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# A Case Study of LBE Selection based on the New Concept of TI-RIPB Methodology for MSRE

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## 1. Introduction

Recently the advanced non-LWR, such as HTGCR, SFR, MSR, etc. are drawing attention from world nuclear society. However, the existing regulatory regime (10CFR50 or 10CFR52) is not feasible to regulate advanced non-LWR. Therefore, the enactment of a new regulatory framework for regulating various advanced non-LWRs is underway in the USA.

In response to this need, NEI proposed a technology-inclusive, risk-informed, and performance-based (TI-RIPB) regulatory methodology in the NEI 18-04 document, and the US NRC endorsed the use of it in R.G. 1.233. Therefore, TI-RIPB methodology will be reflected in the new regulatory framework, 10CFR53.

TI-RIPB methodology [1, 2, 3] is composed of 1) LBE selection, 2) SSC categorization, and 3) DID adequacy evaluation based on the PRA/PSA.

In this paper as the first step of TI-RIPB methodology, LBE selection for MSR Experiment (MSRE) and consequence analysis have been carried out. For this purpose, two initiating events, the plugged drain line and fuel pump failure have been selected from MSRE. The topics on 1) the selection of initiating events, 2) the construction of event trees, and finally 3) the quantification of frequencies and consequences for the selected end state of event trees are surveyed from the literatures and discussed in this paper. A sensitivity of EAB size on the risk is also discussed, finally.

### *1.1 LMP Description*

In order to address the challenge of matching between the regulatory environment and new advanced reactor designs, the U.S. Department of Energy (DOE) established an industry-led the Licensing Modernization Project (LMP). Southern Company started this project in 2016 and finished it in 2020, targets amendment to key elements of the U.S. nuclear regulatory framework to specifically address licensing barriers in advanced reactor concepts. At first, by the end of 2017 LMP released technical reports that formed the foundation for a RIPB licensing structure that is broadly compatible with non-LWRs. These reports discussed techniques addressing selection of licensing basis events (LBEs), the classification of plant structures, systems and components (SSCs), and evaluation of the defense-in-depth (DID) adequacy.

Based on the methodology mentioned in these reports, LMP demonstrated the RIPB licensing structure for the following six types of advanced reactor technologies [4] :

- Sodium Fast Reactors (SFR)
- Lead Fast Reactors (LFR)
- Gas-Cooled Fast Reactors (GCFR)
- High Temperature Gas-Cooled Reactor (HTGR)
- Fluoride High-Temperature Reactors (FHR)
- Molten Salt Reactors

LBEs are conventionally selected and categorized as 1) normal operation, 2) AOO, and 3) accidents, mainly based on their frequency of occurrence for the PWRs [5]. There are many other ways to categorize events in PWRs as shown in Fig.1. Advanced reactors are different from PWRs that there needs new categorization method.

| EVENT FREQUENCY RANGE (per reactor-year) | PLANT CONDITIONS CATEGORIES | OTHER CATEGORIZATION SCHEMES | | | | | |
|---|---|---|---|---|---|---|---|
| | | NRC | | | ANS | | |
| | | 10 CFR | RG 1.48 ASME Code* | RG 1.70 Rev. 2 | 51.1 (N18.2) | 52.1 (N212) | 53.1 (N213) |
| Planned Operations | PC-1 | Normal | Normal | Normal | Condition I | Normal PPC | Plant Condition A |
| 10-1 | PC-2 | Anticipated Operational Occurrences | Upset | Moderate Frequency | Condition II | Frequent PPC | Plant Condition B |
| 10-2 | PC-3 | Anticipated Operational Occurrences | Upset | Infrequent Incidents | Condition III | Frequent PPC | Plant Condition B |
| 10-3 | PC-4 | Accidents | Emergency | Limiting Faults | Condition IV | Infrequent PPC | Plant Condition C |
| 10-4 | PC-4 | Accidents | Faulted | Limiting Faults | Condition IV | Limiting PPC | Plant Condition D |
| 10-5 | PC-5 | Accidents | Faulted | Limiting Faults | Condition IV | Limiting PPC | Plant Condition D |
| 10-6 | Not Considered | Accidents | Faulted | Limiting Faults | Condition IV | Limiting PPC | Plant Condition D |

*This terminology has been eliminated from 1977 version of the ASME Code.

Fig.1 Event Categorization in PWR [5]

### *1.2 MSRE Design Description*

MSRE (Molten Salt Reactor Experiment) had been designed and operated in 1960's at ORNL in USA as a test reactor. The MSRE was a single-phase circulating liquid fuel-salt cooled reactor. The fuel and coolant salt were UF4, and LiF-BeF2-ZrF4, respectively. Main flowsheet of MSRE is shown in Fig.2. MSRE can be categorized into 21 systems as listed below [6, 7, 8].

- Fuel salt loop
- Fuel salt drain/fill system
- Fuel salt processing equipment
- Coolant salt loop
- Coolant salt drain/fill system
- Sampler-enricher system

- Cover gas system
- Leak detection system
- Fuel salt off-gas system
- Coolant salt off-gas system
- Containment ventilation system
- Component cooling system
- Secondary component cooling system
- Instrument air system
- Treated cooling water system
- Tower cooling water system
- Vapor condensing system
- Liquid waste system
- Drain tank afterheat removal system
- Salt pump lube oil system
- Electrical system

MSRE is designed with two barriers concept. For instance, if fuel leakage from the primary fuel salt loop is considered then the fuel salt loop is primary barrier and reactor cell is secondary barrier.

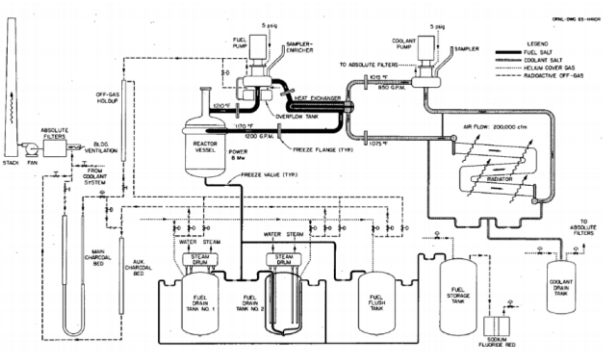

Fig.2 Molten Salt Reactor Experiment

## 2. LBE Selection Process

LBE selection process in LMP [9] is performed by utilizing the ASME PRA Standard [10]. In PSA application to LWR characterized with core and containment, PSA is composed of three steps, Level 1, 2, and 3. However, since the fuel is already at molten state and robust containment does not exist in case of MSRE, the classification of Level 1, 2 and 3 is meaningless. There are only frequency estimate and consequence estimate in PSA for MSRE.

For each IE (Initiating Event), an event tree is constructed in accordance with accident sequences expressed by event tree headings (typically split by success and failure of protection system). And the corresponding fault trees are constructed for quantification of each scenario. A PSA code, AIMS-PSA, which is developed by KAERI [11] is used to calculate event sequence frequencies by ETA and FTA. Finally, LBE is categorized as AOO, DBE, BDBE, and RR (residual risk), according to the occurrence frequencies.

Since the mechanistic source term for MSR is not available at this time frame, the conservative maximum credible accident source term and offsite dose calculation method from MSRE Safety Analysis Report is used. The consequence is expressed as public dose, i.e., rem. Frequency and consequence pairs plot is obtained and compared with F-C target curve as shown in Fig. 3.

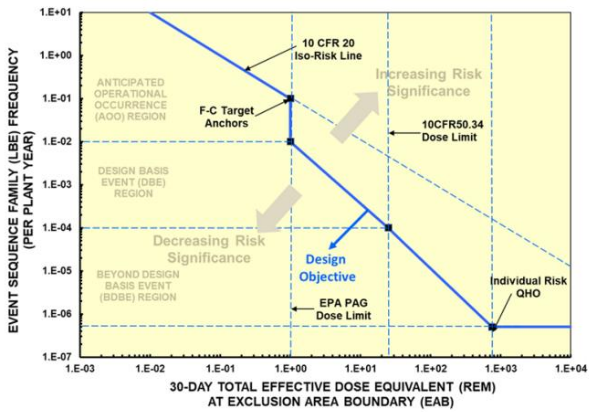


Fig.3 Frequency-Consequence Evaluation Criteria Proposed for LMP

### *2.1 Initiating Event and Event Sequence Analysis*

MSRE is not a power plant but test reactor to demonstrate the implementation of the concept of the fluid-fuel reactor into the real world. Therefore, the applicable PSA methodology will be different from that of power plants. Initiating events for MSRE will not be comprehensive compared to that of commercial power plant. LBE selection process starts with the development of definition of safety functions.

Frequency analysis is composed of 1) initiating event assumption, 2) construction of event trees, 3) construction of fault trees, and 4) data analysis.

Systematic procedures such as HAZOP, FMEA, Master Logic Diagram (MLD) are used to identify potential initiating events and to develop relevant event trees. MSRE is quite different from LWR that there are still a lot of trials to identify vulnerable systems and equipment, the failure of which leads to the release of radioactive materials to the environment.

At first, Chisholm et al [12] categorized plant operating states (POSs) into the following five states: 1) at power (normal operation), 2) filling (fuel salt), 3) shutdown, 4) fuel salt processing, and 5) maintenance.

And then locations of radioactive materials are identified with design and safety analysis reports. The following three locations are identified as radioactive material source locations.

1) Fuel salt in reactor cell and drain tank cell,
2) Off-gas system
3) Fuel processing system

The MLD for the MSRE PIEs was developed according to the following levels [13]:

- Level 1: Release of radioactive material (overall event of interest)
- Level 2: POS during which the release occurs
- Level 3: Inventory of radioactive material with potential for release
- Level 4: Level of barrier between inventories and the public/environment
- Level 5: Interface where barrier fails
- Level 6: Acute vs. latent failures of barrier
- Level 7: Challenge leading to failure of barrier
- Level 8: Functional failure leading to barrier challenge
- Level 9: Occurrence contributing to functional failure
- Levels 10+: Specific subsystem/component failures with similar system consequences

About 26 potential initiating events are defined through the systematic HAZOP and MLD development process in the above study [12].

Many different sets of initiating events are defined among various studies. About 140 initiating events are defined in MSRE IE Workshop [13].

Safety functions should be defined for the event tree construction. The safety functions of MSRE are 1) reactivity control, i.e., control heat generation, 2) control heat removal, and finally, 3) containment/ confinement of radioactive materials inside the building.

The following two types of events are categorized in MSRE SAR [7]. One type is reactivity insertion event and the other is general transient event.

Six reactivity events are analyzed.

- Fuel Pump Failure
- Cold Slug Accident
- Filling Accident
- Loss of Graphite from the Core (filling the empty space with fuel)
- Fuel anomalies (precipitated fuel circulating in core or non-mixed fuel lumps circulating in core)
- Uncontrolled Control Rod Withdrawal

Nine transient events are analyzed.

- Loss of Flow
- Loss of Heat Sink
- Decay Heat Removal
- Criticality in Drain Tanks
- Freeze valve and flange failures
- Excessive wall temperatures
- Corrosion
- Salt spillage
- Beryllium release from a leak

Among the above event types, the following three IEs are chosen and relevant ETs are developed in the LMP demonstration projects for MSRE [14].

1) Failure of component cooling blower (CCP-1) (Fig.4)
2) Uncontrolled withdrawal of control rods (Fig.5)
3) Leak in off-gas holdup piping (Fig.6)

A case study is done for OGS (off-gas system) deviation by EPRI and Vanderbilt University [15]. The result is shown in Fig.7. Fig.6 and Fig.7 have the same event tree names, i.e., off-gas system piping leakage but the differences of event tree structure between Fig.6 and Fig.7 are not discussed anywhere.

Two additional event trees are developed by the authors.

1) Plugging of primary fuel salt system (not shown in this paper)
2) Flow reduction accident of fuel salt pump failure (Fig.8)

*2.2 Fault Tree and Reliability Data Analysis*

More than ten fault trees are developed for the event tree headings after the development of event trees [14]. Construction of event trees and fault trees is possible due to the availability of detailed description and drawings on the design and safety analysis [6, 7, 8]. Event tree headings present the failure or success of safety systems belong to the required safety functions following the occurrence of initiating event. Initiating events requires numerical occurrence frequency values which reflects occurrence experience of similar kinds of reactors or industry. Failure rate or failure probability values are necessary also for the basic events which supports fault trees. Component reliability data gathered by nuclear [16] or chemical industry [15] can be used. Chemical industry data is used in reference [14]. Initiating event occurrence frequencies are summarized in Table I. The failure probabilities of safety function on demand, which are appeared as event tree headings and calculated by relevant fault tree models are shown in Table II.

Table I: Initiating event frequencies [14]

| Initiating event name | Initiating event | Contents of initiating events |
|---|---|---|

| | occurrence frequency (/yr) | |
|---|---|---|
| CCP1 failure | 1.33E-01 | All failure modes of blower fan |
| Uncontrolled control rod withdrawal | 1.18E-03 | Spurious control rod withdrawal |
| Off-gas piping line 522 leak | 1.00E-02 | Catastrophic failure of straight section of metal piping, 100 ft length assumed |

Table II: Failure probability on demand of event tree headings [14]

| Fault tree top event name | Top event occurrence probability on demand | Contents of fault trees i.e., contents of event tree headings |
|---|---|---|
| 565-ISO-FAIL | 2.20E-03 | Fault tree for failure to isolate reactor cell evacuation line |
| CCP-2-NO-START | 1.34E-01 | Fault tree for failure to start standby component cooling blower (CCP-2) |
| DT1-AHRS-FAIL | 1.38E-03 | Fault tree for failure of afterheat removal system in Drain Tank No. 1 |
| DT1-AHRS-F-RAD | 3.83E-04 | Fault tree for failure of afterheat removal system in Drain Tank No. 1 in the case of high radiation levels in the cell atmosphere. |
| DT2-AHRS-FAIL | 1.38E-03 | Fault tree for failure of afterheat removal system in Drain Tank No. 2 |
| DT2-AHRS-F-RAD | 3.83E-04 | Fault tree for failure of afterheat removal system in Drain Tank No. 2 in the case of high radiation levels in the cell atmosphere |
| NO-FS-DRAIN | 3.58E-06 | Fault tree for failure to drain reactor |
| NO-SCRAM-CR-F | 7.68E-06 | Fault tree for failure to scram reactor |
| NO-TX-DT1-DT2 | 2.16E-02 | Fault tree for failure to transfer fuel salt between drain tanks |
| NO-VENT | 2.94E-03 | Fault tree for failure of building ventilation system |

### *2.3 Source Term and Consequence Analysis*

Results of LMP Demonstration Project on MSRE performed by ORNL and Southern Company in 2018 [14] are shown on Table III and Fig.9.

As the consequences of most of the cases analyzed are negligible or minimal except one or two end states of off-gas release events. In the MSRE safety analysis Maximum Credible Accident (MCA) is assumed, and consequence is analyzed for the MCA. Maximum dose of 5 rem or even 100 rem is resulted from the consequence analysis at 3 km EAB of ORNL of USA. More detailed analyses are required for different sites. The results will be changed depending on the reactor power, EAB size, weather characteristic of the site, etc. The dose at the EAB due to an unmitigated leak in the off-gas system depends on the leak rate and duration and would likely be less than 100 rem. A dose of 100 rem at the EAB represents what was believed by the MSRE safety analysis to be a bounding scenario, but further analysis is required to estimate dose more accurately.

Table III: F-C Analysis Results for MSRE [14, 15, 18]

| Event Category | Frequency (/yr) | Consequence (rem) |
|---|---|---|
| AOO-1 | 0.115 | negligible – no release [14] |
| AOO-2 | 1.78E-2 | negligible – no release [14] |
| DBE-1 | 1.18E-3 | negligible – no release [14] |
| DBE-2 | 9.97E-3 | Minimal [14] |
| BDBE-1 | 2.39E-5 | ~5 rem max dose at EAB [14] |
| OGS-2 [15] | 2.44E-3 [15] | ~6 rem max dose at EAB [15] |
| BDBE-2 | 1.56E-6 | negligible – no release [14] |
| BDBE-3 | 3.47E-6 | Minimal [14] |
| BDBE-4 | 2.22E-5 | ~100 rem max dose at EAB [18] |
| | | negligible – no release [14] |

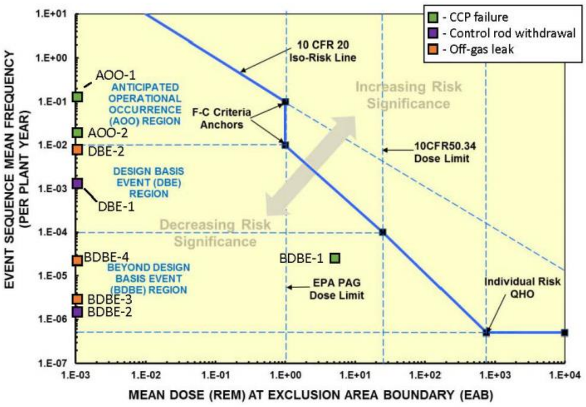

Fig.9 Result of LMP Demonstration on MSRE [14]

## 3. MSRE Case Study Results

### *3.1 A Variation of Results and Sensitivity*

As a variation of demonstration project, a case study for off-gas system (OGS) deviations was performed by EPRI and Vanderbilt University in 2019 [15]. The event tree is shown in Fig.7. Frequency and consequence estimations for event sequences are summarized in Table IV.

Table IV: Frequency and consequence estimations for OGS event sequences [15]

| **Sequence Name** | **Mean Frequency [/reactor-yr]** | **Event Classification** | **Qualitative End-State Point** |
|---|---|---|---|
| OGS-01 | 5.91E-02 | AOO | Off-gas leak to Rx cell for ~1 hour, stack isolation |
| OGS-02 | 2.44E-03 | DBE | Off-gas leak to Rx cell for ~1 hour, release to stack. .Environmental release of ~ 6 rem maximum in ref. [15], however, maximum 100 rem estimated in Ref. [18]. |
| OGS-03 | 4.56E-03 | DBE | Off-gas leak to Rx cell for >1 hour, stack isolation, reactor cell negative differential pressure maintained |
| OGS-04 | 7.10E-09 | Residual Risk | Off-gas leak to Rx cell for >1 hour, stack isolation, potential to lose reactor cell negative differential pressure |
| OGS-05 | 4.06E-07 | Residual Risk | Off-gas leak to Rx cell for >1 hour, release to stack |

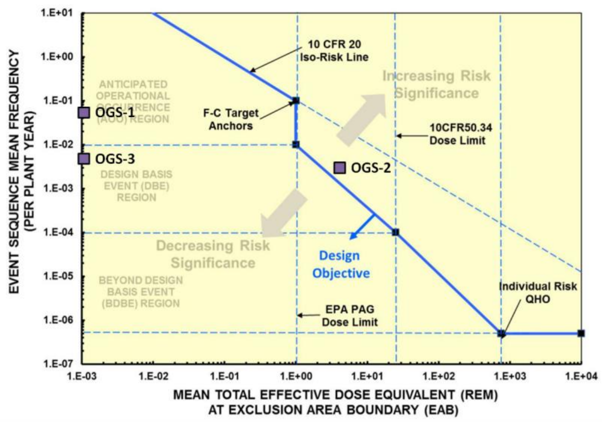


Fig.10 MSRE Case Study result by EPRI and VU [10]

The results of frequency and consequence estimation are shown in Figs. 10 and 11. They estimate the consequence up to 100 rem at maximum [18].

Fig.11 shows the sensitivity on the estimation of frequency and consequence between the two MSRE LBE evaluation studies [14, 15, 18]. Maximum 5 rem of dose is estimated on the maximum credible accident in the MSRE safety analysis report [7]. This result (5 rem) is used in BDBE-1 [14] and OGS-2 sequence [15]. While 5 rem is used for BDBE-4 in ref. [14] but even 100 rem is in ref. [18] presentation material without any reasonable explanation.

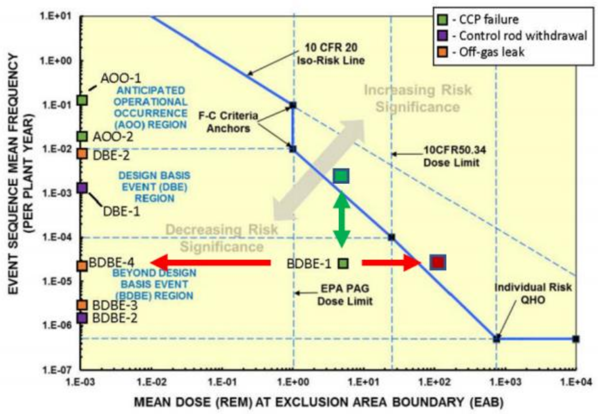


Fig.11 Sensitivity on the event frequency and consequence estimation

A case study for fuel salt pump failure is done in BEES Inc. The result is shown in Fig.8 and Table V. One AOO, one BDE, and one BDBE are identified for this case study. Here also 5 rem maximum public dose, which is calculated in MSRE SAR [7] for the 3 km EAB of MSRE, is used for the BDBE of this event tree.

### *3.2 Sensitivity of EAB Size*

Fig.12 shows the sensitivity of EAB size of MSRE site. If EAB size is reduced from current 3000 m to 500 m, and 100 m, then consequences would be increased from current 5 rem to 100, 1000 rem, respectively. Mechanistic source term is not identified for MSRE. MSRE safety analysis report is used instead of the mechanistic source term. ORNL evaluated the consequence at the site boundary at 3,000 m distance from MSRE facility. MSRE site boundary of 3,000 m is not realistic compared to the power level of MSRE, which is 7.4 MW maximum. Recently, one of the crucial issues of SMR/MSR is to reduce the EAB, EPZ commensurate with the power level of SMR/MSR. For instance, NuScale tried to reduce EPZ within the site boundary (EAB) of 500 m. For consequence estimate, two cases are assumed as follows:

Case A: EAB is determined at 3,000 m (ORNL estimate)

Case B: EAB is determined at 100 m. (BEES estimate)

Table V: Summary of frequency and consequence for Fuel Pump Failure Scenario

| Sequence | Frequency (/yr) | Consequences depending on EAB distances |
|---|---|---|
| AOO | 9.99E-02 | Negligible |
| DBE | 1.35E-04 | Negligible |
| BDBE | 2.99E-06 | (Case A) ~ 5 rem at 3,000m |
| | | (Case B) >1,000 rem at 100m |

If EAB should be reduced to the consequence of 100 m, consequence will be dramatically increased.

The consequence will also be changed depending on the reactor power and the weather conditions of the site.

The results will be changed depending on the comprehensiveness of initiating events categorization, completeness of event tree models, what kind of data use of numerical values on the initiating event frequency values, basic event failure rate or probability data, and assumptions and models in consequence analysis.

As shown in Fig.13, mitigation or prevention measures are necessary for the case of EAB size of 100 m during the design process because the frequency and consequence estimate is over the F-C target suggested in NEI 18-04.

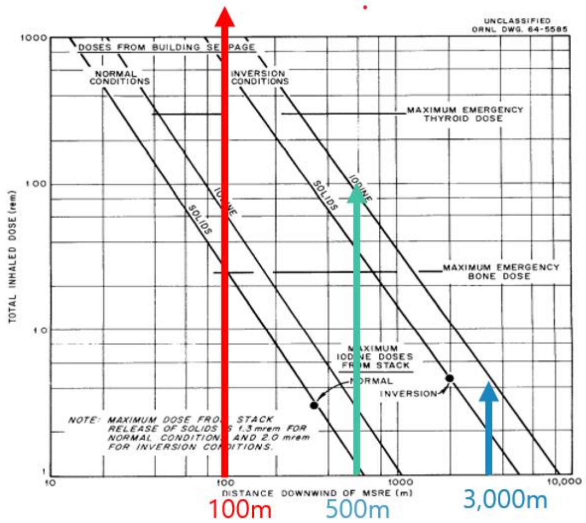


Fig.12 Sensitivity of consequence on the EAB size in MSRE site

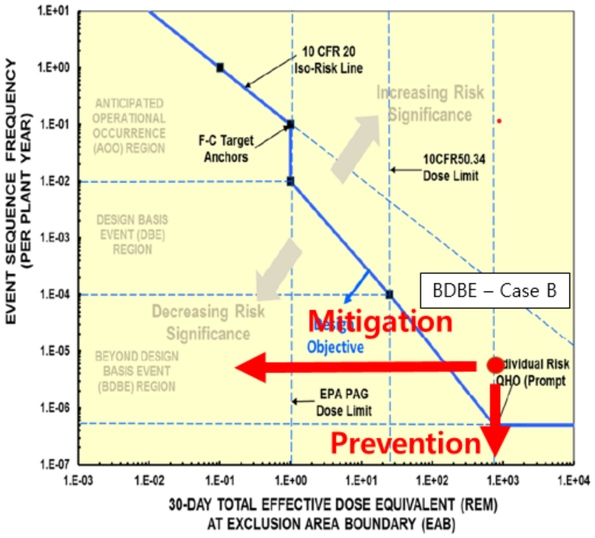


Fig.13 Necessity of Mitigation or Prevention Measures for Case B (EAB size of 100 m)

## 4. Conclusions

As the U.S. nuclear industry has proposed a new concept of TI-RIPB Methodology instead of the existing methods in ANSI/ANS 51.1 as part of the Licensing Modernization Project (LMP), the methodology for selection of LBE required to ensure the safety of advanced reactors is being changed. Based on the TI-RIPB Methodology, targeting the initiating event and accident sequences that can occur in MSRE, we demonstrated the selection of the LBE for MSRE, that is, AOO, LBE, and Beyond LBE, and evaluated the risk including the consequences and finally compared the risk result with F-C target to check the acceptability of the accident sequence.

The new TI-RIPB methodology is effective in assessing the risk of postulated initiating event and accident sequences and presents a framework for systematically selecting LBE. It will help not only advanced reactor developers but also regulators for assessing the risk level of new type of reactor.

To evaluate the risk level of MSRE in accurate way, however, sufficient information for design, PRA data, and sophisticated research for Mechanistic Source Term are required.

The methodology demonstrated in this paper will be implemented to CMSR, which is under development by Seaborg in Denmark, in applying Standard Design Approval in near future in Korea.

## ACKNOWLEDGEMENTS

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(NSSC) of the Republic of Korea. (No. 2202021-0122-CG100)

## REFERENCES

[1] US NRC, SECY-19-0117, 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, December 2, 2019.

[2] Nuclear Energy Institute, "Risk-Informed Performance-Based Technology Inclusive Guidance for Advanced Reactor Licensing Basis Development," NEI 18-04, Revision 0B, May 25, 2019.

[3] US NRC, Regulatory Guide 1.233 Revision 0, 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, June 2020.

[4] W.L. Moe, Amir Afzali, Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Final Project Report, INL/EXT-20-60393-R0, March 2020

[5] American National Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANSI/ANS-51.1-1983

[6] ORNL, Design and Operations Report Part I Description of Reactor Design, ORNL-TM-0728,1965.

[7] ORNL, MSRE Design and Operations Report Part V Safety Analysis Report, ORNL-TM-0732, 1964.

[8] ORNL, Preliminary Hazard Report, ORNL-CF-61-2-46 Addendum No.2, 1962.

[9] Idaho National Laboratory, "Next Generation Nuclear Plant Licensing Basis Event Selection White Paper," INL/EXT-10-19521, September 2010, [ADAMS Accession No. ML102630246].

[10] ASME/ANS, "PRA Standard for Advanced Non-LWR Nuclear Power Plants," ASME/ANS RA-S-1.4-2013, December 2013.

[11] S.H. Han, AIMS-PSA, A PSA Computer Code, KAERI.

[12] B.M. Chisholm, S.L. Krahn, 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, Volume 129, November 2020, 103507

[13] D.E. Holcomb, et al., Molten Salt Reactor Initiating Event and Licensing Basis Event Workshop Summary, ORNL/TM-2019/1246, July 2019

[14] B.M. Chisholm, G.F. Flanagan, S.L. Krahn, G.T. Mays, A New Look at Licensing Basis Events for the MSRE, ORNL-TM-2018-788, August 2018.

[15] EPRI and Vanderbilt University, "Molten Salt Reactor Experiment (MSRE) Case Study Using Risk-Informed, Performance-Based Technical Guidance to Inform Future Licensing for Advanced Non-Light Water Reactors," EPRI AR LR 2019-06, September 3, 2019.

[16] International Atomic Energy Agency (IAEA), "Components Reliability Data for Use in Probabilistic Safety Assessment," IAEA, Vienna, 1988.

[17] Center for Chemical Process Safety (CCPS), "Guidelines for Process Equipment Reliability Data with Data Tables," CCPS, New York, NY, 1989.

[18] B.M. Chisholm, S.L. Krahn, Licensing Basis Event Selection Case Study: The Molten Salt Experiment, Presentation Material, ORNL MSR Workshop 2017, October 3-4, 2017 (Oak Ridge, TN)

## ACRONYMS

ANS, American Nuclear Society
ANSI, American National Standard Institute
AOO, Anticipated Operational Occurrence
ASME, American Society for Mechanical Engineering
BDBE, Beyond DBE
CCP, Component Cooling Pump
CCPS, Center for Chemical Process Safety
CMSR, Compact Molten Salt Reactor
DBA, Design Basis Accident
DBE, Design Basis Event
DID, Defense-in-Depth
DOE, Department of Energy
EAB, Exclusive Area Boundary
EPRI, Electric Power Research Institute
EPZ, Exclusive Population Zone
ETA, Event Tree Analysis
F-C, Frequency vs Consequence
FHR, Fluoride-cooled, High-temperature Reactor
FMEA, Failure Mode and Effect Analysis
FTA, Fault Tree Analysis
HAZOP, HAZard and OPerability
HTGCR, High-Temperature Gas-Cooled Reactor
IAEA, International Atomic Energy Agency
IE, Initiating Event
INL, Idaho National Laboratory
KAERI, Korea Atomic Energy Research Institute
LBE, Licensing Basis Event
LFR, Lead-cooled Fast Reactor
LMP, Licensing Modernization Project
LWR, Light Water Reactor
MCA, Maximum Credible Accident
MLD, Master Logic Diagram
MSR, Molten Salt Reactor
MSRE, MSR Experiment
NEI, Nuclear Energy Institute
NRC, Nuclear Regulatory Commission
OGS, Off-Gas System

ORNL, Oak Ridge National Laboratory
PIE, Postulated Initiating Event
POS, Plant Operation State
PRA, Probabilistic Risk Assessment
PSA, Probabilistic Safety Assessment
PWR, Pressurized Water Reactor
RIPB, Risk-Informed, Performance-Based
SFR, Sodium Fast Reactor
SMR, Small Modular Reactor
SSC, Structure, System, and Component
TI-RIPB, Technology-Inclusive, Risk-Informed, and Performance-Based
VU, Vanderbilt University

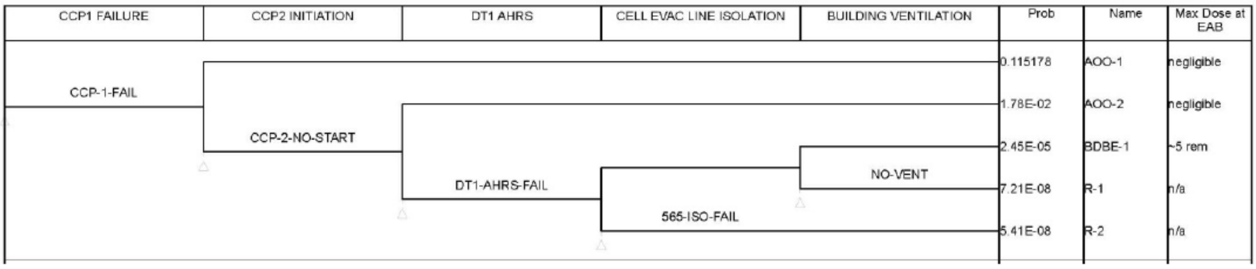


Fig.4 Event tree for failure of component cooling blower (CCP-1) [14]

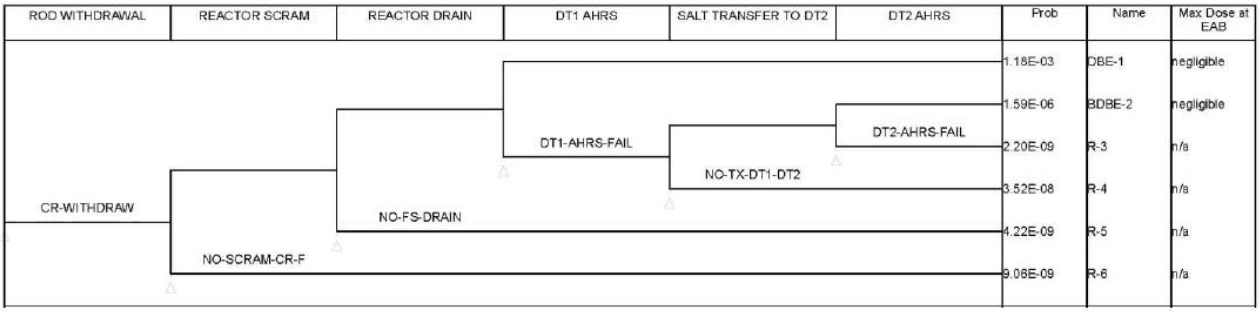


Fig.5 Event tree for uncontrolled withdrawal of control rods [14]

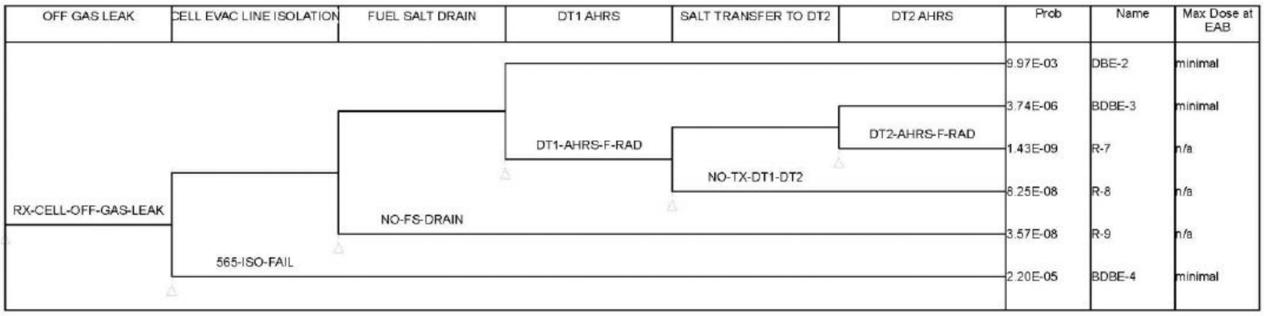


Fig.6 Event tree for leak in off-gas holdup piping [14]

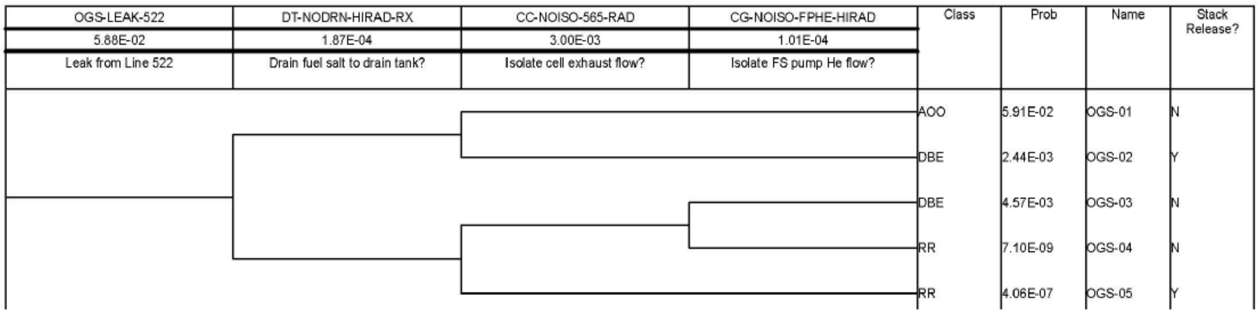


Fig.7 OGS event tree model [15]

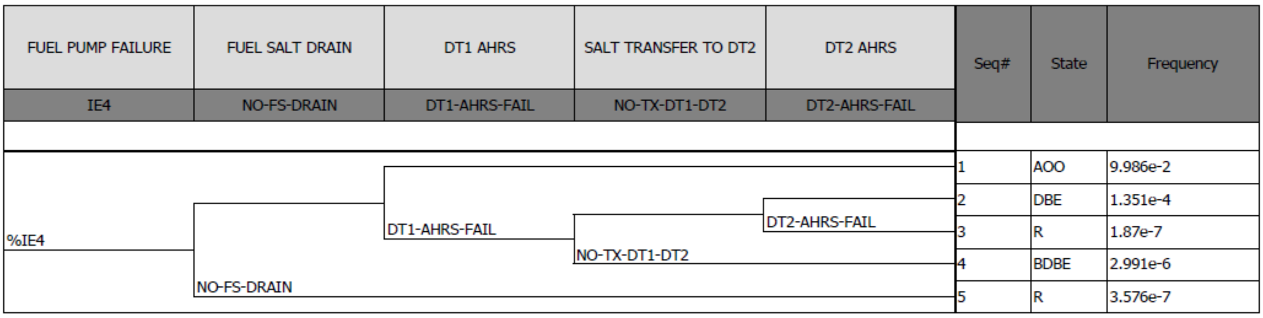


Fig.8 Event tree for fuel pump failure developed in BEES Inc.