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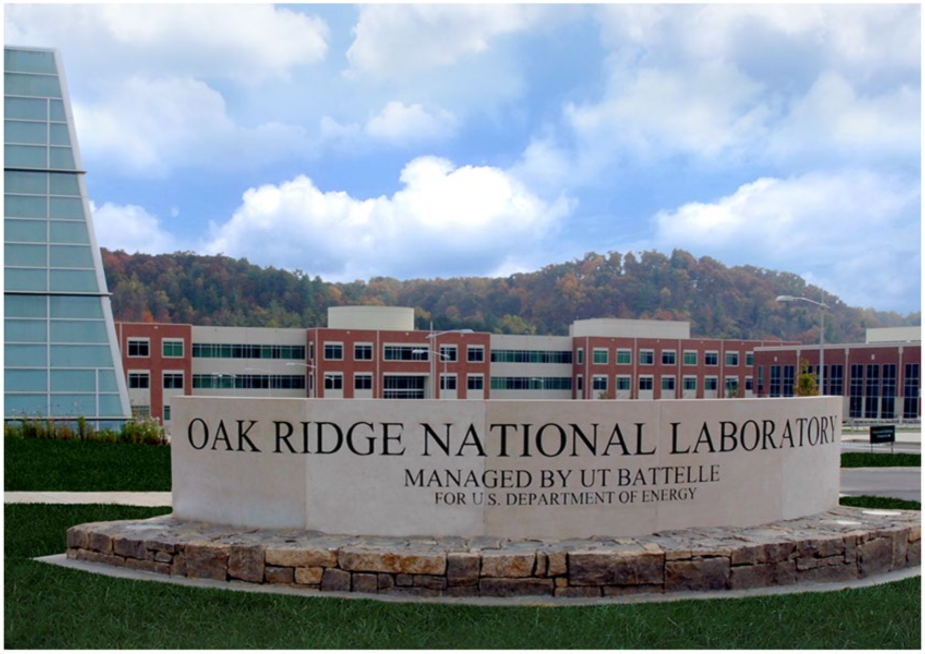
# Molten Salt Reactor Initiating Event and Licensing Basis Event Workshop Summary

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**July 2019**

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OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE US DEPARTMENT OF ENERGY

**ORNL/TM-2019/1246**

Reactors and Nuclear Systems Division

# MOLTEN SALT REACTOR INITIATING EVENT AND LICENSING BASIS EVENT WORKSHOP SUMMARY

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# 1. WORKSHOP OVERVIEW

Oak Ridge National Laboratory (ORNL) hosted a workshop on identifying potential initiating events for radioactive releases as precursors to licensing basis events for a generic liquid-fuel molten salt reactor (MSR). The workshop was held on May 21 and 22, 2019. Participants included representatives from seven prospective reactor vendors, industry bodies, US and Canadian regulators, US and Canadian national laboratories, and the academic community.

Accident sequence evaluation is central to deterministic and risk-informed performance-based reactor safety evaluation processes, and initiating events begin the accident sequence evaluation process. The workshop focused on how MSR initiating events feed into the risk-informed, performance-based reactor safety evaluation process described in DG-1353.[1] This report describes the workshop activities and results, provides a generic list of initiating events involving selected systems, and presents the estimation of the relative frequency and consequences of accident sequences for a few important, high-level initiating events.

The workshop results are intended to support the following:

1. **Reactor designers** by (a) identifying a set of generic initiating events that may help define licensing basis events that need to be prevented or mitigated, and (b) demonstrating a process designers can use to develop a complete set of initiating events for their plants

2. **Regulators** by (a) providing confidence in the ability to implement a systematic, expert evaluation process to establish MSR initiating events, and (b) providing representative initiating events that regulators should expect applicants to address

3. **The DOE-NE MSR R&D campaign** by supporting the planning process to determine the most critical topics for future research

The workshop employed a systematic process to elicit potential initiating events from a broad spectrum of MSR and reactor safety evaluation experts:

Participants were provided with background information on MSR characteristics and accident evaluation methods prior to the workshop to maximize productivity. The recommended reading list is provided below.

1. NRC Draft Regulatory Guide 1353, *Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors.*
2. EPRI TR-3002011801, *Program on Technology Innovation: Early Integration of Safety Assessment into Advanced Reactor Design.*
3. Vanderbilt University, EPRI, Southern Company Services, *Application of a Method to Estimate Risk in Advanced Nuclear Reactors.*
4. EPRI, *Program on Technology Innovation: EPRI Workshop on Process Hazard Analysis to Probabilistic Risk Assessment for Advanced Reactors Proceedings.*
5. ORNL-4541, *Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor.*
6. ORNL-TM-3532, *Design Studies of a Molten-Salt Reactor Demonstration Plant.*

[1] US Nuclear Regulatory Commission (NRC), "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," ML18325A214, 2019.

7. ORNL-TM-732, *MSRE Design and Operations Report: Part V, Reactor Safety Analysis Report*
8. ORNL/CF-61-2-46, *Molten-Salt Reactor Experiment Preliminary Hazards Report, Addendum 2.*
9. NEI-18-04, *Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development, Draft, Revision N.*
10. Oklo-2018-RlO-P, Rev. 0, DG-1353 Pilot, *Submittal to Support NRC Development and Implementation of DG-1353, Guidance to Risk-Inform Application Development and Contents Including Event Selection and SSC Classification.*
11. SC-29980-201, *Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors PRISM Sodium Fast Reactor Licensing Modernization Project Demonstration.*
12. SC-29980-200, *Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors High Temperature, Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration.*
13. ORNL-TM-2013-513, *Identification of Initiating Events for aSMRs.*

The workshop began with a series of talks describing how initiating events (IEs) and licensing basis events (LBEs) are employed in the reactor safety evaluation process. LBEs consist of an IE, the event progression that represents the plant's repose to the IE, and a well-defined end state. Identification of LBEs is described as Task 1 in NEI 18-04, which is endorsed in NRC DG 1353 as one of the methodologies for "determining an appropriate level of information for parts of preliminary or final safety analysis reports for non-LWRs."

The workshop included talks on the uses and roles of multiple safety evaluation methods: process hazards assessment (PHA), failure modes and effects analysis (FMEA), master logic diagraming (MLD), and hazards and operability studies (HAZOPS). A presentation on a recent process hazards assessment of ORNL's molten salt reactor experiment (MSRE) [2,3] was provided.

ORNL's Molten Salt Demonstration Reactor (MSDR) [4,5] was used as a template to structure the initiating event selection process. The workshop evaluated a limited number of systems that were intended to represent liquid-fueled MSRs. It did not attempt to be comprehensive or delve into the specifics of any particular design currently under development by reactor vendors.

On Day 1, workshop participants identified IEs for radioactive releases through (1) a system-by-system guided discussion forum, and (2) an online submittal process.

On Day 2, the discussion centered around estimating the frequency and consequences of the event sequences resulting from the IEs identified on Day 1.

A key workshop goal was to support development of a risk-informed, performance-based regulatory process for MSRs. Development of a representative set of MSR IEs supports this goal by providing a technical basis to align the event selection and evaluation process with MSR characteristics. It is otherwise unsuitable for MSR event selection to rely on the significant experience with large light-water cooled reactors (LWRs) that is embodied in the current regulatory process. The workshop demonstrated the capability of a system-by-system evaluation process to generate representative IEs for MSRs.

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[2] S. Beall and R. Guymon, *MSRE Design and Operations Report: Part VI, Operating Safety Limits for the Molten-Salt Reactor Experiment*. ORNL-TM-733, Rev. 2, 1966.
[3] S. Beall et al., *MSRE Design and Operations Report: Part V, Reactor Safety Analysis Report*. ORNL-TM-732, 1964.
[4] E. S. Bettis, L. G. Alexander, and H. L. Watts, *Design Studies of a Molten-Salt Reactor Demonstration Plant*, ORNL-TM-3832, 1972.
[5] M. S. Greenwood et al., "Demonstration of the Advanced Dynamic System Modeling Tool TRANSFORM in a Molten Salt Reactor Application via a Model of the Molten Salt Demonstration Reactor," *Nuclear Technology* (In Review).

# 2. EVALUATION PROCESS AND RESULTS

The discussion forum focused on specific high-level MSR systems. The choice of systems to focus on was motivated by the need to identify MSR-specific events and to spotlight issues with systems, structures, and components (SSCs) not generally present with other reactor technologies. This focus was designed to assist MSR designers who would consider these events, as well as the generically applicable IEs associated with other technologies, as part of their safety assessments if similar systems are employed.

Based on the open-literature MSR designs presented, a high-level reactor and fuel salt system was considered, along with the following generic subsystems: (1) a drain tank and decay heat removal system(s), (2) an off-gas system, (3) a fuel processing system, and (4) a reactor building.

Representative MSR drawings of these high-level systems for the MSRE, MSDR, and the molten salt breeder reactor (MSBR) were presented to the participants. Critical components, sources of radioactive material, flow paths, and other key features of the system were discussed. The question "What can go wrong?" was asked of the group for each system. Group members discussed potential IEs associated with the high-level systems, and the proposed IEs were recorded. Only a few events outside the current discussion scope were dismissed.

After each high-level system was discussed, other IEs missed during the identification session, those associated with systems not discussed, or more general MSR-related IEs were proposed by the group. It should be noted that there were many similar IEs proposed by the group. Detailed categorization of the IEs was beyond the scope of this workshop since this process was used generically and was not design specific. A moderate attempt was made to eliminate identical IEs. However, it was the intent of the workshop to report a list as close as possible to the originally developed list and to not obscure or inadvertently miss an IE that may be useful to an MSR technology stakeholder.

The list of IEs is not intended to be fully encompassing for any specific liquid fueled MSR. The MSR designer must determine which representative IEs are applicable to the design and identify any additionally IEs needed. The generated IE list is intended to support the IE identification process for any MSR design and to aid in the identification of IEs that are less easily perceived.

The table below presents the workshop-generated list of IEs grouped by high-level system.

| **Date: May 21, 2019** | **Session title: Roundtable Discussions of Identification of IEs** |
|---|---|
| **Session lead(s): Alex Huning, ORNL, Askin Guler Yigitoglu, ORNL** | |

| **System name:** Reactor and Primary Fuel Salt System | **System description:** Contains the fuel salt, generates power, major source of radionuclides | |
|---|---|---|
| **IE ID** | **IE for radioactive releases** | **Critical discussion notes** |
| 001 | Spectrum of fuel salt boundary breach sizes and locations:<br>• Breaches with gas release only<br>• Breaches with gas and fuel salt release | Consider all locations along the primary fuel salt boundary and sizes, including potential multiple simultaneous breaches (e.g., due to a seismic event or other event) |
| 002 | Actuation of dump system with or without pre-existing leak in dump tank coolant boundary | Could be a planned dump system activation; IE is a leak in the dump tank fuel salt boundary |
| 003 | Fuel salt freezing in heat exchanger (primary side plugged) | Root causes include salt composition changes which lower the freezing point, insulation detachment, over-cooling on the secondary side |
| 004 | Plugging on secondary side of heat exchanger | |
| 005 | Salt contamination/impurities | |
| 006 | Spectrum of unanticipated foreign material buildup in the core | Example: oil from leaking fuel salt pump |
| 007 | Improper fuel salt composition during loading to reactor system | Could lead to tube wall thinning, incorrect material compatibility, failure in chemistry control |
| 008 | Fuel pump over-speed / under-speed | |
| 009 | Vapor lock of heat exchanger | |
| 010 | Fuel salt freezing in primary fuel salt system | Could be due to a change in fuel salt composition |
| 011 | Undercooling/overcooling of heat exchanger | |
| 012 | Inadvertent fuel pump trip | |
| 013 | Fuel salt pump shaft shear/seizure | |
| 014 | Fuel salt pump seal failure | |
| 015 | Fuel salt pump shaft vibration due to its length | |
| 016 | Inadvertent freeze valve opening | |
| 017 | Change in core/primary fuel salt system void fraction too high or too low | Root causes include pump cavitation, spray nozzles, helium bubbles (too much injection, entrainment from pump) |
| 018 | Fuel salt pump cavitation | |
| 019 | Excessive graphite radiation damage | Salt penetrates the graphite |
| 020 | Change in fuel salt volume in the primary fuel salt system | Too much or too little fuel salt removal |
| 021 | Core flow blockage | Graphite breaking |
| 022 | Change in core geometry | |

| **System name:** Reactor and Primary Fuel Salt System | | **System description:** Contains the fuel salt, generates power, major source of radionuclides |
|---|---|---|
| **IE ID** | **IE for radioactive releases** | **Critical discussion notes** |
| 023 | Fuel loading error | |
| 024 | Fuel handling/loading detection error | |
| 025 | Graphite leaving the core | Graphite flows away with the salt |
| 026 | Foreign material damage to fuel salt pump | Graphite or other material damages the fuel salt pump |
| 027 | Uncontrolled rod withdrawal causes sudden power increase and undercooling | Due to eddy currents, localized concentration gradients, spurious rod withdrawal |
| 028 | Uncontrolled rod insertion causes sudden power decrease and overcooling | |
| 029 | Undesired poison insertion | |
| 030 | Spurious scram | |
| 031 | Undesired increase in cooling results in excessive overcooling | |
| 032 | Undesired decrease in cooling when required results in excessive overcooling | |
| 033 | Control rod thimble break | |
| 034 | Accidental soluble poison insertion | |
| 035 | Asymmetric cooling (flow distribution) in the core | Unbalanced, low flow in loop, not uniform |
| 036 | Redox control failure | |
| 037 | Plugging of gas lines | |
| 038 | Flooding of gas lines | |
| 039 | Thermal shield failure | |
| 040 | Core instrumentation failure | |
| 041 | Spectrum of unanticipated foreign material buildup in the core | Example: oil from leaking fuel salt pump |
| 042 | Overflow line plugging | |
| 043 | Air ingress/injection (oxygen/moisture) | |
| 044 | Overfilling/underfilling of fuel salt | |
| 045 | Inadvertent electrical heating of fuel salt | |
| 046 | Loss of offsite power | |
| 047 | Station black-out | Loss of all electricity, including backup sources (batteries, generators, etc.) |
| 048 | Loss of mechanical hold-down of buoyant core structures | Including graphite core structures |
| 049 | Failure of core support structures | |

| **System name:** Reactor and Primary Fuel Salt System | | **System description:** Contains the fuel salt, generates power, major source of radionuclides |
|---|---|---|
| **IE ID** | **IE for radioactive releases** | **Critical discussion notes** |
| 050 | Failure of core flow distributors | |
| 051 | Failure to separate gas from primary fuel salt | |
| 052 | Inadvertent over-pressurization | Can be caused by a design flaw or blockage |
| 053 | Gas entrainment by the primary fuel salt pump(s) | Off-gas separation performance deviation from its design state |
| 054 | Sudden change thermophysical properties causing a change in coolant flow rate | Sudden increase/decrease in salt viscosity |
| 055 | Partial loss of coolant flow results in insufficient heat removal | |
| 056 | Total loss of coolant flow results in insufficient heat removal | |
| 057 | Failure of instruments due to failure of radiation shielding surrounding the primary fuel salt system | |
| 058 | Failure of control system that leads to system instability (e.g., growing oscillations of pump flow) | Instabilities resulting from behavior of the control system. |
| 059 | Operation of passive systems in unstable regions (e.g., growing oscillations in mass flow of one of the natural circulation cooling loops) | Instabilities resulting from response of the system to external disturbances. |
| 060 | Break/leak/rupture in steam generator, releasing steam into the associate secondary/tertiary heat transfer salt | |
| 061 | Premature criticality due to control rod misalignment during primary fuel salt fill | |
| 062 | Core flow channel deformation due to nonuniform graphite swelling (radiation effect and thermal expansion) | |
| 063 | Injection of fuel salt of wrong concentration or temperature | |
| 064 | Void reduction or collapse due to sudden primary fuel salt pressure increase | |
| 065 | Primary heat exchanger tube flow-induced vibration (i.e., for shell-and-tube type heat exchangers) | |
| 066 | Increase/decrease in heat removal by fuel salt heat exchanger | |

| **System name:** Drain Tanks | **System description:** Contains the fuel salt primarily for decay heat removal | |
|---|---|---|
| **IE ID** | **IE** | **Critical discussion notes** |
| 067 | Failure of drain tank heating system combined with actuation of drain tank system | Thermal shock/reverse thermal shock could cause a leak/rupture in the drain tank system |
| 068 | Spectrum of drain tank fuel salt boundary breach sizes and locations | |
| 069 | Precipitation of fissile material in drain tank | Potential for a criticality event |
| 070 | Breach of drain tank decay heat removal (NaK) loop | NaK, under natural circulation, is assumed to be used as the fluid for closed loop cooling of the drain tank |
| 071 | Freezing of closed-loop cooling, causing a loss of heat sink | |
| 072 | Drain of fuel salt too slow | |
| 073 | Over-pressurization of water pool for drain tank | A water pool/tank is assumed as the ultimate heat sink for the drain tank |
| 074 | Loss of water make-up to the water pool of the heat removal system | |
| 075 | Over-pressurization of drain tank | |
| 076 | Improper reheating of drain tank (after system activation) | |
| 077 | NaK electromagnetic pump wrong direction | Flow stalls, decay heat removal is affected |
| 078 | Insulation failure on NaK, natural circulation loops | Could lead to freezing of closed-loop cooling |
| 079 | Open vent valve on drain tank | |
| 080 | Inadvertent closure of a normally open drain tank valve | Freeze valves will not open / failure to drain |
| 081 | Inadvertent opening of a normally closed drain tank valve | |
| 082 | Failure to drain or partial drain of fuel salt | |
| 083 | Drain tank vent valve fails closed | |
| 084 | Drain tank line to off-gas system is plugged | |
| 085 | Drain tank freeze valve leaks | |
| 086 | Loss of power to drain tank NaK pump | |
| 087 | Foamy salt / high void fraction in drain tank | |
| 088 | Unexpected thermophysical properties:<br>• Drain tank fuel salt<br>• NaK | |
| 089 | Water ingress to drain tank | |
| 090 | Failure of steam relief on water pool (ultimate heat sink) | |
| 091 | Flooding of area around drain tank | |
| 092 | Foreign material ingress to drain tank | Insulation failure |

| **System name:** Fuel Processing System | | **System description:** Responsible for disposal of used fuel salt and uranium extraction |
|---|---|---|
| **IE ID** | **IE** | **Critical discussion notes** |
| 093 | Inadvertent criticality, including inside the fuel storage tank and waste salt tanks | Many possible root causes: introduction of moderator after separation, existing water in system, etc. |
| 094 | Spectrum of fuel processing system breach sizes and locations | |
| 095 | Too much or too little fluorine flow | |
| 096 | Incorrect gas flow composition, contains impurities such as water and oxygen | |
| 097 | Loss of temperature control in NaF bed | |
| 098 | Transfer of fuel storage tank too soon | |
| 099 | Breach in the fuel salt sampler system | |
| 100 | Water-flooding of fuel storage tank | |
| 101 | Waste salt tank not connected during transfer | |
| 102 | Failure to empty fuel storage tank | Potential criticality concern |
| 103 | Breach in the caustic neutralizer | |
| 104 | Loss of inventory control of fissile material | |
| 105 | Mister failure | |
| 106 | Loss of heating/cooling in fuel storage tank | |
| 107 | Loss of shielding (as example of the fuel processing system) | |

| **System name:** Reactor Building | **System description:** Includes the cell cooling and heating for the reactor and support structures | |
|---|---|---|
| **IE ID** | **IE** | **Critical discussion notes** |
| 108 | Leak between containment cells | Leak path between containment cells, one root cause is thermally induced stress |
| 109 | Salt leak through liner, leading to direct contact with concrete | |
| 110 | Failed containment liner | |
| 111 | Inadvertent isolation of containment cells | |
| 112 | Loss of cooling to primary containment | |
| 113 | Loss of cooling to secondary containment | |
| 114 | Failure of decay heat removal system | |
| 115 | Over-cooling of reactor vessel auxiliary cooling system (RVACS) | |
| 116 | Flooding of decay heat removal system (e.g., RVACS) | |
| 117 | Adverse concrete-metal interactions | |
| 118 | Contamination of containment gas | |
| 119 | Loss of inert containment gas | |
| 110 | Heat exchanger leaks in secondary salt or power conversion systems | |
| 111 | Nitrate salt (heat transport salt) thermal breakdown | |
| 112 | Water-flooding of the reactor cell | |
| 113 | External hazards including high-winds, missile and aircraft impact | |
| 114 | Internally generated missiles | Compressor, turbine blades |
| 115 | Graphite oxidation | Requires high temperature source |
| 116 | Failure of the cooling system for the concrete (resulting in overheating and failure of the concrete support structure for the reactor) | |
| 117 | Break in the water lines for the concrete cooling system, resulting in water pooling within the cells; can cause overcooling in the nearby vessels/pipes | |

| **System name:** General IEs | **System description:** Around the room, general suggestions for any MSR system | |
|---|---|---|
| **IE ID** | **IE** | **Critical discussion notes** |
| 118 | Excessive noble metal plate-out on heat exchangers | |
| 119 | Cold slugging in the reactor core | |
| 120 | Support structure failure by seismic event | |
| 121 | Thermal degradation of concrete | |
| 122 | Inadvertent recirculation in the off-gas system | |
| 123 | Thermal shield stuck closed/open due to thermal damage | |
| 124 | Fuel salt in the off-gas system | |
| 125 | Reverse operation of cathodic protection system | Plutonium buildup, could lead to reactivity event |
| 126 | Failure in solid/liquid waste systems | |
| 127 | Failure of instrumentation | |
| 128 | Control system error or failure | |
| 129 | Incorrect measurement, prediction, or assumption of fuel salt thermophysical properties | |
| 130 | Inadvertent transfer of fuel salt | |
| 131 | Fuel salt hammer | |
| 132 | Loss of free liquid-gas interface, primary fuel salt system goes solid | |
| 133 | Thermal striping along primary fuel salt system | |
| 134 | Volatile gas formation from ingress/overloading of gases | |
| 135 | Stuck control rods | |
| 136 | Loss of control rod cooling | |
| 137 | Fire in reactor building | What impact would fire suppression system have during an event sequence? |
| 138 | Harsh environment in control room (chemical, radiation, tritium, high temperature, etc...) | If operators suddenly abandon the control room, what is the impact? |
| 139 | Local pressure build-up due to overheating in gas-pockets that cannot vent | |
| 140 | Cooling failure of systems or components that cannot operate far from ambient temperature (e.g. < 100 °C) | Some examples: electronics, concrete. |

# 3. FREQUENCY AND CONSEQUENCE ESTIMATES

On the second day of the workshop, the relative frequency and consequences of the identified MSR IEs were discussed. Because of time constraints, only a few high-level, critically important IEs were selected for extended discussion to estimate their anticipated frequency ranges and potential consequences should the event occur. The process provided insights that could be extended for further development of potential event sequences. The high-level initiating events selected for extended discussion included the following:

- Primary fuel salt heat exchanger failures
- Primary fuel salt boundary breaches
- Primary fuel salt composition changes
- Primary fuel salt void fraction changes
- Drain tank/decay heat removal failure
- Drain tank breaches
- Off-gas system breaches and other failures

For the workshop discussion, the qualitative frequency ranges used were as follows:

- Anticipated operational occurrence (AOO), $f_{AOO} \geq 10^{-2}$ per-plant-year
- Design basis event (DBE), $10^{-2} > f_{DBE} \geq 10^{-4}$ per-plant-year
- Beyond design basis event (BDBE), $f_{BDBE} < 10^{-4}$ per-plant-year

Qualitative consequences were also estimated in terms of severity and ranged anywhere from not determined, none, low, or high, where possible.

| **Date:** May 22, 2019 | **Session Title:** Roundtable Discussions of IE Frequencies and Consequences |
|---|---|
| **Session lead(s):** Alex Huning, ORNL; Askin Guler Yigitoglu, ORNL | |

| **IE:** Primary fuel salt heat exchanger failures | **Description:** Includes tube breaches between salt fluid systems and general events which lead to a failure to remove heat from the primary fuel salt |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 001, 003, 004, 007, 011, 065, 066 | **Frequency:**<br>• Specific failure modes depend on heat changer type (shell-and-tube vs. microchannel or “printed circuit” type)<br>• Failure modes should include corrosion, vibration, thermal cycling<br>• Using LWR experience, the consensus is that this type of events would be an **AOO**<br>**Consequences:**<br>• Highly dependent on what the secondary fluid is. If it is another similar salt at low pressure, then the radionuclides would stay contained in the primary fuel salt, with some very small or minor contamination of the secondary salt<br>• Minimal challenge to plant safety functions<br>• For very small or minor breaches and failures, there is some potential for plant controls and operations to compensate for the event and remain online until it is practical to shutdown<br>• For these types of events, the consensus severity of potential consequences is **low** |



| **IE:** Primary fuel salt boundary breaches | **Description:** Includes the spectrum of possible breach sizes and locations, except the primary salt heat exchanger |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 001, 002 | **Frequency:**<br>• Break frequency highly depends on the MSR design and primary fuel salt system type: loop, integral/pool, or modular<br>• Generally, as with LWRs and other reactor types, medium and large breaks are expected to be a **DBEs** or lower. However, valid data must be present to support such a frequency estimate. A certain IE frequency should not be “prescribed.” If a medium or large break “should be” a DBE, then data and plant design must confirm this conclusion<br>**Consequences:**<br>• Any breach along the primary fuel salt boundary is expected to drain into the reactor containment cell, which then drains to the drain tank system<br>• Consequences could be limited, depending on the defense-in-depth and functional containment approach<br>• The reactor containment cell could be a more critical barrier to radionuclide release than the primary fuel salt boundary<br>• Consequences of a primary fuel salt boundary breach are much higher if the reactor containment cell is initially open, has been breached, or contains air or oxygen instead of an inert gas<br>• For these types of events, the consensus severity of potential consequences is anywhere from **low to high**, depending on other event sequence safety function successes and failures |

| **IE:** Primary fuel salt composition changes | **Description:** Any IE involving primary fuel salt chemistry which leads to unanticipated changes in fuel salt properties, precipitation, plate-out, etc. |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 005, 054, 063 | **Frequency:**<br>• Root causes could include faulty redox control or sensor failures<br>• Contaminants could be present in the fuel salt during loading or helium gas being used to purge the salt<br>• What is an allowable chemical composition drift? What setpoints should be established?<br>• Liquid fueled MSRs are expected to have several systems that affect the fuel salt composition and chemistry. The failure of any of these systems, instrumentation, or controls could alter the fuel salt composition and chemistry. Therefore, this type of event is expected to be an **AOO**<br>**Consequences:**<br>• Enhanced corrosion could affect the life of the plant components, availability, and economic goals<br>• Potential component burn-through<br>• Potentially long timescales for any plant response or required control actions<br>• More information about specific event sequences is needed to establish general consequence estimates. However, one scenario is that the plant shuts down without any radiological consequence |



| **IE:** Primary fuel salt void fraction changes | **Description:** Any IE which leads to a change in the primary fuel salt bulk or local void fraction |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 017, 051, 064 | **Frequency:**<br>• Bubbles in the salt cause density wave oscillations in the core<br>• There are several root causes for too many bubbles. From the MSRE, cover gas was entrained in the pump bowl. There could also be too much helium injection, poor gas sparging, or other gas-ingress events<br>• No consensus frequency range was determined, as these events are highly dependent on the design, and there is a large uncertainty<br><br>**Consequences:**<br>• Members expressed a large level of uncertainty about the impact of fuel salt void fraction changes<br>• Control systems compensate for oscillations at the “noise” level. Locally, there can be large changes in void fraction. Globally, however, the system is stable<br>• The need for good, verified analytical tools was reiterated for these types of events<br>• The consequences and effects of such events should be addressed by the design. Therefore, event sequences consequences are highly dependent on the success or failure of the other systems such as control rods, decay heat removal, etc. |

| **IE:** Drain tank/decay heat removal failure | **Description:** Any IE which causes the drain tank failure to perform its function of decay heat removal from the primary fuel salt |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 071, 072, 073, 074, 080, 082, 090 | **Frequency:**<br>• Arrangement and types of valves is critical for ensuring that primary fuel salt enters the drain tanks<br>• Freeze valve reliability should be investigated in detail. Vanderbilt and EPRI performed an FMEA for the MSRE freeze valves<br>• For passive decay heat removal cooling, an “always on” type of system would eliminate failures related to startup operation of the decay heat removal system<br>• The frequency of these events is highly dependent on the design, the selection of safety related components, and the additional defense-in-depth systems and components available which can perform the safety function. However, the consensus was that major decay heat removal failure events would be in the **BDBE** range<br>**Consequences:**<br>• Vessels, pipes, and other steel components could melt leading to a breach and release of radionuclides<br>• Consequences can be mitigated by other containment barriers such as the reactor containment cell<br>• For these types of events, the consensus severity of potential consequences is anywhere from **low to high**, depending on other event sequence safety function successes and failures |

| **IE:** Drain tank breaches | **Description:** Includes the spectrum of possible primary fuel salt or off-gas lines or tanks associated with the drain tank system |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 067, 068, 075, 076 | **Frequency:**<br>• Breaches in the drain tank system could be caused by activities during or after the last activation or use of the system (e.g., failure to close a valve, over-cooling of the system)<br>• Thermal fatigue, thermal cycling, and thermal striping and ratcheting affect the breach probability over time<br>• Could have a leak that is pre-frozen by salt and is exposed during the next drain<br>• A transient event in the primary salt system, or slower than anticipated drain, leading to hotter-than-normal salt going into the drain tank, could cause a breach<br>• Like other primary fuel salt boundary breaches, the frequency is expected to be **DBE** or lower<br>**Consequences:**<br>• Like other primary fuel salt boundary breaches, severity is anywhere from **low to high**, depending on other event sequence safety function successes and failures |

| **IE:** Off-gas system breaches and other failures | **Description:** Any IE associated with the off-gas system which could lead to a plant disruption or release of radionuclides |
|---|---|
| **Example IEs** | **Discussion Notes** |
| 084, 096, 124, 134 | **Frequency:**<br>• Individual components must be examined independently<br>• Frequency of breach and other failure events depends on many factors: system design, batch vs. continuous operation, frequency of off-gas system component actuations, procedures and automatic actions, selection of safety related off-gas system components, defense-in-depth and other components available to perform safety related functions<br>• No consensus frequency was determined given the large uncertainty<br>**Consequences:**<br>• Potential for large gaseous fission product, highly corrosive, and hazardous material releases<br>• Consequences are limited by location and size where the breach occurs within the system. Breaches towards the end of the off-gas system are much lower in severity than those at the start of the system<br>• Consequences depend on holdup time and system capacity<br>• Other non-breach types of events could have significant impact on plant operations (e.g., pressure buildup and backflow to the reactor or drain tank system, impure helium going to the primary fuel salt system)<br>• Expected consequence severity is anywhere from **low to high** |

# 4. CONCLUSIONS AND NEXT STEPS

A generic list of MSR IEs was generated using a process of expert elicitation based on a combination of PHA and FMEA. It is the reactor developer's role to apply an IE identification method to his or her specific design at the beginning of the accident sequence evaluation process.

The workshop highlighted a number of commonly beneficial follow-on MSR safety-evaluation activities, as follows:

1. Create scenario-specific MSR PIRTs for both fast and thermal MSRs
2. An important adjunct of assessing what can go wrong is developing an understanding of phenomena that impact how failures can occur
3. Develop and validate accident progression modeling tools
4. Develop and/or acquire fuel salt performance models and data
5. Develop fuel salt radionuclide release models and data

It is anticipated that the stakeholders will cooperate in the development of tools, data, and their application.

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