For pipe break scenarios in other Flibe containing SSCs (except the vessel) not connected to the reactor, the core does not experience a transient from reactor trip.

In order to ensure that the design features mitigating a salt spill event are sufficient to keep the consequences bounded by the MHA, the following key figures of merit must be evaluated:

- Peak TRISO temperature to limit diffusion of radionuclides
- ~~Peak TRISO temperature~~TRISO failure probability to limit incremental TRISO layer failures
- Peak Flibe-cover gas interfacial temperature to limit evaporation mass transfer of radionuclides
- Peak vessel and core barrel temperatures to prevent vessel failure and maintain long term cooling
- Aerosols generated by released Flibe to limit the materials at risk released
- Volatile products formed from the chemical reaction between Flibe and air, Flibe and stainless steel, and Flibe and insulation to limit the materials at risk released
- Mass loss of structural graphite due to oxidation to limit tritium release
- Mass loss of pebble carbon matrix due to oxidation to limit tritium release and prevent additional release of materials at risk
- Peak temperature of structural graphite to limit the tritium release
- Peak temperature of pebble carbon matrix to limit the amount of tritium release

3.2.2.2 Insertion of Excess Reactivity

A control system error or operator error causes a continuous withdrawal of the highest worth control element at maximum reactivity control and shutdown system (RCSS) drive speed. The reactivity insertion is detected by the reactor protection system which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The reactor decay heat removal system limits reactor temperature and fulfills the heat removal function.

A safe state is established when:

- The core is subcritical and long-term reactivity control is assured.
- The decay heat is being removed and long-term cooling is assured, where figures of merit temperatures are steadily decreasing and Flibe temperature remains above Flibe freezing temperature during the mission time of the decay heat removal system.

This narrative captures the limiting event of this postulated event category. Other events grouped in this category include:

- Reactivity insertion events caused by fuel loading error (e.g., errors in rate of fresh fuel injection, incorrect order of fuel insertion)
- Reactivity insertion events with concurrent pump trip
- Reactivity insertion events with normal heat rejection available
- Local phenomena leading to ramp insertion of reactivity
- Change in reactivity due to shifting of graphite reflector blocks

- Venting of gas bubbles accumulated in the active core
- Local phenomena leading to step insertion of reactivity
- Local negative reactivity anomaly (e.g., inadvertent single element insertion, cover gas injection)
- Reactivity insertion events during startup

The control element withdrawal at maximum speed, described above, is assumed to be the limiting event of this category. However, the amount and rate of reactivity insertion from other grouped events under insertion of excess reactivity (e.g., during the pebble loading error event, venting of accumulated gas bubbles in the active core) is compared with those from the control element withdrawal events. Additionally, the reactivity insertion due to Increase in Heat Removal events and design basis seismic event, respectively, is compared to the reactivity insertion of control element withdrawal events.

In order to ensure that the design features mitigating a reactivity insertion event are sufficient to keep the consequences bounded by the MHA, the following key figures of merit must be evaluated:

- Peak TRISO temperature to limit diffusion of radionuclides
- ~~Peak TRISO temperature~~TRISO failure probability to limit incremental TRISO layer failures
- Peak Flibe-cover gas interfacial temperature to limit evaporation mass transfer of radionuclides
- Peak vessel and core barrel temperatures to prevent vessel failure and maintain long term cooling
- Peak temperature of structural graphite to limit the tritium release
- Peak temperature of pebble carbon matrix to limit the amount of tritium release

3.2.2.3 Increase in Heat Removal

The primary coolant pump overspeeds, causing a surge insertion of cold Flibe into the core. The event is detected by the reactor protection system, which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The reactor protection system also trips the primary coolant pump. The reactor decay heat removal system limits reactor temperature and fulfills the heat removal function.

A safe state is established when:

- The core is subcritical and long-term reactivity control is assured.
- The decay heat is being removed and long-term cooling is assured, where figure of merit temperatures are steadily decreasing and Flibe temperature remains above Flibe freezing temperature during the mission time of the decay heat removal system.

This narrative captures the limiting event of this postulated event category. Other events grouped in this category include:

- Increase in heat removal due to overspeed of intermediate salt pump
- Increase in heat removal during low power operation

The increase in heat removal events are demonstrated to be bounded by the insertion of excess reactivity postulated event.

In order to ensure that the design features mitigating an increase in heat removal event are sufficient to keep the consequences bounded by the MHA, the following key figures of merit must be evaluated:

- Peak TRISO temperature to limit diffusion of radionuclides
- ~~Peak TRISO temperature~~TRISO failure probability to limit incremental TRISO layer failures
- Peak Flibe-cover gas interfacial temperature to limit evaporation mass transfer of radionuclides
- Peak vessel and core barrel temperatures to prevent vessel failure and maintain long term cooling
- Peak temperature of structural graphite to limit the tritium release
- Peak temperature of pebble carbon matrix to limit the amount of tritium release

3.2.2.4 Loss of Forced Circulation

The failure of the primary salt pump results in the loss of forced circulation. The reduced flow is detected directly or indirectly by the reactor protection system, which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The reactor decay heat removal system limits reactor temperature and fulfills the heat removal function.

A safe state is established when:

- The core is subcritical and long-term reactivity control is assured.
- The decay heat is being removed and long-term cooling is assured, where figures of merit temperatures are steadily decreasing and Flibe temperature remains above Flibe freezing temperature during the mission time of the decay heat removal system.

This narrative captures the limiting event of this postulated event category. Other events grouped in this category include loss of forced circulation due to:

- Blockage of flow path external to the reactor vessel in the primary heat transport system,
- Spurious pump trip signal
- Pump seizure
- Shaft fracture
- Bearing failure
- Pump control system errors
- Supply breaker spurious opening
- Loss of net-positive suction head (e.g., pump overspeed, low level)
- Loss of normal electrical power
- Loss of normal heat sink

There are two bounding events within this event category to evaluate the long-term passive cooling performance. One is to bound the overheating consequence, and another is to bound the downcomer freezing consequence. Two scenarios are considered for these two bounding events:

- The first event scenario (overheating) considers the limiting case to analyze the peak vessel and core barrel temperatures to prevent vessel failure and maintain coolable geometry. The most limiting reactor operation power and operating history are assumed.
- The second scenario (long-term overcooling) aim at the reactor performance evaluation in terms of coolant freeze prevention at downcomer. A spectrum of reactor decay heat levels and operating power levels are analyzed for this purpose.

For the overheating bounding event, the loss of forced circulation due to loss of normal electrical power is bounded by the primary salt pump failure scenario. The loss of power to the reactivity control and shutdown system mechanisms results in release and insertion of the control and shutdown elements. As such, the reactor power is reduced faster compared to other loss of forced circulation scenarios where the reactor trips on a reactor trip signal. For the long-term overcooling bounding event, the loss of normal electrical power event bounds other loss of circulation scenarios since this event has the least stored energy.

In order to ensure that the design features mitigating a loss of forced circulation event are sufficient to keep the consequences bounded by the MHA, the following key figures of merit must be evaluated:

- Peak TRISO temperature to limit diffusion of radionuclides
- ~~Peak TRISO temperature~~TRISO failure probability to limit incremental TRISO layer failures
- Peak Flibe-cover gas interfacial temperature to limit evaporation mass transfer of radionuclides
- Peak vessel and core barrel temperatures to prevent vessel failure and maintain long term cooling
- Peak temperature of structural graphite to limit the tritium release
- Peak temperature of pebble carbon matrix to limit the amount of tritium release

The only figure of merit for the long-term overcooling scenario is:

- Minimum reactor vessel inner surface temperature to prevent partial freezing within downcomer

3.2.2.5 Pebble Handling and Storage System Malfunction

There are three types of events in this event category: pebble handling and storage system (PHSS) break, loss of PHSS cooling, and grinding of a pebble in the pebble handling machine. However, the loss of PHSS cooling is an event mitigated through design of pebble storage system, and the grinding of pebble mitigated through the design of pebble extraction machine. The consequences of these two events are expected to be limited by the design specifications which are bounded by MHA consequence. Therefore, the PHSS break event is the assumed limiting event to be analyzed for this category.

The pebble handling and storage system transfer line breaks when pebbles are getting removed from the core, resulting in spilling of the pebbles within the transfer line into the reactor cell. This condition is detected directly or indirectly by the reactor protection system, which trips the pebble handling and storage system to stop pebble movement. For the spilled pebbles, the reactivity control function is fulfilled by the low fissile inventory of pebbles, which precludes criticality safety concerns, while heat transfer mechanisms within the room fulfills the heat removal function. The structural integrity of the pebbles maintains the confinement function. For the pebbles remaining in the pebble handling and storage system, the reactivity control, heat removal and confinement functions continue to be fulfilled by the system design resulting in a safe and stable state. The heat up of the pebbles in the PHSS mobilizes the Flibe accumulated on the piping. Air ingress into the PHSS and reactor cover gas region occurs through the break.

A safe state is established when:

- The movement of pebbles outside of the core has stopped and criticality safety is assured.
- Decay heat is being removed from pebbles outside of the core and long-term cooling is assured, where figure of merit temperatures are steadily decreasing.

This narrative captures the limiting PHSS break event of this postulated event category. Other PHSS break events grouped in this category include:

- A transfer line break when pebbles are getting inserted into empty core
- A transfer line break when pebbles are getting inserted into the core at power
- A transfer line break when pebbles are getting transferred to storage canisters
- A mishandling of fuel outside the reactor (e.g., containment box, at the material balance areas and key measure points)

The PHSS break event when pebbles are extracted from the core is considered bounding among the grouped events because the spilled pebbles have higher temperatures and burnups, therefore, the highest decay heat and MAR loading compared to other events in the group.

In order to ensure that the design features mitigating a PHSS break event are sufficient to keep the consequences bounded by the MHA, the following key figures of merit must be evaluated:

- Peak TRISO temperature ex-vessel to limit diffusion of radionuclides
- Mobilized Flibe and graphite dust released
- Peak TRISO temperature in vessel to limit diffusion of radionuclides
- ~~Peak TRISO temperature~~TRISO failure probability to limit incremental TRISO layer failures
- Peak Flibe-cover gas interfacial temperature to limit evaporation mass transfer of radionuclides
- Peak vessel and core barrel temperatures to prevent vessel failure and maintain long term cooling
- Mass loss of pebble carbon matrix due to oxidation to limit tritium release and prevent additional release of materials at risk
- Mass loss of structural graphite due to oxidation to limit tritium release
- Peak temperature of structural graphite to limit the tritium release
- Peak temperature of pebble carbon matrix to limit the amount of tritium release

3.2.2.6 Radioactive Release from a Subsystem or Component

An external hazard event causes a failure of components not protected from the hazard to fail and release MAR stored in these systems. These systems include:

- Tritium management system
- Inert gas system
- Chemistry control system

Figures of merit method can significantly reduce analysis cost since the dose of the same release pathway can be bounded by one bounding case.

Figures of merit for the postulated event must be demonstrated to meet acceptance criteria derived from the MHA conditions. The figures of merit for each postulated event are developed based on the release pathways of radionuclides during the event. The acceptance criteria for figures of merit are developed to ensure the radionuclide releases from the postulated events through the same pathways as the MHA are less than those from the MHA. Therefore, if the acceptance criteria for all figures of merit for a postulated event are met, the dose of the postulated event is bounded by the MHA.

For the postulated events with additional release pathways that do not exist in the MHA, the third method is used. This method has three steps:

1. Bounding doses are calculated for each release pathway; bounding dose for each release pathway is then used to derive acceptance criteria for figures of merit according to the bounding release pathway conditions for the postulated event.
2. For each specific postulated event, if figures of merit for the involved release pathways meet acceptance criteria, the corresponding bounding dose values for the pathways can be used instead of direct dose analysis. Direct dose analysis for certain release pathways can also be performed.
3. All the dose values for each release pathway for the postulated event are summed to compare with the MHA total release dose. The total dose for the postulated event must be lower than the MHA dose.

As an example, for the figures of merit method (i.e., second method), during a core transient, radionuclides diffuse through the TRISO fuel layers as a function of temperature. Radionuclides in Flibe evaporate from the Flibe-cover gas interface as a function of temperature. Tritium desorbs from the graphite and pebble carbon matrix. Therefore, the peak TRISO temperature-time, peak Flibe-cover gas interfacial temperature, peak graphite temperature and peak pebble carbon matrix temperature profiles during the event are figures of merit for a postulated event that involves the core. Additionally, ~~peak TRISO temperature~~TRISO failure probability is also a figure of merit to limit incremental fuel failure to a negligible level during the transient; peak vessel and core barrel temperatures are key figure of merit to ensure the reactor vessel performs its safety function.

The figures of merit used for systems code analysis (KP-SAM) are a surrogate for demonstrating that consequences are bounded by MHA doses, or for maintaining a coolable geometry. However, if dose is the figure of merit for an event (i.e., a dose analysis is performed for the event), then those surrogate figures of merit for dose do not need to meet acceptance criteria, because the dose acceptance criterion is being explicitly evaluated. Likewise, when a figure of merit has been analyzed separately for bounding conditions (e.g., a structural analysis of the vessel is performed separately from the systems analysis) then that figure of merit does not need to be analyzed in the systems code to meet an acceptance criterion.

The figures of merit derived for each postulated event and the associated acceptance criteria are provided in Table 3-2.

3.4.2.1 Peak TRISO Temperature-Time

The release pathway for fuel is diffusional release as a function of temperature. During a postulated event, peak TRISO temperature is bounded by temperature-time curve derived from the assumed MHA fuel temperature-time curve to limit diffusion of radionuclides to less than the amount during the MHA. Bounding temperature-time curve derived from the assumed MHA temperature-time curve can be based on integrated effects on dose.

3.4.2.2 ~~Peak~~ TRISO Failure Probability ~~temperature~~

Based on TRISO fuel qualification efforts as described in (Reference 26), it is expected that d~~D~~uring a postulated event, incremental failure of TRISO fuel is limited to a negligible level if the peak temperature is below 1600°C. Failure probability of TRISO fuel can increase due to overpressure in the TRISO particles, which is a function temperature. The failure probability of TRISO fuel is evaluated using the methodology described in Section 4.2. ~~Incremental failure is demonstrated to be negligible for peak fuel temperatures up to 1600°C.~~

~~Alternatively, using the methodology described in Section 4.2, the incremental fuel failure during the postulated event is calculated using the assumed MHA fuel temperature profiles and is demonstrated to be negligible. If the peak TRISO temperature during a postulated event is bounded by the assumed MHA fuel temperature-time curve, incremental failure fuel failure is demonstrated to be negligible.~~

3.4.2.3 Peak Flibe-cover gas interfacial temperatures

Radionuclide release from Flibe is through evaporation. During a postulated event, peak Flibe-cover gas interfacial temperature is bounded by temperature-time curve derived from the assumed MHA Flibe-cover gas interfacial temperature-time curve to limit evaporation mass transfer of radionuclides to less than the amount during the MHA. Bounding temperature-time curve derived from the assumed MHA temperature-time curve can be based on integrated effects on dose.

3.4.2.4 Peak vessel and core barrel temperature

To prevent vessel failure and maintain long term cooling during a postulated event, the peak vessel and core barrel temperatures must be less than both (a) a maximum allowable temperature derived to limit excessive creep deformation and damage accumulation and (b) 816°C. The maximum allowable temperature is calculated so that the creep strain induced by primary membrane stresses within the vessel and the core barrel does not exceed 1% at the end of reactor life. Its derivation relies on the following assumptions:

- All regions of the vessel and core barrel in contact with Flibe are exposed to temperatures lower than or equal to 650°C for the hot operating time of the vessel and temperatures lower than or equal to the vessel and core barrel peak temperatures for a maximum duration of 360 hours (15 days).
- The maximum primary stresses undergone by the vessel and core barrel can be bounded by a maximum stress value derived as described in the evaluation model for structural integrity.

3.4.2.5 Minimum reactor vessel inner surface temperature

During the long-term cooling phase of a postulated event when decay heat is being removed passively by the DHRS, freezing must be avoided within the downcomer to preserve the natural circulation characteristics in the vessel that allow for uninterrupted decay heat removal. A conservative treatment is to limit the reactor vessel inner surface temperature to always higher than the Flibe freezing temperature of 459°C. If this condition is met, no freezing happens within the downcomer.

3.4.2.6 Airborne release fraction of spilled/splashed Flibe

During a salt spill event, aerosols can be generated through jet breakup, and spilling and splashing. The airborne release fractions due to aerosolization must be limited so that the dose consequences of the salt spill events are bounded by the MHA.

3.4.2.7 Volatile products from Flibe chemical reactions

Flibe could be exposed to air during a salt spill event. The key release pathway of radionuclide from Flibe is through evaporation, which is a function of vapor pressure of the radionuclide species. When Flibe is exposed to air, the Flibe-air chemical reaction does not result in excessive reactive vaporization which would form radionuclide chemical species that have a higher vapor pressure than those already exists in Flibe circulating activity. It is expected that a few specific RN chemical species will have a higher vapor pressure after reacting with air than those in the circulating activity. However, those species are expected to be present at very low concentrations and the resulting difference in evaporation rate will be of minimal significance. For example, CsF dissolved in Flibe does not react with air to form a highly volatile cesium hydroxide. As such, Flibe-air reaction does not result in significant additional release of radionuclides from Flibe through evaporation.

The reactor cell floor is assumed to be designed to preclude Flibe-concrete reaction. When Flibe is spilled, it has the potential to come in contact with stainless steel and insulation material. Flibe interactions with stainless steel and insulation do not result in formation of radionuclide chemical species that have a higher vapor pressure than those already exists in Flibe circulating activity. Therefore, Flibe-stainless steel and Flibe-insulation reactions in the Hermes design basis do not result in additional release of radionuclides from Flibe through evaporation.

During a salt spill event, Flibe is not exposed to water, and therefore no Flibe-water reaction need to be considered. However, if a common cause failure (e.g., seismic) causes a water-containing SSC and Flibe-containing SSC to fail concurrently, the amount of water that Flibe could be exposed to is assumed to be limited to an upper bound limit by design. When interacting with this upper bound amount of water, Flibe redox potential is still maintained within the bounds of salt chemistry conditions defined for the evaporation model; therefore, does not result in additional release of radionuclides from Flibe through evaporation.

During a postulated event that involves the primary heat exchanger (PHX), Flibe could mix with nitrate salt and react chemically. The volatile products formed from Flibe and nitrate mixing are addressed in Section 4.5.

3.4.2.8 Mass loss of structural graphite and pebble carbon matrix

Pebbles and structural graphite not submerged in Flibe can oxidize when exposed to air. If the mass loss of the pebble carbon matrix does not extend to the fueled zone, tritium release is the only additional MAR release pathway to be considered when fuel pebble oxidizes. Tritium is puff released from oxidized pebble carbon matrix and oxidized structural graphite. In the MHA analysis, the assumed temperature for pebble carbon matrix is so high that all available tritium is effectively puff-released from the pebble carbon matrix. The portion of structural graphite unsubmerged in Flibe is small. The inventory of tritium puff released (instead of as a function of temperature) from oxidization of structural graphite not submerged in Flibe is accommodated by the following inherent conservatism in the treatment of tritium in the MHA:

- Conservative inventory of tritium available for release
- Conservatively high assumed temperature of pebbles

the volume branch, valve, pump, tank, point kinetic models, thermal radiation, and gap conductance models.

The widely used point kinetics equations model for multiple groups of delayed neutron precursors was implemented in KP-SAM with fully implicit time scheme options available up to 5th order accuracy. The decay heat power can be calculated from the user provided decay curve, or the ANSI/ANS-5.1-2005 standard method. Whichever method is used, uncertainty factors will be applied to ensure it is conservative. ~~or from the decay heat model based on a standard decay heat curve.~~ For the predictive decay heat model, the fissile material fission fractions include U-235, U-238, Pu-239, and Pu-241 and are provided by the reactor core design calculation. The fission ratios of fissile materials are provided for various stages of operation (build up). A sensitivity factor can also be applied to the decay heat fraction in order to conservatively account for uncertainties in decay heat.

Closure relations are correlations and equations that help to model the terms in the field equations by providing code capability to model and scale particular processes. Typical closure models include wall friction factor and form loss models for different flow geometries, convective heat transfer correlations for different heat transfer surfaces and pump performance curves. Fluid and solid properties, including equations of state are also needed to close the field equations. The fluids to be simulated include Flibe, intermediate salt, water, simulant oil, air, and argon gas.

Table 4-1 summarizes the models and the field equations used by KP-SAM.

4.1.1.2 Control System Description

The SAM control system is used to perform the evaluation of algebraic and simple ordinary differential equations; the trip system is used to perform the evaluation of logical statements. The fundamental approximation made in the design of control/trip system is that the execution of control/trip system is de-coupled from the other parts of the hydraulic systems. The main execution of individual control/trip units is set at the end of each time step.

4.1.1.3 Numerical Methods

SAM uses a continuous finite element methods formulation for the spatial discretization of the 1-D or 2-D field equations. The detailed discretization for both time and space is managed by MOOSE, with the code formulated such that the numerical method orders are controlled through user inputs. For fluid models, a spatial stabilization method is required to suppress checkerboard type spatial oscillations that manifest when solving advection dominated problems using continuous finite element methods. The Stream Line-Upwind/Petrov-Galerkin and the Pressure-Stabilizing/Petrov-Galerkin scheme are implemented in SAM to resolve the numerical instability issues (Reference 10).

The physics in SAM is integrated into a single fully coupled nonlinear equation system. The discretized nonlinear equation system is solved using a pre-conditioned Jacobian Free Newton Krylov method. The combination of the Jacobian Free Newton Krylov nonlinear solver and high order numerical methods for both time and space enables the capability to minimize numerical errors.

4.1.1.4 Quality Assurance and Configuration Control

The software quality assurance plan is designed to provide a framework for solving computational engineering problems. The software quality assurance plan includes roles and responsibilities for the software developer, reviewer, tester, and user as well as documentation and software review requirements. The software quality assurance plan also describes configuration management, change control, audit requirements, software engineering methods, standards, practices, conventions, and

- Compromised TRISO $d_T + (1 - d_I - d_S - d_O - d_T) \times f_T$

Where d_I, d_S, d_O, and d_T are the defective fractions of the IPyC layer, SiC layer, OPyC layer, and TRISO particle (i.e., exposed kernel), respectively, while f_I (cracked IPyC), f_{IS} (cracked IPyC + failed SiC), f_S (failed SiC), and f_T (failed TRISO) are the in-service failure fractions for the TRISO fuel failure modes.

Radionuclide release is calculated for each of the intact and five compromised states and the overall radionuclide release from the population of TRISO particles is obtained by weighting the resulting release fractions by the probabilities of occurrence of these states. Dispersed uranium is assumed to be fully released from the TRISO particles and its contribution is added to the release from the intact and compromised particles.

The verification and validation plans for the KP-BISON code are summarized in Reference 7.

4.3 NEUTRONICS

The Serpent2 code is used for neutronics calculations. The Star-CCM+ code is used for both discrete element modeling of the pebble flow and porous media approximation for thermal-hydraulics feedback. The description of these tools and models along with validation, verification, and uncertainties are presented in Reference 8.

4.4 STRUCTURAL ANALYSIS

The materials qualification plan for high temperature metallic materials is provided in Reference 9. The materials qualification plan for graphite materials is provided in Reference 11. These qualification plans inform the figures of merit for the reactor vessel and internals described in this report. The structural analysis of the materials under postulated event conditions will be performed prior to submittal of an Operating License Application.

4.5 EVENT-SPECIFIC METHODS

This section provides the event-specific methods that use the evaluation models with conservative inputs to analyze the transients discussed in Section 3. Key model uncertainties and initial conditions are conservatively applied to the methods to ensure figures of merit are conservatively predicted. Sample results for the postulated event categories are provided in Appendix A to illustrate the transient methodologies.

4.5.1 Salt Spills

The salt spill event category is described in Section 3.2.2. The analysis of the bounding salt spill event is composed of the following models:

- Single phase break flow model – the mass flow rate with time through the break and the final upper plenum free surface level are the two major modeling results. Two-phase flow due to gas entrainment is prevented through the primary pump design. Two modeling options are available: (a) KP-SAM model based on the slight modification of the baseline plant model to include the single-phase break flow model; and (b) a conservative analytical model
- Long term performance of passive decay heat removal model – this is similar as the model used for loss of forced circulation overheating bounding case but with reduced free surface level.
- Radioactive source term release models to estimate the bounding total release from the event. Two major source term models are required:

where E is an entrainment coefficient. A conservatively high ~~low~~ value of E is 2.1E-6 (Reference 19). The aerosol generation due to the spilling and splashing is then obtained through the Flibe spilling rate and Equations 16 and 17.

MAR release associated with the aerosol generation is evaluated through the aerosol amount and the concentration of MAR in the spilled Flibe.

Evaporative Release from Spilled Flibe

The evaporative release is the phase when the discharge of the Flibe from the vessel ends and the spilled Flibe completes spreading on the reactor cell floor. Small amount of Flibe is likely to spread only a fraction of the reactor cell floor area before it is completely solidified. It is not a major concern for MAR release for partially spreading Flibe because it freezes quickly. More concern is large amount of spilled Flibe which spreads the entire area of the reactor cell floor. In this case, a Flibe pool is expected to form with a depth of molten Flibe. The bottom of the pool contacts with steel liner which is placed to prevent Flibe-concrete interaction. The top of the pool transfers heat to air through convection and to surrounding structures through radiation. No water and no water sources are present where the Flibe spreads, and Flibe-water interaction is excluded.

MAR release from the Flibe pool is dominated by evaporation over the top surface of the pool. It continues until the top surface is solidified. To evaluate the amount of MAR released, Flibe temperatures are evaluated first. The Flibe temperature is based on energy balance of the pool. For the downward heat transfer, a layer of solidified Flibe is expected between the liquid Flibe and the liner. A 1D moving boundary equation needs to be solved for the temperature profile within the solidified layer, and growth (or shrinkage) of the layer. The boundary condition at the interface between the liquid Flibe and the solidified layer is determined by Globe-Dropkin correlation (Reference 20). The boundary condition at the interface between the solidified layer and the underneath liner is given by gap conductance between the solidified layer and the liner, or through continuity conditions of temperature and heat flux if no gap is assumed. The heat transfer between the liquid Flibe to the top surface is determined by Globe-Dropkin correlation again, and the heat transfer on the air side is based on McAdams correlation (Reference 21) for natural convection and radiation with a low temperature heat structure. These heat transfer terms are combined to determine the energy change of the liquid Flibe due to heat transfer and solidification at the bottom, and eventually the temperatures of the liquid Flibe and at the top surface.

Once the temperatures are determined, evaporation rates are assessed with the same method as the MHA for MAR. The evaporation rate and integral release amount are evaluated until the temperature of the top surface is lower than the Flibe melting temperature.

4.5.2 Insertion of Excess Reactivity

The limiting insertion of excess reactivity is described in Section 3.2.2. The analysis of the limiting event in this category (a control element withdrawal) includes a systems analysis with conservative neutronics and fuel performance input.

4.5.2.1 Initial Conditions

The initial conditions of the transient are biased to ensure a conservative evaluation of the figures of merit. The limiting control rod withdrawal scenario is assumed to initiate from the highest possible reactor power because the higher power provides the highest heat input to challenge the identified figures of merit. However, sensitivities must be performed to ensure that reactivity insertions from lower power levels do not unexpectedly challenge a figure of merit. A power uncertainty is applied to reactor power to cover uncertainties associated with detection and signal delays. Since the reactor power is biased high in the assumed limiting reactivity insertion event, the initial reactor power is modeled at 102% power. Additional initial condition values are provided in Table 4-4.

4.5.2.2 Transient Analysis Methods

The reactivity insertion transient involves a change in core reactivity that adds heat to primary system. Therefore, the event analysis requires information from the systems code, fuel performance, and neutronics EMs. The systems code, KP-SAM analyzes the event progression with inputs from the neutronics EM and provides inputs to the fuel performance EM.

The nuclear fission power profile within the pebble bed is affected by the neutron flux distribution in the core region and the fuel burn-up status of the pebbles. The current approach to modeling core power density is an axially resolved radially averaged method and does not explicitly account for radial power peaking in the core. The radial power profile and its effect on the coolant and fuel temperature are not explicitly modeled; therefore, local peak coolant and fuel temperatures are not fully resolved. The hot channel factor methodology described in Section 4.1 accounts for both power peaking and the possibility of flow being poorly distributed in the core.

~~The KP-SAM base model in Section 4.1 is used with modifications to the reactor core model. The nuclear fission power profile within the pebble bed is affected by the neutron flux distribution in the core region and the fuel burn-up status of the pebbles. With a single channel modeling of the core zone, the axial power profile can be defined by providing the power-shape-function in the KP-SAM code input deck. The radial power profile and its effect on the coolant and fuel temperatures are not explicitly modeled, however, because the single channel model uses the average power at each axial level. In order to address the radial power distribution and model its effects on the coolant and fuel temperature, especially to capture their maximum values, a separate core channel representing high radial power is analyzed as a hot channel. Consequently, the core is modeled as two channels, i.e., an average channel and a hot channel. The hot channel model assumes complete thermal isolation from the adjacent average channel. In reality, however, since there is no physical distinction between the two channels, some thermal-hydraulic interactions are expected. The isolation assumption, therefore, would predict higher fuel and coolant temperatures in the hot channel, resulting in more conservative predictions. The hot channel flow area is set to be small enough to represent the radial high-power zone. A core flow rate corresponding to the area is assigned to the hot channel.~~

In order to ensure a conservative evaluation of the limiting reactivity insertion event, the following conservatisms are applied to model inputs:

- Highest worth control element is assumed to be withdrawn.
  - The limiting reactivity insertion rate is determined from the limiting reactivity rod worth per length from neutronics EM, combined with the maximum control element withdrawal speed.
  - A range of reactivity insertion rates, depending on the control element control design, is analyzed in the final safety analysis to ensure that the highest reactivity insertion rate is identified that bounds the reactivity insertion rates possible for other events in the category.
  - At full power and hot zero power, the initial control element position is assumed to be fully inserted in the reactor core.

19. M. Epstein, and M. Plys, "Prediction of Aerosol Source Terms for Dose Site Facility Applications," Proceedings of EFCOG Safety Analysis Working Group, April-May 2006.
20. S. Globe, and D. Dropkin, *Natural-Convection Heat Transfer in Liquids Confined by Two Horizontal Plates and Heated from Below*, J. Heat Transfer, Vol 81, No. 1, pp. 24-28, February 1959.
21. W. McAdams, *Heat Transmission, 3rd Edition*, McGraw-Hill, New York, p. 180, 1954.
22. K. Jafarpur, Yovanovich, M.M., *Laminar Free Convective Heat Transfer from Isothermal Spheres: a New Analytical Method*, Int. J. Heat Mass Transfer, Vol 35, No. 9, pp. 2195-2201, 1992.
23. M. F. Young, "Liftoff Model for MELCOR," SAND2015-6119, Sandia National Lab, 2015.
24. S. Levy, *Two-Phase Flow in Complex Systems,* 1st Edition, Wiley-Interscience, August 1999.
25. S.K. Friedlander, *Smoke, Dust, and Haze: Fundamentals of Aerosol Dynamics*, 2nd Ed., Oxford University Press, 2000.
26. Electric Power Research Institute, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," Topical Report EPRI-AR(NP)-A, 3002019978, November 2020.

Table 3-2: Derived Figures of Merit and Acceptance Criteria for Postulated Events

Figure of Merit	Acceptance Criterion	Applicable Events
Peak TRISO temperature-time	Generally bounded by temperature-time curves derived from the assumed MHA fuel temperature-time curve	Salt Spills, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break, Seismic, PHX Tube Break
~~Peak TRISO temperature-time~~TRISO failure probability	~~Below incremental~~Negligible TRISO fuel failure ~~temperature~~ probability	Salt Spills, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break, PHX Tube Break
Peak Flibe-cover gas interfacial temperature	Generally bounded by temperature-time curves derived from the assumed MHA Flibe-cover gas interfacial temperature-time curve	Salt Spills, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break, PHX Tube Break
Peak vessel and core barrel temperatures	Bounded by both the maximum allowable temperature derived to limit excessive creep deformation and damage accumulation and by 816°C (highest temperature considered by ASME Section III Division 5 for 316H)	Salt Spills, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break, PHX Tube Break
Minimum reactor vessel inner surface temperature	Above Flibe melting temperature	Loss of Forced Circulation
Airborne release fraction of spilled/splashed Flibe	Below airborne release fraction limit derived to bound total releases of the postulated event to less than the MHA	Salt Spills, Seismic, PHX Tube Break
Volatile product formation from Flibe-air reaction	Negligible amount of additional volatile products formed	Salt Spills, PHSS break, PHX Tube Break
Volatile product formation from Flibe chemical reaction with water, concrete, and/or construction materials (e.g., insulation, steel)	Negligible amount of additional volatile products formed	Salt Spill
Volatile product formation from Flibe chemical reaction with nitrate	Negligible amount of additional volatile products formed	PHX Tube Break
Mass loss of pebble carbon matrix due to oxidation	Mass loss does not extend into the fueled zone	Salt Spills, PHSS break
Mass loss of structural graphite due to oxidation	Bounded by the MHA release	Salt Spills, PHSS break