Enclosure 1

Changes to Postulated Event Analysis Methodology Technical Report (KP-TR-018)

(Non-Proprietary)

- 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. Parameter ranges considered for all events are provided in Table 4-4. 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:
  - Aerosol generation rate and amount due to single phase coolant jet.

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 bias the power high. ~~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 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, up to and including the maximum reactivity insertion rate, ~~depending on the control element control design, is~~ are 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.
  - A conservative treatment is applied to address the impact of a dynamic change in power shape associated with the control element movement.
- Least negative reactivity feedback coefficients are used to minimize the power suppression effect by the negative reactivity feedback in preliminary safety analysis.
- Most negative reactivity feedback coefficients are also be applied and analyzed to investigate the effect of delayed reactor trip in the final safety analysis.

This event is also identified as one of the bounding fuel performance cases and must be analyzed with the KP-BISON using the methodology described in Section 4.2.

4.5.3 Loss of Forced Circulation

The limiting loss of forced circulation scenario is described in Section 3.2.2. The analysis of the limiting event in this category includes a systems analysis with conservative neutronics input.

4.5.3.1 Initial Conditions

The initial conditions of the transient are biased to ensure a conservative evaluation of the figures of merit. The limiting loss of forced circulation 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 loss of forced circulation events from lower power levels do not unexpectedly challenge a figure of merit. ~~Initial condition values are provided in Table 4-5.~~

4.5.3.2 Transient Analysis Methods

The important thermal and hydraulic phenomena during the transient include the flow friction (negative head) at the pump, heat transfer between the coolant and various interfacing structures such as pebble, reactor vessel wall and internals. Because the forced circulation is lost, the fluid friction through the coolant loop, including the reactor core, is more important than other events where forced flow is maintained.

KP-SAM is used to analyze the event progression with inputs from the neutronics EM and provides inputs to the structural integrity EM. Upon a loss of forced circulation, the reactor experiences an immediate increase in the fuel (pebble) temperature because of the reduced heat transfer to the coolant. The coolant temperature also rises because heat removal from the reactor core to the PHX is reduced and eventually stops. The increased temperature of the coolant could challenge the integrity of reactor vessel and core barrel structures.

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 temperatures 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 described in Section 4.1 is used to analyze a loss of forced circulation event with the following modifications:

- Typically, the interaction between the fluid system and pump, during the transient, is modeled using head and torque curves of the pump. For the loss of forced circulation analysis, the coolant flow response is modeled without the detailed pump characteristics, by conservatively assuming the pump head after the transient starts. Since the pump rotor is assumed to stop instantly, the pump torque information is not needed.
- The reactivity feedback effect on power is minimized for conservative calculation by using least negative reactivity coefficient values to minimize the effect of power reduction from the initial temperature increase by the reduced coolant flow.
- The uncertainties in material properties of the Flibe coolant and vessel structures are addressed conservatively. The thermal mass of the material is reduced such that the temperatures of fuel and

Table 4-4: Input Parameters Considered for Postulated Events

Parameter	Value	Rationale
Reactor initial power	Range of values up to and including maximum power level including uncertainty	Ranges of power levels analyzed
Coolant average temperature	Range over controller deadband and measurement uncertainty	Limiting value may be event dependent
System pressure	Nominal for all events except for salt spill	The effect of the system pressure is insignificant for all events except for salt spill events
Power distribution	Axial + radial power distribution for peaking factor Both fresh core and equilibrium core are considered as limiting conditions	Most limiting power distribution is considered
Shutdown margin	Considers most reactive shutdown rod is unavailable	Provide margin for malfunctions
Shutdown rod insertion time	Conservative shutdown rod insertion times assumed	Delays the shutdown of the reactor
Reactivity coefficients	Values assumed on an event specific basis and account for uncertainty	Limiting values may be event dependent
DHRS Capacity	Minimum and maximum performance assumed on an event specific basis Minimum performance assumes loss of a train of DHRS and minimum performance requirements Maximum performance assumes full capacity of DHRS plus uncertainty	Minimum DHRS performance is expected to be bounding for heatup events Maximum DHRS performance is expected to be bounding for overcooling events
Decay heat	Minimum and maximum values assumed on an event specific basis	Maximizing decay heat is expected to be bounding for heatup events Minimizing decay heat is expected to be bounding for overcooling events
Material properties	Ranged within uncertainties	Uncertainty in material properties for coolant and

		structures treated on an event specific basis
Reactor Protection System analytical limits	Actuation on: - High Reactor Power - High Flux Rate - High Coolant Temperature - Low Level	Analytical limits provide margin to safety limits Measurement uncertainty applied to setpoints are derived from analytical limits
Reactor Protection System actuation delay	Conservative delay times applied	Delay reactor trip
Plant Control Systems	Potential event mitigation capabilities of the plant control systems are not credited Suitably conservative treatment of relevant plant control features is applied in the safety analysis	Plant control systems are not safety related Potentially adverse performance of plant control systems needs to be considered

~~**Table 4-4: Initial conditions for Insertion of Excess Reactivity**~~

~~Parameter~~	~~Initial Condition~~	~~Rationale~~	~~Note~~
~~Reactor Initial power~~	~~[102%]~~	~~Potential power meter uncertainty~~	~~Modeled explicitly~~
~~Coolant average temperature~~	~~Nominal + 3%°C~~	~~Controller deadband and measurement uncertainties~~	~~Modeled explicitly~~
~~System pressure~~	~~Nominal~~	~~The effect of the system pressure is insignificant~~	~~Not modeled~~
~~Power distribution~~	~~Axial + radial power distribution for peaking factor~~ ~~Both fresh core, and equilibrium core are considered as limiting conditions~~	~~Most liming power distribution is considered~~	~~The axial radially averaged power profile is modeled explicitly in KP-SAM. Radial peaking and uncertainties are handled via hot channel factors~~

~~Table 4-5: Initial conditions for Loss of Forced Circulation Overheating Bounding Event~~

~~Parameter~~	~~Initial Condition~~	~~Rationale~~	~~Note~~
~~Reactor Initial power~~	~~102%~~	~~Potential power meter uncertainty~~	~~Modeled explicitly~~
~~Coolant average temperature~~	~~Nominal + 3%°C~~	~~Controller deadband and measurement uncertainties~~	~~Modeled explicitly~~
~~System pressure~~	~~Nominal~~	~~The effect of the system pressure is insignificant~~	~~Not modeled~~
~~Power distribution~~	~~Axial + radial power distribution for peaking factor~~ ~~Both fresh core, and equilibrium core are considered as limiting conditions~~	~~Most liming power distribution are considered~~	~~The axial radially averaged power profile is modeled explicitly in KP-SAM. Radial peaking and uncertainties are handled via hot channel factors~~
~~DHRS capacity~~	~~75%~~	~~Assume one DHRS train is out of operation~~	~~Modeled explicitly by reducing radiation view factor~~
~~Heat structure heat capacity~~	~~75%~~	~~Account for any uncertainty related to the heat capacity of solid materials in the model~~	~~Modeled explicitly by applying a scale factor to solid material heat capacities~~
~~Flibe heat capacity~~	~~95%~~	~~Account for uncertainty in the heat capacity of Flibe~~	~~Modeled explicitly by applying a scale factor to Flibe heat capacity~~
~~Reactivity coefficient magnitude~~	~~75%~~	~~Reduced to conservatively bias the impact of reactivity feedback prior to reactor trip~~	~~Modeled explicitly~~

APPENDIX A. SAMPLE TRANSIENT RESULTS

A.1 Insertion of Excess Reactivity

Event Description

A control element with 3.02$ reactivity worth is assumed to be withdrawn completely over 100 seconds. The rate of reactivity insertion depends on a worth curve and the progression of the rod withdrawal. When the power level exceeds the trip setpoint, -16.8$ of reactivity is inserted to the core over 10 seconds according to an element worth curve. After 10 seconds, this reactivity is maintained, simulating the total assumed element worth. The assumptions made are summarized as below and initial conditions are provided in Table A1-1.

Power trip setpoint = 120%

Upper plenum temperature trip setpoint = 958.1K (665°C + 3%)

Power trip delay time = 2s

Temperature trip delay time = 2s

Element insertion delay after trip = 2s

Time to fully insert rods after trip = 10s

Element worth = 16.8$

Primary salt pump halving time = 2s

Intermediate velocity halving time = 1s

KP-SAM analysis results

The transient is initiated at 0 seconds with the start of reactivity insertion. Prior to a reactor trip, this positive reactivity insertion is counteracted in part by negative Doppler, moderator, and coolant feedback respectively in order of magnitude. Soon after reactor trip is initiated, the total change in reactivity of the system becomes negative and remains so despite the continuation of the reactivity insertion, as shown in Figure A1-1

When the reactor trip is initiated, the PSP is tripped as well, causing a decrease in flow rate throughout the system. This has notable impacts on heat transfer throughout the system during the entire simulation, as this will characterizes flow behavior in the core during earlier stages of the transient and facilitate the transition to natural circulation in the long term.

KP-SAM Conclusions

A reactivity insertion of 3.02$ over 100 seconds was assumed to simulate an uncontrolled control element withdrawal. The reactor is tripped by a high flux protection signal (120%) at 9 seconds after the event initiation. Figure A1-2 shows key predicted temperatures relative to the temperature used in the MHA analysis. The temperature rises in the TRISO and fuel matrix were observed, with very little change in the Flibe temperature. The resulting temperatures, with the exception of reflector temperatures, are within the acceptance level, with significant margins. The short deviation (i.e., on the order of a few minutes) of the reflector temperature slightly above the MHA temperature is acceptable due to the time-at-temperature nature of diffusion of tritium out of graphite grains.

Fuel Performance Analysis

The power and temperature profiles were used as inputs to KP-BISON. The transient is modeled at the end of a normal operation phase that provides the adequate state of the TRISO fuel particles (e.g., failure fractions, fission product distribution, fission gas inventory, etc.).

The normal operation phase is modeled using the irradiation conditions shown in Table A1-2~~1~~.

Table A1-3~~2~~ shows the failure probabilities calculated by KP-BISON within the Monte Carlo calculation scheme for the TRISO failure modes for normal operation and reactivity insertion event. The results in Table A1-3~~2~~ indicate that the temperature during normal operation and transient is not high enough to challenge the TRISO fuel with overpressure or Pd attack. Furthermore, Table A1-3~~2~~ shows that the reactivity insertion event does not lead to any significant incremental failure.

Table A1-1: Initial conditions for Insertion of Excess Reactivity Assumed Bounding Event

Parameter	Initial Condition	Rationale
Reactor initial power	102%	Assumed power measurement uncertainty
Coolant average temperature	Nominal + 3%°C	Controller deadband and measurement uncertainties
System pressure	Nominal	The effect of the system pressure is insignificant
Power distribution	Axial + radial power distribution for peaking factor Both fresh core, and equilibrium core are considered as limiting conditions	Most limiting power distribution is considered

Table A1-2~~1~~: 95% Confidence Level Upper Limit on In-Service Failure Fractions for Normal Operation and Reactivity Insertion Postulated Event

Failure Probability	Normal Operation	Normal Operation + Reactivity Insertion
Probability of IPyC cracking	9.75×10^{-1}	9.75×10^{-1}
Probability of SiC failure	2.26×10^{-3}	2.26×10^{-3}
Contribution due to palladium penetration	3.00×10^{-6}	3.00×10^{-6}
Contribution due to IPyC cracking	2.26×10^{-3}	2.26×10^{-3}
Probability of TRISO failure	3.00×10^{-6}	3.00×10^{-6}

Table A1-3~~2~~: Compromised Fractions for Normal Operation and Reactivity Insertion Postulated Event

Release Fraction	**Normal Operation**	**Normal Operation + Reactivity Insertion**
Intact	$2.25x10^{-2}$	$2.25x10^{-2}$
Compromised IPyC	$9.65x10^{-1}$	$9.65x10^{-1}$
Compromised IPyC +SiC	$2.24x10^{-3}$	$2.24x10^{-3}$
Compromised SiC	$1.03x10^{-4}$	$1.03x10^{-4}$
Compromised OPyC	$1.00x10^{-2}$	$1.00x10^{-2}$
Compromised IPyC + SiC + OPyC	$5.30x10^{-5}$	$5.30x10^{-5}$

A.4 Loss of Forced Circulation

Event Description

The purpose of this event is to determine if the reactor is adequately designed for long term heat up events. As such, one of the key assumptions is that only 75% of DHRS capacity is available. The loss of forced circulation overheating bounding event applied to the plant model is initiated by manually tripping the pump and reducing the head to zero nearly instantaneously. The complete loss of flow defines the beginning of the transient and occurs concurrently with a loss of intermediate coolant flow. Intermediate coolant flow is not likely to be lost during a loss of forced circulation event but is imposed in this analysis to demonstrate that intermediate coolant flow is not needed to protect the plant during a loss of forced circulation event. During this transient, it is expected that the large reduction in coolant flow through the core region results in a significant rise in temperature across the core. The rise in temperature eventually causes the reactor to trip, leading to a long-term cooling transient and the safe shutdown condition of the reactor. Initial conditions for the overheating loss of forced circulation assumed bounding event are provided in Table A4-1. A set of assumptions key to this analysis are listed in Table A4-4~~2~~.

The loss of forced circulation transient was run over the course of 72 hours of simulation time. During the transient, the upper plenum temperature exceeds the trip setpoint after 23 seconds, with rod insertion following a trip delay. Prior to the rod insertion, power is reduced by reactivity feedback as the core heats up, afterwards the strong insertion of negative reactivity from the rod insertion brings the reactor power down to decay heat levels. Figure A4-1 shows key predicted temperatures relative to the temperature used in the MHA analysis.

The compromised fractions for the six states are obtained from the defect and in-service failure fractions in Table 4-3 (see Section 4.2) and Table A4-5~~3~~. These are shown in Table A4-6~~4~~, assuming the upper specification or bounding values.

Loss of Forced Circulation Overheating

A loss of forced circulation transient biased for overheating was performed using KP-SAM. In this simulation, it was demonstrated that decay heat removal through the DHRS can compensate for the loss of the intermediate salt flow to achieve stable cooling after the fast stage of the transient.

The TRISO temperature profile is bounded by the MHA curve, which demonstrates that the diffusional release of radionuclides from fuel is bounded by the MHA. The Flibe-cover gas interfacial temperature profile is bounded by the MHA curve, which demonstrates that the release from Flibe through evaporation is also bounded by the MHA.

The graphite reflector and fuel pebble temperature profiles are bounded by the MHA curves, which demonstrates that the tritium release is bounded by the MHA. It is shown that temperatures stay below those defined by the MHA except for the upper plenum and reflector/graphite temperatures. The MHA release analysis is conservative. The MHA margin is maintained since deviations are minimal and of short duration (as scaled relative to the corresponding X/Q window associated with the deviation) due to the conservative evaporative boundary conditions in the MHA (i.e., aggressive temperature gradients driving natural circulation) and times associated with those temperatures corresponding in the MHA (i.e., evaporation and diffusion are time-at-temperature release mechanisms). Freezing does not occur in this event and that the vessel remains below the defined temperature limit.

The power and temperature profiles were used as inputs to KP-BISON. The transient is modeled at the end of a normal operation phase that provides the adequate state of the TRISO fuel particles (e.g., failure fractions, fission product distribution, fission gas inventory, etc.). The normal operation phase is modeled using the irradiation conditions shown in Table A4-3~~1~~.

The failure probabilities associated with the potential failure modes listed in Section 4.2 were obtained by a Monte Carlo simulation of 106 samples. Note: the sample size was chosen to optimize computing time. From the Monte Carlo simulation results, upper limits on the failure probabilities associated with each failure modes are obtained at a 95% confidence level using the Copper-Pearson exact method. These limits are reported in Table A4-5~~3~~ for the normal operation and loss of forced circulation postulated event.

The results in Table A4-5~~3~~ indicate that the temperatures during normal operation and the transient are not high enough to challenge the TRISO fuel with overpressure or Pd attack. In particular, the upper limit on TRISO failure by overpressure is only a few percent (6%) of the as-manufactured exposed kernel fraction of 5.0×10^{-5}. Furthermore, Table A4-5~~3~~ shows that the TRISO fuel is more likely to fail during normal operation and that the loss of forced circulation event does not lead to any significant incremental failure. Because of the conservative assumptions used to set up the low- and high-temperature trajectories, the calculated failure probabilities are also conservative and represent upper limits for expected failure probabilities.

Loss of Forced Circulation Overcooling

While the overheating version of this event is designed to challenge the margin to maximum temperatures, the overcooling scenario is designed to challenge the margin to minimum temperatures. In this case the limiting minimum temperature is taken as the point at which Flibe freezes. In order to conservatively preclude freezing, the minimum vessel inner surface temperature is taken as a bounding surrogate for the minimum Flibe temperature. The event is initiated by manually initiating a control rod insertion, primary pump trip and intermediate flow trip at t = 0. The primary pump and intermediate flow are allowed to coast down normally. Additionally, the DHRS is modeled at 100% capacity. Initial conditions for the loss of forced circulation overcooling event are provided in Table A4-2. A set of key assumptions for ~~The input parameters assumed in~~ the example calculation is~~are~~ provided in Table A4-4~~2~~.

The example calculation of a cooldown biased loss of forced circulation transient was run over the course of 72 hours of simulation time. Figure A4-2 shows key predicted temperatures relative to the temperature used in the MHA analysis. Temperatures predicted by the KP-SAM model are below the temperatures defined by the MHA and freezing does not occur within 72 hours.

Table A4-1: Initial Conditions for Loss of Forced Circulation Overheating Assumed Bounding Event

Parameter	**Initial Condition**	**Rationale**
Reactor initial power	102%	Assumed power measurement uncertainty
Coolant average temperature	Nominal + 3%°C	Controller deadband and measurement uncertainties
System pressure	Nominal	The effect of the system pressure is insignificant
Power distribution	Axial + radial power distribution for peaking factor Both fresh core, and equilibrium core are considered as limiting conditions	Most limiting power distribution is considered
DHRS capacity	75%	Assume one DHRS train is out of operation
Heat structure heat capacity	75%	Account for any uncertainty related to the heat capacity of solid materials in the model
Flibe heat capacity	95%	Account for uncertainty in the heat capacity of Flibe
Reactivity coefficient magnitude	75%	Reduced to conservatively bias the impact of reactivity feedback prior to reactor trip

Table A4-2: Initial Conditions for Loss of Forced Circulation Overcooling Assumed Bounding Event

Parameter	Initial Condition	Rationale
Reactor initial power	98%	Assumed power measurement uncertainty Minimized stored energy
Coolant average temperature	Nominal - 3%°C	Controller deadband and measurement uncertainties
System pressure	Nominal	The effect of the system pressure is insignificant
Power distribution	Axial + radial power distribution for peaking factor Both fresh core, and equilibrium core are considered as limiting conditions	Most limiting power distribution is considered
DHRS capacity	100%	Full capacity of DHRS
Heat structure heat capacity	75%	Account for any uncertainty related to the heat capacity of solid materials in the model Minimizes stored energy and accelerates cooldown
Flibe heat capacity	95%	Account for uncertainty in the heat capacity of Flibe Minimizes stored energy and accelerates cooldown
Reactivity coefficient magnitude	Nominal	Reactor trip initiated immediately following event initiation

Table A4-3~~1~~: Irradiation Conditions for Simulated Normal Operation of Hermes

Parameter	Value
Irradiation length (EFPD)	300
Power density (fission/m^3 s	5.7 x 10^{19}
Burnup (%FIMA)	6.0
Fast flux (n/m^2 s, E > 0.1 MeV)	7.7 x 10^{17}
Fast fluence (n/m^2 s, E > 0.1 MeV)	2.0 x 10^{25}

Table A4-4~~2~~: Inputs for Loss of Forced Circulation Postulated Events

Loss of Forced Circulation – Overheating		**Loss of Forced Circulation - Overcooling**	
Parameter	Value	Parameter	Value
Temperature trip delay time (s)	2	Time to fully insert rods after trip (s)	10
Element insertion delay after trip (s)	2		
Time to fully insert rods after trip (s)	10	Trip delay after event initiation (μs)	20
Trip worth ($ of reactivity)	16.8	Trip worth ($ of reactivity)	16.8
Primary salt pump halving time (pump seizure approximation) (s)	0.01	Primary salt pump halving time (s)	2
Intermediate velocity halving time (s)	1	Intermediate velocity halving time (s)	1
DHRS capacity (%)	75	DHRS capacity (%)	100

Table A4-5~~3~~: 95% Confidence Level Upper Limits on In-Service Failure Fractions for Normal Operation and Loss of Forced Circulation Postulated Events

Failure Probability	Normal Operation	Normal Operation + Loss of Forced Circulation
IPyC Cracking	9.75×10^{-1}	9.75×10^{-1}
SiC Failure	2.26×10^{-3}	2.26×10^{-3}
Contribution due to palladium penetration	3.00×10^{-6}	3.00×10^{-6}
Contribution due to IPyC cracking	2.26×10^{-3}	2.26×10^{-3}
TRISO Failure	3.00×10^{-6}	3.00×10^{-6}

Table A4-64: Compromised Fractions for Normal Operation and Loss of Forced Circulation Postulated Event

Release Fraction	Normal Operation	Normal Operation + Loss of Forced Circulation
Intact	2.25×10^{-2}	2.25×10^{-2}
Compromised IPyC	9.65×10^{-1}	9.65×10^{-1}
Compromised IPyC + SiC	2.24×10^{-3}	2.24×10^{-3}
Compromised SiC	1.03×10^{-4}	1.03×10^{-4}
Compromised OPyC	1.00×10^{-2}	1.00×10^{-2}
Compromised IPyC + SiC + OPyC	5.30×10^{-5}	5.30×10^{-5}

