Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 8-10, 2024

Comparative Analysis of Pressurized Accidents in Accident Management Program for Innovative Small Modular Reactor

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***Keywords :** Innovative Small Modular Reactor, SMR, Accident Management Program, AMP, Station Blackout, SBO, Loss of Ultimate Heat Sink, LUHS, Extended Loss of All AC Power, ELAP, SPACE

1. Introduction

This paper presents a comprehensive analysis of accident management strategies for innovative Small Modular Reactor (i-SMR), focusing on the pressurized accidents as categorized with Beyond Design Basis Accident (BDBA) in Accident Management Program (AMP). The methodology involves a comparative study between the thermal hydraulic behaviors of the i-SMR to deal with the pressurized accidents scenarios such as Station Blackout (SBO), Extended Loss of All AC Power (ELAP) and Loss of Ultimate Heat Sink (LUHS). AMP derived from commercial nuclear power plants and those tailored for i-SMR. The analysis encompasses scenarios SBO, ELAP and LUHS, identifying and evaluating differences in reactor thermal hydraulic response, code implementation, and boundary conditions. Additionally, the study investigates key parameters including pressure, temperature, and fuel rod temperatures, and compares the heat removal capabilities facilitated by Passive Auxiliary Feedwater System (PAFS) and Passive Containment Cooling System (PCCS), along with heat removal rates and External Cooling Tank (ECT) water levels.

2. Methods and Results

The methodology involves a detailed examination of the pressurized accidents in AMP, with a specific focus on thermal hydraulic behavior of the Reactor Coolant System (RCS). The comparative analysis utilized the SPACE code and applied an optimal evaluation methodology to compare and analyze accident scenarios such as SBO, ELAP and LUHS.

2.1 Comparison between the Pressurized Accidents

Commercial nuclear power plants can simulate three types of pressurized accidents. The first, SBO, occurs due to the initial loss of power and unavailable Emergency Diesel Generators (EDGs). Loss of offsite power and unavailable EDGs result in the loss of power to the Turbine, leading to interruption of coolant supply and secondary side heat removal by Feedwater Pump (FWP). Additionally, loss of power to Reactor Coolant Pumps (RCPs) and Charging Pump (CP) leads to the cessation of forced circulation in the RCS. Consequently, there is a decrease in Steam Generator (SG) water level and pressure, leading to reactor shutdown. Auxiliary Feedwater System (AFWS), powered by batteries and Auxiliary AC Diesel Generator (AAC DG), can be injected to reduce decay heat and maintain reactor safety during this period.

In the case of ELAP, similar to SBO but with the additional unavailability of AAC DG, mobile generators are deployed to supply power. Strategies are implemented to restore the AFWS and Safety Injection Cooling System (SCS).

If SBO and ELAP occur, which require a power plant response due to power supply failure, LUHS represents an accident where power is available but heat removal is lost. In the initial phase of an LUHS accident, Essential Service Water System (ESWS) becomes inoperable, and the inability of Circulating Water System (CWS) on the secondary side is considered. Due to the ESWS failure, Component Cooling Water System (CCWS) heat removal becomes impossible, leading to the inability of RCP and CP heat removal. The inability of CWS results in the loss of vacuum in the condenser, causing the FWP operation to cease. Consequently, RCS pressurization occurs, but RCP and CP remain operational for a certain period during the early stages of the accident. When the operator becomes aware, they stop CP and activate Auxiliary Charging Pump (ACP) to prevent RCS leakage. RCS pressurization shuts down the reactor and activates AFWS, leading to a reduction in decay heat removal through the secondary side.

The design of i-SMR involves evaluating the applicability of commercial nuclear power plant systems, with most systems and configurations progressing at a similar level. The most distinctive features of i-SMR are Passive Auxiliary Feedwater System (PAFS) and Passive Containment Cooling System (PCCS), designed passively unlike conventional commercial reactors. The availability of systems and devices for operation in response to the above pressurized accidents is summarized in Table I, showing no significant difference in system availability for the three types of accidents. Therefore, conducting an analysis for the most conservative accident among the aforementioned pressurized accidents enables qualitative safety evaluations for accidents not analyzed. Additionally, as indicated in Table I, passive systems of i-SMR are continually available for use to ensure system safety regardless of accident conditions.

Table I: Availability of System and Component

	SBO	ELAP	LUHS
RCP	X	X	△
CP	X	X	△
Turbine	X	X	X
FWP	X	X	X
EDG	X	X	O
AAC DG	O	X	O
PAFS	O	O	O
PCCS	O	O	O

2.2 Assumptions

To quantitatively evaluate a pressurized accident, the following assumptions were made. For the AMP accident, the Best Estimated (BE) methodology is applicable for BDBA, so the nominal design conditions were applied. Despite differences in the applicability of various systems and components for each pressurized accident, only PAFS and PCCS were applied. The selected accident for evaluation is LUHS.

2.3 SPACE Code Simulation

The Best Estimated methodology through the SPACE code has been widely presented and utilized[Refer. 1], and in this assessment, the thermal hydraulic behavior is simulated using the SPACE code. The node diagram for the SPACE code analysis is presented in Fig. 1.

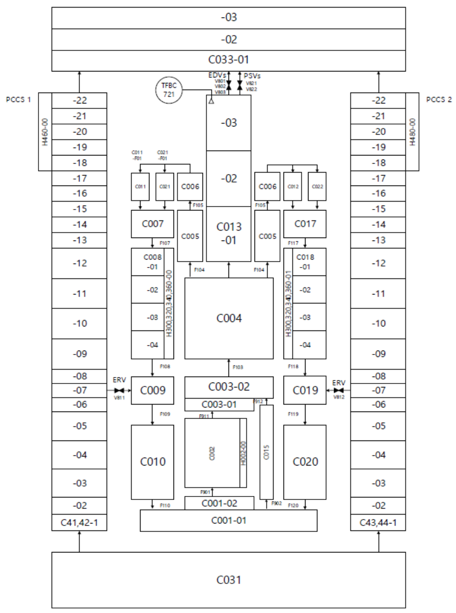

Fig. 1 SPACE Nodalization

3. Results

During normal operation of the innovative SMR, the loss of seawater as the ultimate heat removal source results in turbine trip and cessation of feedwater supply. Subsequently, due to the decrease in heat removal to the secondary side, the reactor is shut down by the reactor protection system. Following reactor shutdown, long-term core cooling is maintained through the operation of passive safety systems utilizing the ECT water source.

Table II: Sequence of Event

Time (sec)	Event
0.0	LUHS - Turbine Stop - Feedwater Pump Stop
11	Reactor Trip Signal Initiation - by High Pressurizer Pressure
12	Reactor Trip PAFS actuated
23,600	Safe Shutdown Temperature reached

The main events of the LUHS accident are presented in Table II, and the major thermal-hydraulic behaviors are illustrated in Fig. 2 through 5. The assumption is made that turbine trip and main feedwater stoppage occur simultaneously with the onset of the LUHS event.

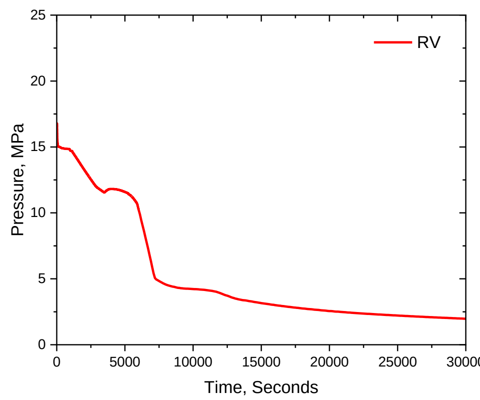

Fig. 2. RCS Pressure

The decrease in heat transfer to the secondary side leads to an imbalance between core heat generation and heat removal from the secondary side. Consequently, the RCS pressure increases as shown in Fig. 2, causing reactor shutdown by the Reactor Protection System (RPS) based on the reactor pressure vessel high pressure or the main steam line high pressure reactor trip signals.

Upon reactor shutdown signal, the isolation valve of the PAFS, which utilizes ECT water as a heat sink, opens to remove residual heat from the core. Cooling by PAFS reduces the temperature of the RCS coolant below the safe shutdown temperature criteria as shown in Fig. 3 and Fig. 4.

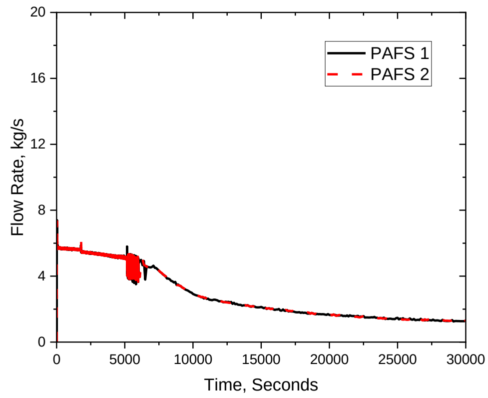

Fig. 3. PAFS Flowrate

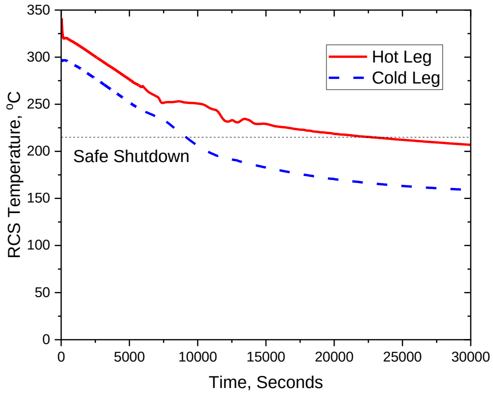

Fig. 4. RCS Temperature

4. Conclusions

A comparative analysis was conducted on the accidents categorized under BDBA as presented in the AMP, and the cooling capability of i-SMR was evaluated for the most representative accident. As a result, i-SMR was assessed to have sufficient response capability during accidents through passive safety systems, and it was evaluated to be capable of cooling until safety stoppage.

ACKNOWLEDGEMENTS

This work was funded by the Korea Hydro & Nuclear Power Co., Ltd (KHNP) to develop the innovative SMR technology.

REFERENCES

[1] H. K. Ahn, I. H. Song, S. J. Park, J. M. Lee, Evaluation of the Increase in Feedwater Accident with Common Cause Failure during Flexible Operation using SPACE Code, Transactions of the Korean Nuclear Society Spring Meeting, 2023