

Probabilistic Risk Assessment of the Molten Chloride Reactor Experiment Conceptual Design

Jan Grobbelaar^a, Brian Johnson^b, Zander Mausolf^c, Jonathon Walker^d and Brandon Chisholm^e

^{a, b, c, d} TerraPower, LLC, Bellevue, USA, jgrobbelaar@terrapower.com

^e Southern Company, Birmingham, USA, bmchiso@southernco.com

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 and the preliminary results have been plotted against frequency-consequence targets based on relevant Department of Energy regulatory limits.

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 would 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 would 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 would be a sodium chloride and highly enriched uranium trichloride (NaCl-UCl_3) eutectic. The mean fuel temperature would be 650 °C and the normal operating pressure would 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 LOTUS testbed is planned to be a filtered and vented confinement structure. In the event of a reactor trip (RT) without the ability to add heat to the salt, the bulk freezing temperature would only be approached about 40 hours later. 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 would be accomplished pneumatically using argon. A rapid fuel offload is planned in the design for station blackout (SBO). This design would 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 would be accomplished by passive conduction through the vessel or drain tank walls and convection into the LOTUS atmosphere. Active decay heat removal would not be required.

The MCRE general arrangement is shown in Figure 1 (with the shielding and LOTUS confinement structure cut away). The reactor (1), pump (2) and fuel salt drain tank (3) are located inside the shielding. There are two primary cooling trains (4) circulating cooled nitrogen through the reactor enclosure system heat exchanger which consists of an annulus around the reactor. The hot nitrogen is cooled by cooling coils (10) that circulates Dowtherm-Q cooling fluid through an air-cooled heat exchanger outside the LOTUS. Delta T would be 10 °C. The argon redundant supply tank (5), flush salt drain tank (6), fuel cask loading glove box (7), cover gas system delay tank (8) and carbon beds (9) are outside the shield.

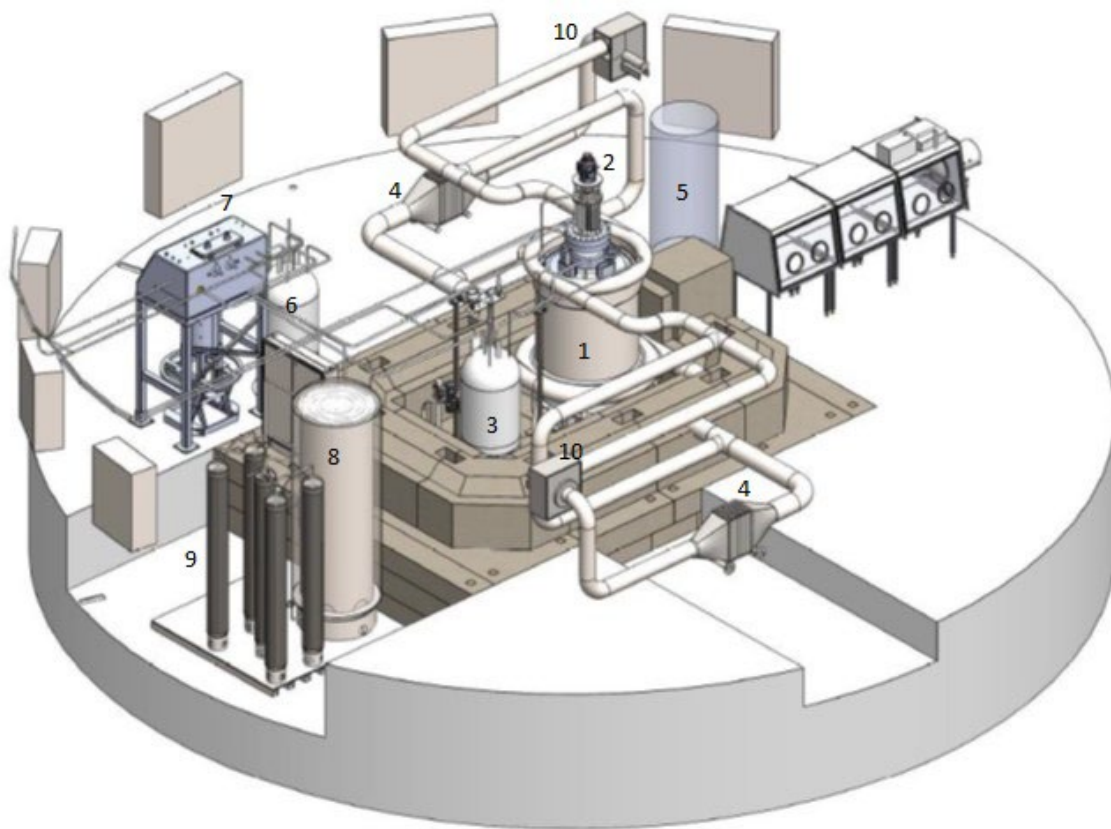


Figure 1: The MCRE General Arrangement in the LOTUS testbed

Not shown in Figure 1 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 [2] is shown in Figure 2. The LBEs are to be plotted on this F-C 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.

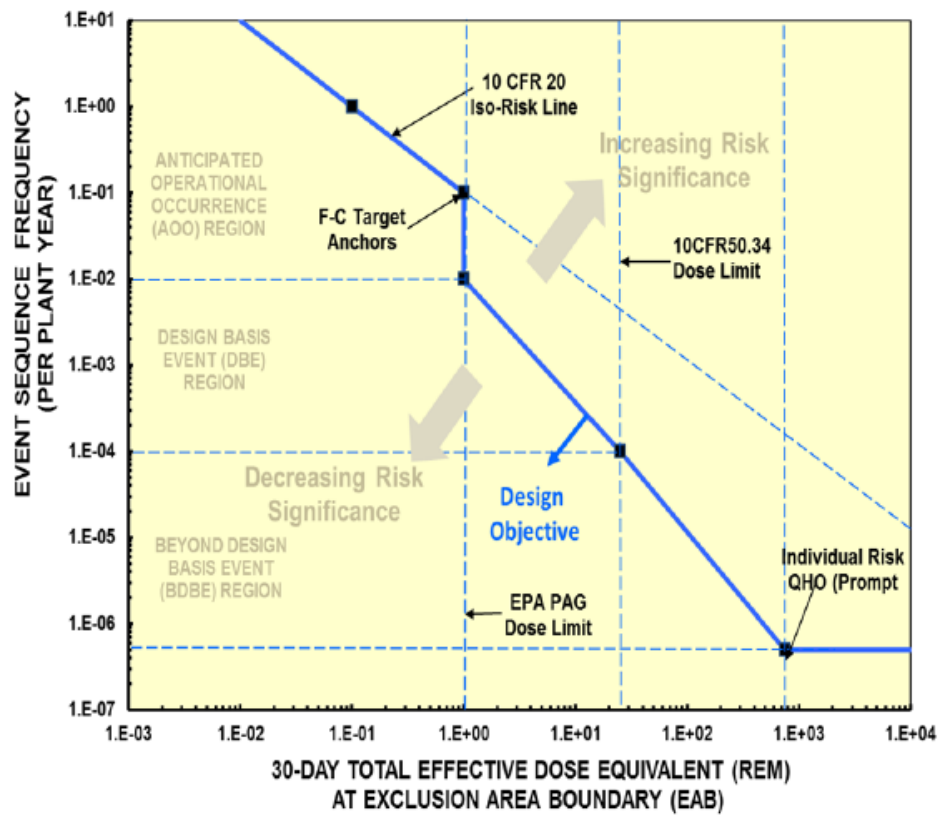


Figure 2: LMP Frequency-Consequence Target

Because the MCRE safety basis will be approved by the DOE, the PRA will use the DOE radiological frequency-consequence (F-C) guidelines [5] shown in Table 1. The dose limits are given in total effective dose equivalent (TEDE). 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	$f \geq 1E-2$	< 5	< 5
Unlikely	$1E-2 > f \geq 1E-4$	< 5	< 25
Extremely Unlikely	$1E-04 > f \geq 1E-06$	< 25	< 100
Beyond Extremely Unlikely	$f < 1E-06$	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 American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard [8] as a framework. As the MCRE would have a relatively small radionuclide inventory, few SSCs, and inherently safe responses to initiating events, not all requirements of the PRA Standard are deemed applicable. For example, mechanistic source term analysis would not be required given the small inventory and expected low maximum unmitigated dose. The internal events analysis would be 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 would obviate the need for detailed hazards and fragilities analyses, unless the F-C dose targets are challenged.

4. PRA DEVELOPEMENT

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 [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 [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 of 1.2E-06 (with EF = 4.18) 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 would 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 overflow.

Post-initiators HFEs have been defined for

- Fuel offload failure, as fuel offload (with AC power available) would be by operator action. This offload would probably be by normal shutdown procedures with no appreciable time constraints.
- Failure to align and start a portable diesel generator. This would be a FLEX (diverse and flexible cooling 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 would 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 would 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 can 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.

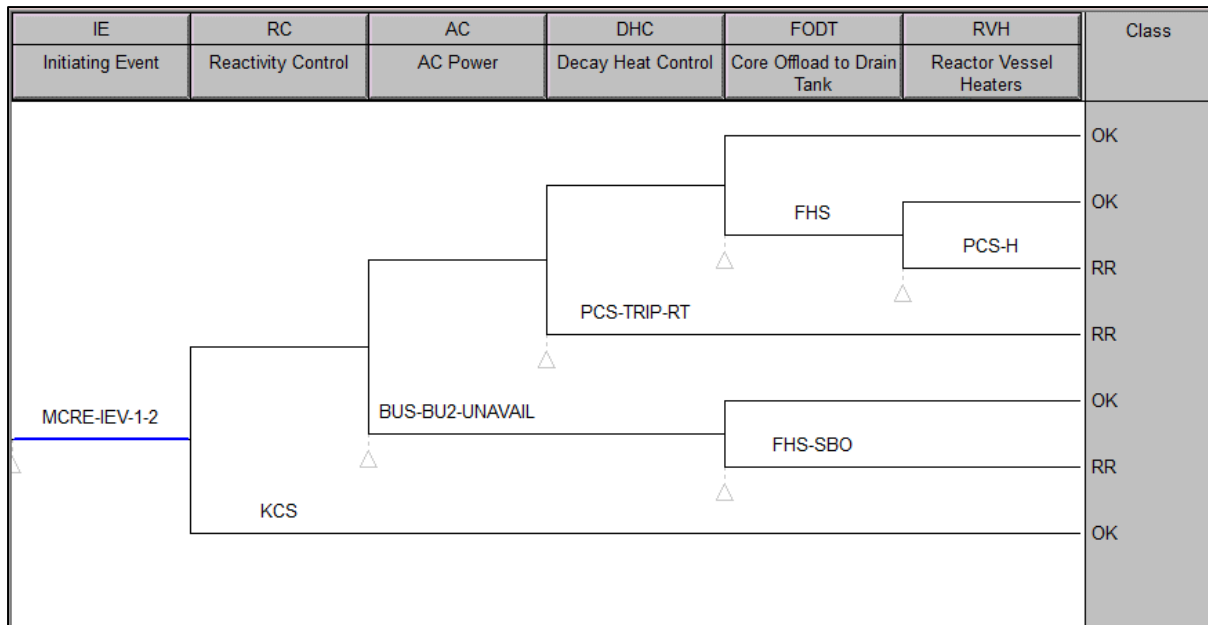


Figure 3: MCRE Transient Event Tree for POS-1

The PRA safety functions are:

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

The MCRE TEDE dose consequences are summarized in Table 3. These are unmitigated doses i.e., a leak path factor of 1 has been used thus not crediting any confinement in the LOTUS testbed, which is very conservative. The bounding accident doses are applied to all SBEs that have been derived from event tree failure sequences. Conservative assumptions include that frozen core end states would lead to releases when, in fact aerosols (the biggest contribution to the dose) would not be released from a frozen core in any significant amounts. The head space dose has only been applied to ruptures in the cover gas space.

Table 3: MCRE Dose Consequences [REM]

Case	Onsite TEDE	Offsite TEDE
Bounding head space (cover gas)	2.7	0.03
Bounding accident (fuel and cover gas)	86	0.9

5. FREQUENCY-CONSEQUENCE CHART

The MCRE Offsite F-C chart is shown in Figure 4. The frequencies are 95th percentile frequencies. No DOE dose limits are exceeded by any SBEs that have been assigned the maximum dose. These results imply that no safety class PRA equipment is needed from the perspective of offsite F-Cs. and some MCRE functions may still be elevated to safety class for defence-in-depth adequacy.

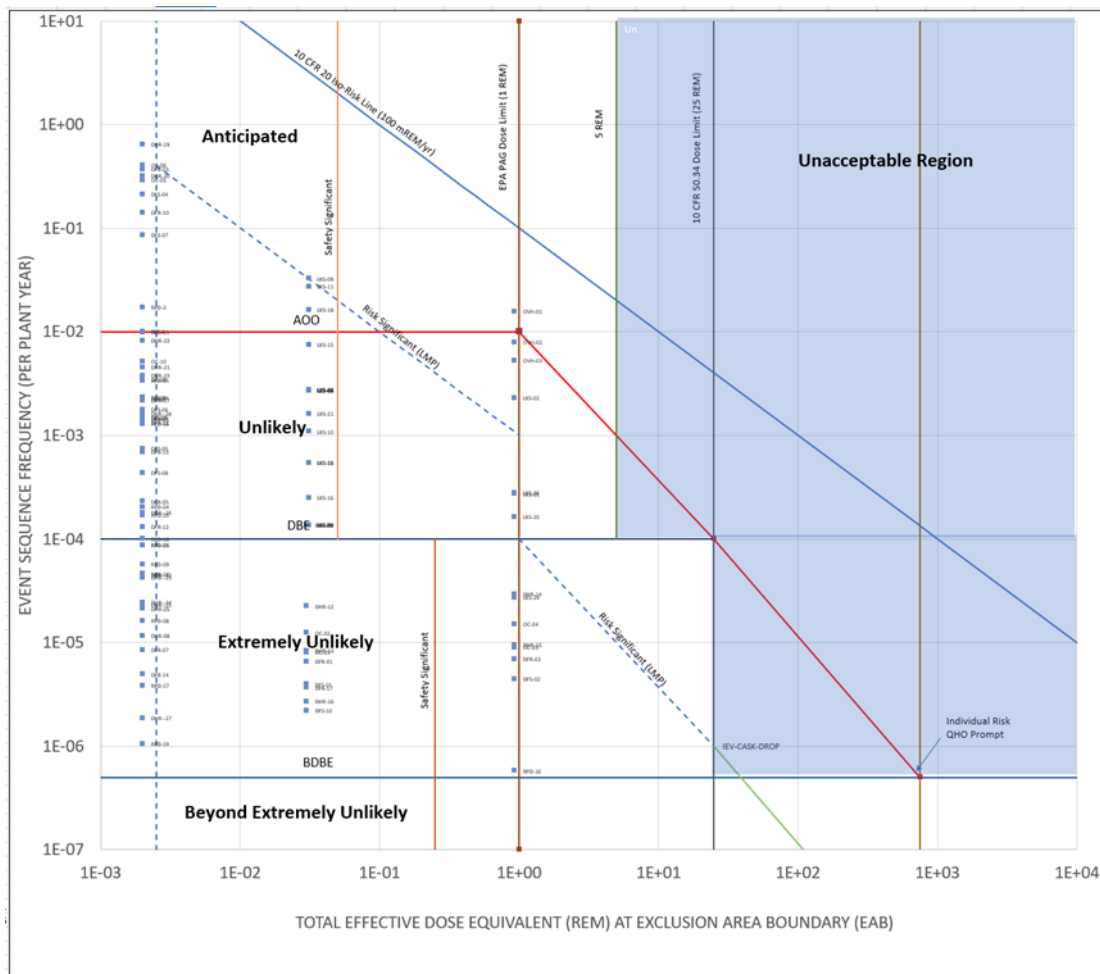


Figure 4: MCRE Offsite Frequency-Consequences

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. Preliminary results imply that safety class equipment would not be necessary from a RIPB point-of-view.

The PRA development will continue to evolve in parallel with subsequent design phases. Further work includes addressing the internal spatial hazards and external PRA 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.

Acknowledgements

The following persons provided review comments:

Troy Reiss, INL
Jason Andrus, INL
Lauren Lathem, Southern Company
Clint Medlock, Southern Company
Daniel Walter, Terrapower, LLC

“This material is based upon work supported by the Department of Energy under Award Number DE-NE0009045.” Disclaimer: “This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favouring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.”

6. REFERENCES

- [1] D. Walter and et al., “Overview of the Molten Chloride Reactor Experiment (MCRE) Mission and Design,” in *2022 ANS Annual Meeting*, Anaheim, CA, 2022.
- [2] Nuclear Energy Institute, “NEI 18-04, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, Revision 1,” 2019.
- [3] U.S. NUCLEAR REGULATORY COMMISSION, “GUIDANCE FOR A TECHNOLOGY-INCLUSIVE, RISK-INFORMED, AND PERFORMANCE-BASED METHODOLOGY TO INFORM THE LICENSING BASIS AND CONTENT OF APPLICATIONS FOR LICENSES CERTIFICATIONS, AND APPROVALS FOR NON-LIGHT-WATER REACTORS,” Washington, DC 20555-0001, January 2020.
- [4] F. R. Farmer, “Reactor Safety and Siting: A Proposed Risk Criterion,” *Nuclear Safety*, vol. 8539, 1967.
- [5] D. Grabaskas and e. al., “Application of the Licensing Modernization Project Approach to the Authorization of the Versatile Test Reactor,” in *2019 American Nuclear Society Winter Meeting*, Washington DC, USA, November 18 - 21, 2019.

- [6] Department of Energy, “DOE-STD-3009-2014, PREPARATION OF NONREACTOR NUCLEAR FACILITY DOCUMENTED SAFETY ANALYSIS,” Washington, DC 20585, November 2014.
- [7] Department of Energy, “DOE-STD-1628-2013, Development of Probabilistic Risk Assessment for Nuclear Safety Applications,” 2013.
- [8] ASME and ANS, “ASME/ANS RA-S-1.4-2021, Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” 2021.
- [9] Oak Ridge National Laboratory, “ORNL-1835, Aircraft Reactor Test: Hazards Summary Report, 1955”.
- [10] Oak Ridge National Laboratory, “ORNL TM-908, Volume II, MSRE DESIGN AND OPERATIONS REPORT PART VIII, OPERATING PROCEDURES,” January 1966.
- [11] Oak Ridge National Laboratory, “ORNL-TM-732, MSRE DESIGN AND OPERATIONS REPORT, Part V, Reactor Safety Analysis Report, August 1964”.
- [12] D. E. Holcomb, A. Huning, A. G. Yigitoglu, M. D. Mulheim, W. P. Poore and G. F. Flanagan, “ORNL/TM-2019/1246, Molten Salt Reactor Initiating Event and Licensing Basis Event Workshop Summary.,” ORNL, 2019.
- [13] B. M. Chisholm, S. L. Krahn and K. N. Fleming, “A systematic approach to identify initiating events and its relationship to Probabilistic Risk Assessment: Demonstrated on the Molten Salt Reactor Experiment,” *Progress in Nuclear Energy*, vol. 129, no. 103507, 2020.
- [14] EG&G IDAHO, INC, Idaho Falls, ID, “EGG-SSRE-8875, The Generic Component Failure Database for light Water and Liquid Sodium Reactor PRA,” 1990.
- [15] Westinghouse Savannah River Company, Aiken, SC, “WSRC-TR-S3-262, SAVANNAH RIVER SITE GENERIC DATA BASE DEVELOPMENT, Rev. 1”.
- [16] Idaho National Laboratory, “INL/EXT-17-42376 Rev. 1, Analysis of Loss-of-Offsite-Power Events, 1987-2016,” August 2017.
- [17] Idaho National Engineering and Environmental Laboratory, “NUREG/CR-5485, Guidelines on Modeling Common Cause Failures in Probabilistic Risk Assessment,” U.S. Nuclear Regulatory Commission, November 1998.