Probabilistic Risk Assessment of the Molten Chloride Reactor Experiment Conceptual Design

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Abstract: A probabilistic risk assessment (PRA) is being developed for the Molten Chloride Reactor Experiment (MCRE). The approach chosen for developing the analyses to support the MCRE safety basis is based on the Licensing Modernization Project's risk-informed performance-based approach documented in NEI 18-04 of which PRA is an integral part. The internal events PRA was developed to support the MCRE Conceptual Design that concluded in April 2022. The PRA has been applied to select the safety basis events (SBEs) and the preliminary results have been plotted against frequency-consequence targets based on relevant Department of Energy (DOE) regulatory limits. A comparison of the preliminary frequency and dose results of MCRE SBEs indicates significant margin to the DOE expected frequency-consequence targets.

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1. INTRODUCTION

A probabilistic risk assessment (PRA) is being developed for the Molten Chloride Reactor Experiment (MCRE) [1]. The Department of Energy (DOE) awarded a contract to Southern Company Services (SCS) Inc. for the design, development, and operation of the MCRE under the Advanced Reactor Demonstration Program. SCS leads the project team including TerraPower (TP), Idaho National Laboratory (INL), Orano Federal Services, CorePower, 3M Company and the Electric Power Research Institute. It is proposed to site the MCRE at the National Reactor Innovation Center's Laboratory for Operation and Testing in the United States (LOTUS) facility. The final operating location at INL will be determined following a National Environmental Protection Agency review. The PRA is being developed to support the request for DOE approval of the MCRE safety basis.

2. MCRE CONCEPTUAL DESIGN DESCRIPTION

The MCRE is planned to be a 200-kW fast fission reactor which will provide key reactor physics data to support the design and licensing of a commercial Molten Chloride Fast Reactor (MCFR). It is a first of a kind for the world as the two historical molten salt reactors (MSRs) in the USA namely the Oak Ridge National Laboratory (ORNL) Molten Salt Reactor Experiment (MSRE) and the Aircraft Reactor Test (ART) had been thermal fission reactors.

The MCRE fuel will be a sodium chloride and highly enriched uranium trichloride ($NaCl$-UCl_3) eutectic. The mean fuel temperature will be 650 °C and the normal operating pressure will be 100 kPa, thus avoiding the underlying phenomena that lead to the type of pressure vessels needed by light water reactors (LWRs). The MCRE design employs the concept of functional containment instead of a pressure retaining containment – the proposed INL testbed is planned to be a filtered and vented confinement structure. The fuel salt has a very high heat capacity, and in the event of a reactor trip (RT), the bulk freezing temperature will only be approached about 40 hours later (with neither decay heat nor the ability to add heat to the salt). The current design does not envision freezing in the reactor vessel (below 523 °C) – only in the fuel salt drain tank.

The reactor has a very strong negative fuel density reactivity feedback coefficient which contributes to the inherent safety as overheating or overpower transients could be arrested without RT. Subcriticality could be accomplished by control elements or ultimately by core offload into the fuel salt drain tank that has a subcritical geometry where the fuel could be allowed to freeze.

Fuel handling will be accomplished pneumatically using argon. A rapid fuel offload is planned in the design for station blackout (SBO). This design will rely on automatic "fail safe" (argon and offgas) valve alignments to establish the necessary delta-P (±340 kPa) across the reactor vessel head space and drain tank. Unlike the MSRE design, there is not a freeze valve in the fuel drain line that needs to be thawed to allow fuel offload (there is a freeze valve in the flush salt tank drain line).

Decay heat removal will be accomplished by passive conduction through the vessel or drain tank walls and convection into the LOTUS atmosphere. Active decay heat removal will not be required.

The MCRE general arrangement is shown in Figure 1 (with the shielding and LOTUS confinement structure cut away). The reactor and fuel salt drain tank are located inside the shielding. There are two primary cooling trains circulating cooled nitrogen through the reactor enclosure system heat exchanger which consists of an annulus around the reactor. The hot nitrogen will be cooled by cooling coils that circulates Dowtherm-Q cooling fluid through an air-cooled heat exchanger outside. Delta T will be 10 °C. The cover gas system filters and tanks are to the front left. The fuel cask loading glove box and the flush salt tank are to the left rear. The argon supply tank is to the right rear.

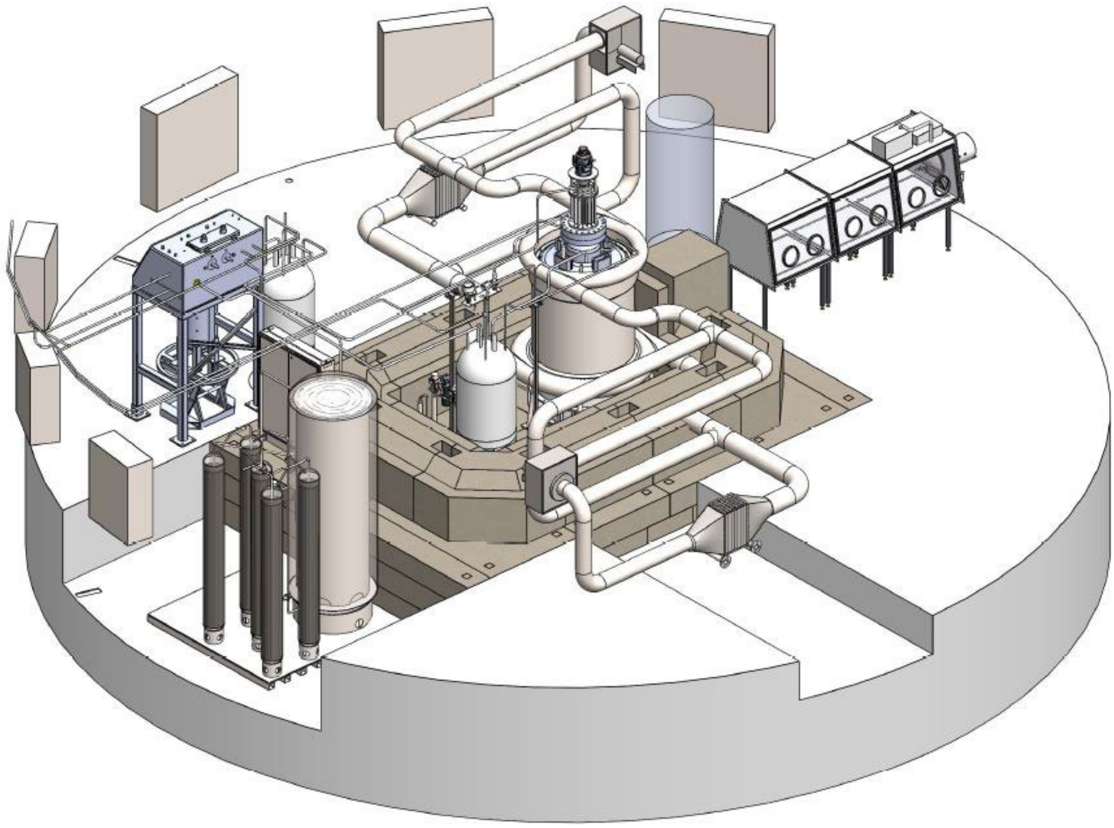

Figure 1: The MCRE General Arrangement in the LOTUS testbed

Not shown in the figure above is the switchyard that includes a standby diesel generator and provisions for hooking up a portable/mobile diesel generator.

3. PRA APPROACH

The approach chosen for developing the analyses to support the MCRE safety basis is based on the Licensing Modernization Project's (LMP) risk-informed performance-based (RIPB) approach for Nuclear Regulatory Commission (NRC) - licensed plants documented in NEI 18-04 [2] and endorsed in R. G. 1.233 [3]. This approach uses PRA to identify licensing basis events (LBEs) from which design basis accidents (DBAs) are selected and provides a systematic way for the safety classifications of systems, structures, and components (SSCs). Central to this approach is the use of frequency-consequence (F-C) targets that evolved from Farmer's Criterion [4]. The LMP F-C target from NEI 18-04 is shown in Figure 1. The LBEs are to be plotted on this target as (dose, frequency) coordinates representing risk for comparison against dose limits. It should be noted that certain factors, including defense-in-depth adequacy and accounting for uncertainty, means that the F-C targets are not acceptance criteria for meeting the applicable nuclear safety regulations.

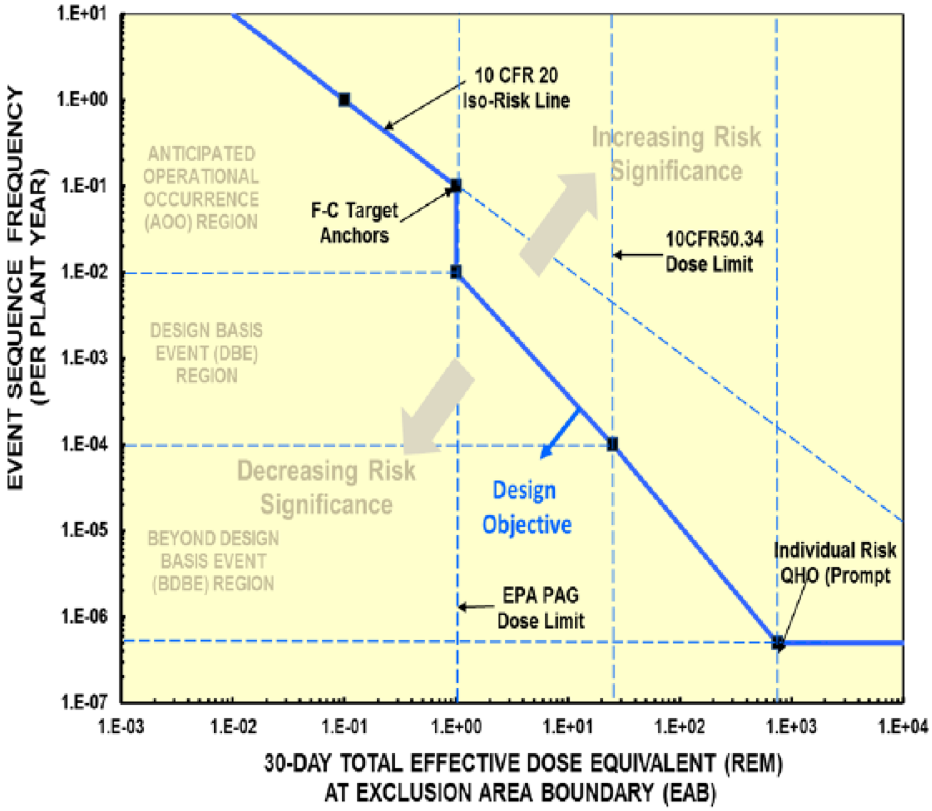

Figure 2: LMP Frequency-Consequence Target, @NEI 2019, All rights reserved

Because the MCRE safety basis is to be approved by the DOE, the PRA will use the DOE radiological F-C guidelines [5] shown in Table 1. Note that in DOE terminology LBEs are referred to as safety basis events (SBEs) and "licensing" is referred to as "authorization".

Table 1: MCRE Radiological Consequence Guidelines

SBE Category	Frequency Range [yr^{-1}]	TEDE [REM]	
		Offsite	Onsite
Anticipated	f ≥ 1E-2	< 5	< 5
Unlikely	1E-2 > f ≥ 1E-4	< 5	< 25
Extremely Unlikely	1E-04 > f ≥ 1E-06	< 25	< 100
Beyond Extremely Unlikely	f < 1-06	No criteria	No criteria

To support a RIPB authorization approach, a PRA is being developed in conjunction with plant design. As the DOE PRA Standard [7] is not prescriptive with regards to PRA requirements but incorporates them by reference, the PRA is to be developed using applicable requirements of the ASME/ANS PRA Standard [8] as a framework. As the MCRE will have a relatively small radionuclide inventory, few SSCs, and inherently safe responses to initiating events, not all requirements of the ASME/ANS PRA Standard are deemed applicable. For example, mechanistic source term analysis will not be required given the small inventory and expected low maximum unmitigated dose. The internal events analysis is being developed in accordance with applicable ASME/ANS PRA Standard supporting requirements for the following PRA elements: plant operating states, initiating events, event sequence analysis, success criteria, data analysis, system analysis, human reliability analysis, event sequence quantification, dose consequence analysis, and risk integration

Other internal and external hazards may be developed using a simplified approach of estimating the frequencies and assuming the maximum unmitigated dose as the consequence, which will obviate the need for detailed hazards and fragilities analyses, unless the F-C dose targets are challenged.

4. PRA DEVLOPEMENT

4.1. Plant Operation States

The plant operation states (POSs) are correlated with the operation modes (OMs) as shown in Table 2. The annual duration fractions (ADFs) are calculated from the planned duration in each POS.

Table 2: MCRE Plant Operation States

POS	OM	OM Description	Duration [hrs.]	ADF
POS-1	OM-1	Power Operation	1000	1.14E-01
POS-2	OM-2	Startup	1200	1.37E-01
POS-3	OM-3	Hot Standby	1200	1.37E-01
POS-4a	OM-4	Fuel Load/ Unload (Fuel in RCS)	250	2.85E-02
POS-4b		Fuel Load/ Unload (Fuel in drain tank)		
POS-5	OM-5	Cold Shutdown	2500	2.85E-01

4.2. Initiating Events

Initiating events have been identified from the ART hazards analysis [9], ORNL MSRE unusual operating conditions [10], ORNL MSRE reactor safety analysis [11], a master logic diagram developed at TP, a recent MSR workshop publication [12], and a recent journal article [13]. These initiating events have been screened for applicability to MCRE. The initiating events have been categorized into the following hazard groups:

- Decrease in fuel salt flow rate
- Increase in fuel salt flow rate
- Decrease in heat removal
- Increase in heat removal
- Reactivity and power distribution anomalies
- Fuel handling failures
- Leaks or ruptures
- Loss of offsite power

4.2. Data Analysis

With little operational and component reliability data available for MSRs, available liquid sodium component reliability data [14] have been used for salt-wetted components, as the pressure and temperature operating environments for salt and sodium are similar - in contrast to that of LWRs. For other components, the Savannah River database [15] has been used, in general. A mean loss of offsite power frequency has been obtained from INL/EXT-17-42376 [16].

The generic mean values for the Alpha Factor method from Table 5-11 in NUREG/CR 5485 [17] are applied to develop common cause failure event probabilities. For common cause failure of the control elements to drop into the core, a demand failure probability has been obtained from NUREG/CR-5500 [22].

4.3. Human Reliability Analysis

No misalignment pre-initiator human failure events (HFEs) have been identified yet, as the MCRE does not have standby safety systems that require re-alignment for maintenance or testing. Miscalibration of instrumentation channels are ruled out as bistable setpoints will be continuously monitored by the digital instrumentation and control system, and no calibrations are planned during the MCRE lifetime.

At-initiator HFEs include fuel salt cask drop and cask overfill.

Post-initiators HFEs have been defined for

- Fuel offload failure, as fuel offload (with AC power available) will be by operator action. This offload will probably be by normal shutdown procedures with no appreciable time constraints.
- Failure to align and start a portable diesel generator. This will be a FLEX (diverse and flexible cooing strategies, NEI 12-06) type action (should it be decided to implement this in the design), to back up the standby diesel generator in the event of an extended SBO caused by external events.
- Manual back-up of RT failure. This will be based on a recovery action in the event automatic RT fails. This won't be an important action due to the high reliability of RT.
- Manual back-up of primary coolant system isolation failure. This will be based on a recovery action in the event automatic primary coolant system isolation fails. This won't be an important action due to the high reliability of the primary coolant system isolation.

4.4. Success Criteria

Although a safe, stable state is "a plant condition, following an initiating event, in which plant conditions are controllable at or near desired values" per the ASME/ANS PRA Standard [8]; to satisfy DOE expectations, a safe, stable state also needs to be subcritical. Two safe, stable end states have been defined for the MCRE:

1. The fuel salt is maintained indefinitely inside the vessel in a non-critical liquid state by insertion of the control elements and applying the vessel heaters to add negative reactivity and prevent fuel freezing.
2. The fuel salt is off-loaded from the reactor vessel into the fuel salt drain tank which has a sub-critical geometry and decay heat could be indefinitely dissipated from the tank walls by convection to the LOTUS Testbed atmosphere.

Thermal-hydraulic analyses are in progress to support the development of initiating events, accident sequences and success criteria. A new version of the GOTHIC code has been developed that can model the delayed neutron fraction as a function of flowrate.

4.5. Accident sequences

The MCRE transient event tree is shown in Figure 3. The event tree nodes represent the PRA safety functions.

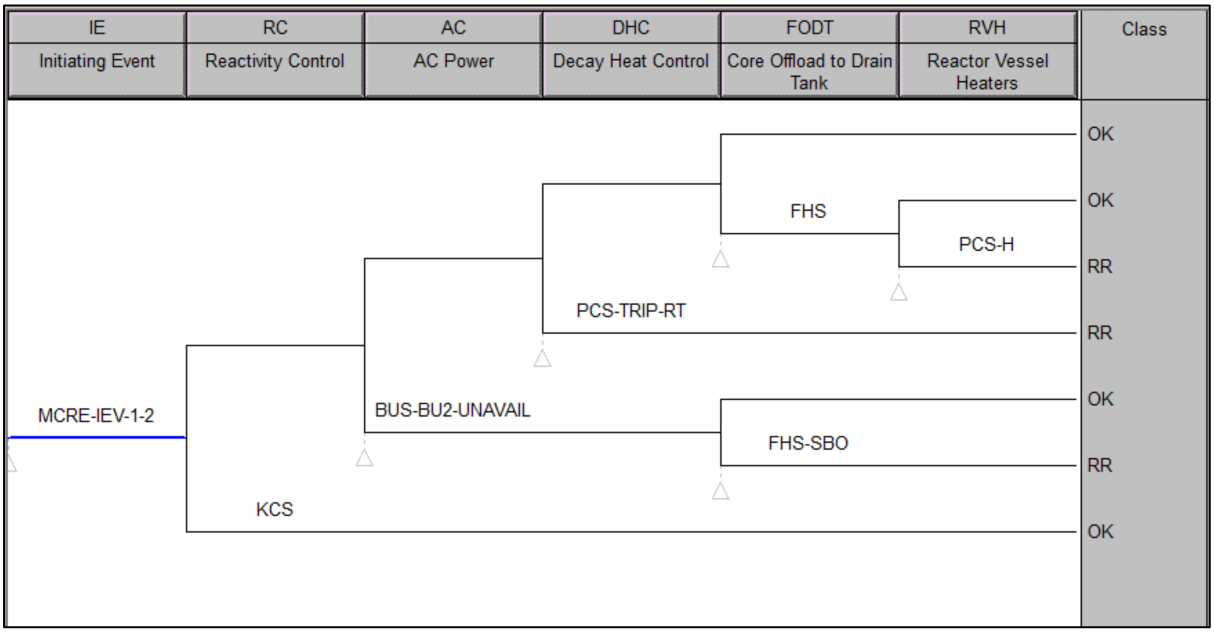

Figure 3:MCRE Transient Event Tree for POS-1

The PRA safety functions are:

- Reactivity control. This will be accomplished by control elements. This function will be supported by the reactor protection system, nuclear instrumentation system, and instrumentation and control system.
- AC Power. Normal AC power will be from offsite power which will be backed up by a standby diesel generator. Provisions will be made for connecting a mobile diesel generator as needed. AC power will only be needed to support hot shutdown by powering the vessel heaters. In the event of SBO, the core will be offloaded automatically.
- Decay heat control. This will be to trip the reactor coolant system blowers to prevent overcooling. Decay heat removal will be by passive dissipation of heat in the structures and convection into the LOTUS atmosphere.
- Core offload. The core will be pneumatically offloaded to the fuel drain tank for cold shutdown. This function will be supported by the offgas system that supplies clean argon to the reactor vessel and vents offgas from the and drain tank to the offgas system.
- Reactor vessel heaters. The core will be maintained inside the reactor vessel and heated to prevent core freezing.

Fault trees have been developed for each PRA safety function.

4.6. Radiological consequences

Preliminary scoping calculations for the MCRE total effective dose equivalent (TEDE) dose consequences have been performed. These calculations consider a leak path factor of 1 and do not credit any confinement in the LOTUS testbed, which is conservative. The accident doses are applied to all SBEs that have been derived from event tree failure sequences. Conservative assumptions include that frozen core end states will lead to releases when, in fact aerosols (the biggest contribution to the dose) will not be released from a frozen core in any significant amounts.

5. FREQUENCY-CONSEQUENCE CHART

The preliminary dose coordinates were plotted on a MCRE offsite F-C chart as shown in Figure 3. The frequencies are 95th percentile frequencies. The preliminary results indicate that the accident scenarios will be below the DOE evaluation guidelines with appropriate controls.

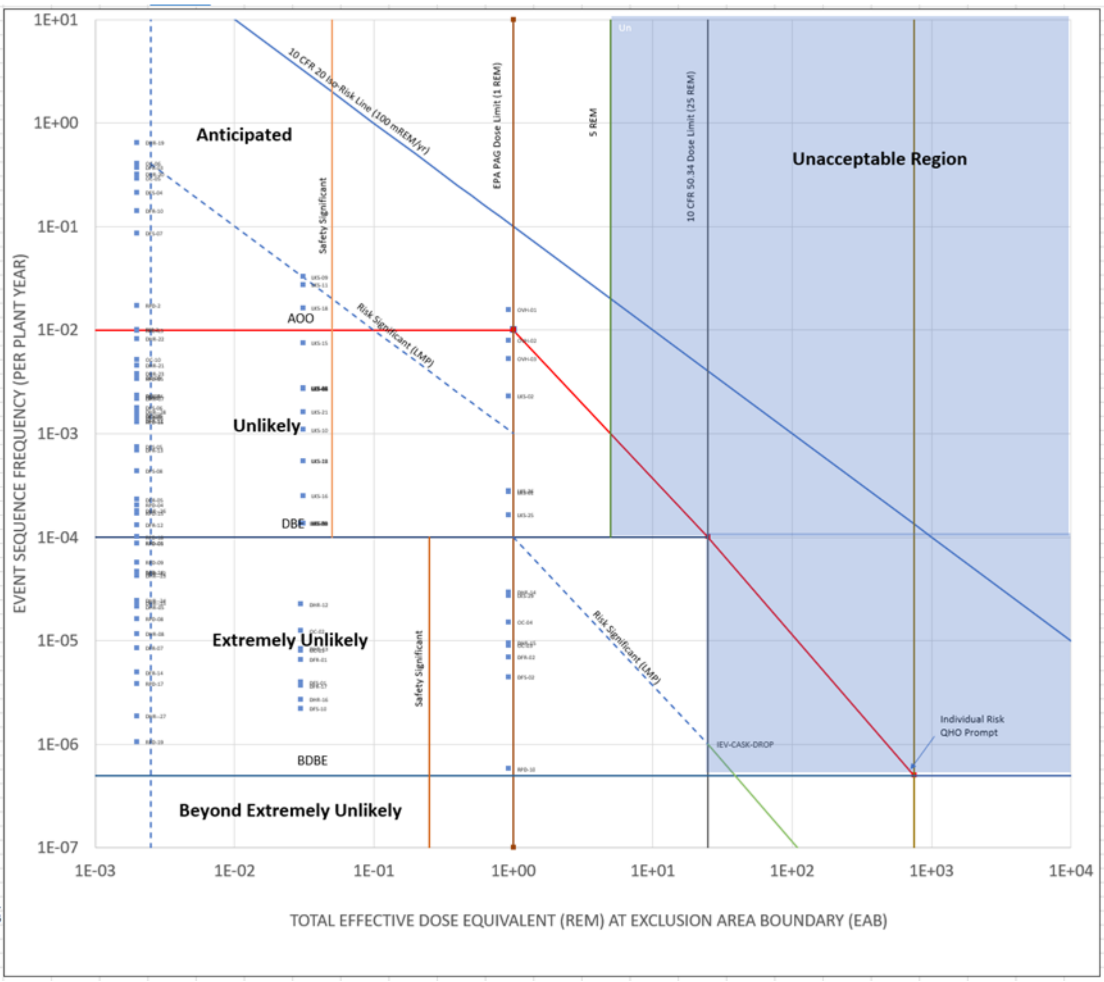

Figure 4: MCRE Offsite Frequency-Consequences

5. CONCLUSIONS

An internal events PRA has been developed to support the request for DOE approval of the MCRE Conceptual Design safety basis. SBEs have been identified and plotted on DOE F-C targets. Final consequence calculations will be performed and plotted on the F-C chart and an SSC safety classification will be performed.

The PRA development will continue to evolve in parallel with subsequent design phases. Further work includes addressing the internal spatial hazards and external hazards and incorporating the dose-consequence analysis and plotting on the F-C plots. A systematic process hazards analysis will also be performed during the preliminary design phase to complement the PRA.

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