooling is primarily conducted by the radiative heat transfer in the active core zone which was approximately calculated by the effective intercell view factors.[43] Though this is an option for a multi-rod core, the accuracy of this approach and the view factors used in this report have not been validated. In fact, an independent computational fluid dynamics (CFD) analysis of the shutdown cooling showed a higher radiative heat transfer capability in the core when compared with the MELCOR analysis.[44]
- This analysis used three fuel zones with an individual average power in each zone. So, if the peak cladding temperature of the average fuel rod exceeds the cladding temperature limits of 1800ºC, all the fuel rods in that zone release the fission gases into the core. This is a very conservative assumption because not all the fuel rods in that zone exceed the temperature limit.

[43] L. Humphries, “MELCOR Multi-Rod Model,” SAND2019-13344PE, Sandia National Laboratories, 2019.
[44] J. Rohrbacher, “Verification of Reactor Vessel Cooling System Passive Safety,” 30599601R00034, General Atomics Electromagnetic Systems, 2023.

- This analysis assumed that there are passages of gas flow between the containment and reactor building and between the reactor building and environment. The FMR containment is sealed. The leak rate of the containment has not yet been determined for normal operation and accident conditions. More reasonable leak rates should be used for the source term calculation to accurately estimate the environmental effects.

It is true that the results of modeling and simulation will vary as there are uncertainties remaining in the data, models, computing tools, and the FMR design parameters. For the source term to be ultimately used for the licensing applications, it is recommended to continue and expand the source term modeling and simulation as follows:

- Verify the data and models used for the source term calculation to be consistent with the FMR design.
- Evaluate other licensing basis events (LBEs) to identify the main characteristic and envelopes of the source term associated with the FMR design.
- Conduct the sensitivity analysis to the data and models to identify the major uncertainties significantly affecting the source term, including approximations used to model power distribution in the core.

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