Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, Oct 20-21, 2022

A Study on Internal PSA for a KAERI Molten Salt Reactor

Kilyoo Kim*, SangJi Kim
Korea Atomic Energy Research Institute
1045 Daedeokdaero, Yuseong, Daejeon, 34057, South Korea
**Corresponding author: kykim@kaeri.re.kr*

1. Introduction

Recently, KAERI has started a long term research project to build a molten salt reactor (MSR) which can be used for commercial ship.

Probabilistic Safety Assessment (PSA) which has played an important role in the safety for the commercial large reactors is also important in the design phase of small modular reactors (SMRs), even in the MSRs which do not have core melting accidents.

This paper describes how different the MSR PSA is, and how PSA is prepared in the KAERI MSR design.

2. Methods

2.1 A Characteristic of Molten Salt Reactor PSA

The typical characteristic of MSR is: There is no core melt accident which is usual used as a risk metrics in the commercial nuclear reactors, and thus, it is difficult to determine one top event in a large fault tree which could be developed from the all fault trees and event trees. Also, radionuclides could be released from the other process such as off-gas system rather than nuclear reactor.

2.2 *New Regulation in SMR Design*

To accept various non-light water SMR design, US NRC prepared a new rule, RG. 1.233[1] which indorsed NEI 18-04[2]. According to the RG. 1.233, each accident sequences of new non-light water SMRs should satisfy the Frequency-Consequence (F-C) Target, and design base event (DBE) and beyond design base event (BDBE) are determined by the frequency of the event.

The KAERI MSR PSA is strongly related to this F-C Target, and it is explained in the next section.

2.3 PSA in KAERI MSR

Since the design of KAERI MSR is in the beginning status, it is not possible to perform the PSA correctly. However, we can prepare the PSA by applying the new F-C curve regulation, and by setting up a backbone PSA structure.

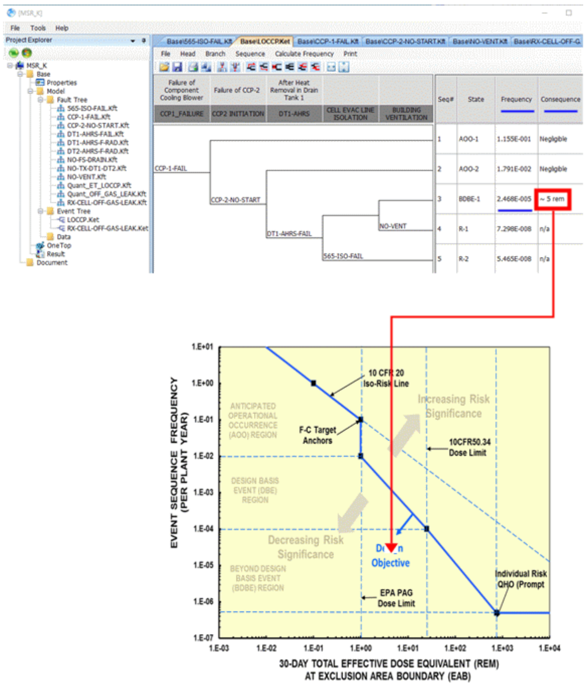

Fig. 1. KAERI MSR accident sequences and Frequency-Consequence Target of RG. 1.233

We assumed that KAERI MSR would require the following three (3) systems; 1) freeze valve system 2) off-gas system 3) reactor control system. Thus, a loss of freeze valve system and a leak of off-gas system were modeled by event trees and fault trees with AIMS/PSA software [3] as shown in Fig. 1. However, since a detailed design for KAERI MSR is currently not prepared, the MSRE design [4] and ORNL analysis [5] are used. Uncontrolled control rod withdraw event was not yet modeled since the safety analysis for the event was not yet completed.

PSA can calculate the frequencies of event sequences which correspond the y-axis values in the F-C curve. In AIMS/PSA, Calculate Frequency (→ Use branch probability) menu easily generate each event sequence frequency. However, we should also calculate consequences of the event sequences which correspond the x-axis values in the F-C curve through a source term analysis. We should design KAERI MSR whose (x, y) values of the event sequences should locates below the F-C curve.

2.4 Source Term Analysis

In Fig. 1, the event tree of the loss of freeze valve due to the failure of component cooling blower is shown. The consequence of the event sequence #3 (BDBE-1) was derived as about 5 rem by manual calculation without any mechanical source term analysis code. According to ASME PSA standard for Advanced Non-LWR Nuclear Power Plants [6], manual source term calculation would be regarded as 'Capability Category I' in the MST (Mechanical Source Term) Analysis part, and which is acceptable for conceptual design.

The 5 rem calculated by manual was described well in the Ref. [5] except the following Eq.(1).

$$TID_{max} = \frac{(2Q * \text{breathing rate})}{\pi * e * \text{wind speed} * \text{stack height}} \quad (1)$$

The maximum total integrated dosage TID_{max} can be derived by letting the derivative of the below Gaussian plume model (for diffusion from a continuous point source) be zero.

$$\chi = \frac{2Q}{\pi C_y C_z \mu x^{2-n}} \exp\left(- \frac{y^2+h^2}{c^2 x^{2-n}}\right)$$

where χ = activity concentration, μc/cm^3, for unit emission.

Currently, MST analysis code for MSR is not yet available, the manual calculation of event consequences could be useful for the conceptual design.

3. Conclusions

Since the core damage frequency (CDF) is no more risk metrics for MSR and the risk of the other process is also important, new safety regulation such as F-C Target of RG. 1.233 (or NEI 18-04) is inevitably required. By PSA modeling the loss of freezing valve and the leak of off gas system, and by checking whether the event sequences of the event trees satisfy the F-C Target, a new PSA approach for KAERI MSR is experienced and prepared.

Acknowledgement

This work was supported by the Korean government (MIST) [KAERI grant number 522310-22].

REFERENCES

[1] 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", Regulatory Guide 1.233, Rev. 0, June 2020.

[2] Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Rev. 1, August 2019.

[3] AIMS/PSA, a PSA software developed by KAERI in 2006, KAERI/RR-2719/2006, "Development of Integrated PSA DB and Application Technology"

[4] Brandon M. Chisholm, et. al, "A New Look at Licensing Basis Events for the MSR Experiment", ORNL/TM-2018/788, Aug. 2018.

[5] ORNL-TM-732, "MSRE Design and Operations Report Part V Reactor Safety Analysis Report" Aug. 1964

[6] ASME/ANS, Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants, ASME/ANS RA-S-1.4-2013, Dec. 2013