Sensitivity analysis of ATWS for Shin-Kori units 1 and 2 with various reactivity feedback conditions

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1. Introduction

The nuclear safety regulation in Korea was revised that it should be necessary to develop the Accident Management Plan (AMP) including the evaluation of multiple failure accidents. The multiple failure accidents means postulated accident conditions that are not considered for DBA, but can be evaluated in accordance with the best estimate methodology, and in which release of radioactive material is kept within acceptable limits. The Anticipated Transient Without Scram (ATWS) accident is included in the multiple failure accident that must be considered as shown in table I. In this paper, the evaluation of ATWS was performed for Shin-Kori units 1 and 2.

Table I: Type of multiple failure accidents that must be considered

Classifications	Type of accident
Accidents that should be considered	ATWS
	SBO
	MSGTR
	TLOFW
	ISLOCA
	LOSCS
	LOUHS
	Loss of SI or recirculation with SBLOCA
	LOSFPC

An ATWS is an anticipated operational occurrence accompanied by a failure of the reactor trip when required. The Reactor Protection System (RPS) is normally available to prevent and mitigate any transients. The RPS includes the electrical and mechanical devices and circuitry required to perform the functions of the RPS and the ESFAS. In case of CE-type NPPs, the Diverse Protection System (DPS) additionally provides a diverse backup to the RPS. The DPS initiates a reactor trip signal on high pressurizer pressure to decrease the possibility of an ATWS and provides an auxiliary feed-water actuation signal to assure that an ATWS will be mitigated if it does occur.

To reduce the risk of ATWS, the U.S. NRC issued code 10 CFR 50.62 and its amendments SECY-83-293. According to SECY-83-293, an ASME SERVICE Level C pressure of 22.06 MPa (3,200 psig) is an unacceptable plant condition [1]. The major concern of the ATWS is the expected high pressure in the primary system, which is caused by the reactor no trip. An ATWS results in a rapid increase of RCS pressure and threaten the integrity of RCS. However, this pressure excursion must not cause any mechanical damage. The intensity of this condition is strongly dependent of the Moderate Temperature Coefficient (MTC), Pressurizer Safety Valves (PSVs), and heat removal capacity of the SGs. Therefore, the sensitivity analysis was performed using the reactivity data at Beginning of Cycle (BOC) and End of Cycle (EOC) condition.

2. Evaluation Methods and Modeling

2.1 Nodalization model

The nodding diagram for Shin-Kori 1&2 is shown in figure 1. The nodding model was developed based on the nominal conditions. The reactor core is modeled as each one hot and average channel with 20 axial volumes. Some models such as Steam Bypass Control System (SBCS), Pressurizer Safety Valve (PSV), and Safety Depressurization System (SDS) were added.

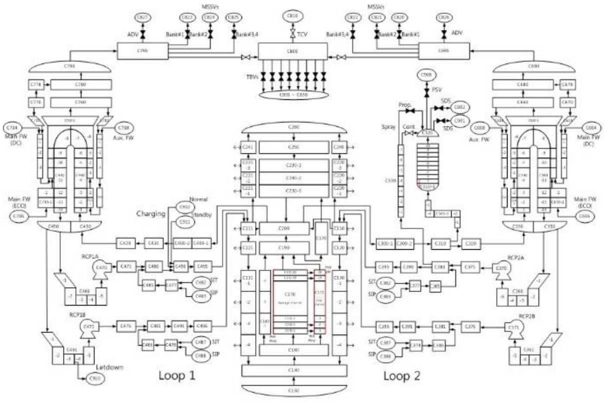

Fig. 1. RELAP5 nodalization for Shin-Kori units 1&2

2.2 DPS set-point and assumption

The ATWS scenario begins with the LOFW which causes a depletion in the SG inventory and reduce the secondary heat sink. After the accident occurs, the RPS is not operated and DPS will be operated by high pressurizer pressure. The set-points of reactor trip and AFW injection by DPS are shown in table II.

For the sensitivity analysis, reactivity data is used to the Nuclear Design Report (NDR) for cycle 2 of Shin-Kori unit 1 [2]. The values of reactivity related to moderator temperature, fuel temperature, and control rod worth are used at BOC and EOC for the sensitivity analysis. The BOC have Hot Full Power (HFP) and equilibrium xenon condition.

Table II: Major set-point related to ATWS analysis

Parameters	Value
Reactor trip set-point of DPS by PZR high pressure (psia)	2,397
Auxiliary feed-water actuation set-point by SG low level (%WR)	22.2

3. Analysis Results

The PZR pressure during the ATWS is shown in figure 2. The pressure decrease during 60 s after accident occurs because the turbine operates. However, the heat transfer of steam generator is rapidly decreased so that the RCS pressure is rapidly increased at 64 s. The negative reactivity is inserted due to the increase of RCS temperature as shown in figures 3 and 4. As shown in figure 5, the reactivity due to the MTC feedback is increased in opposition to the RCS temperature behavior. The RCS temperature is decreased after the peak pressure occurs so that the positive reactivity is inserted. The peak pressure during the BOC and EOC is about 2,500 psig and 2,494 psig, respectively.

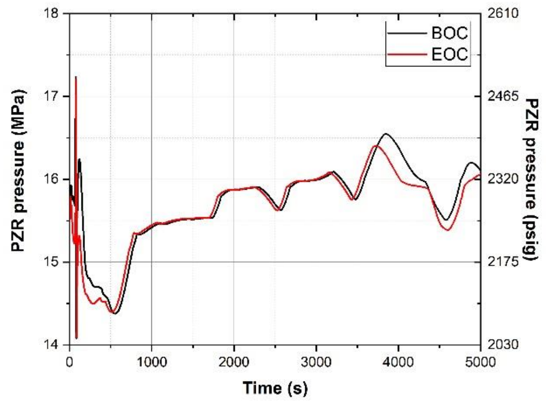

Fig. 2. PZR pressure behavior at BOC and EOC

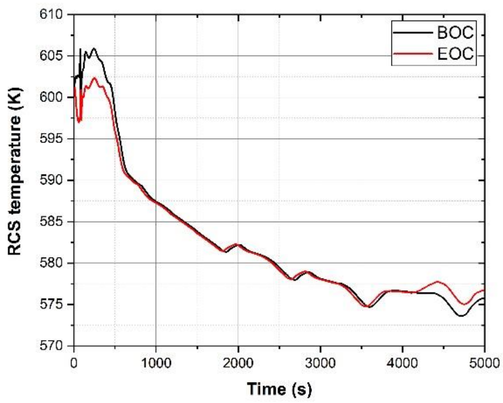

Fig. 3. RCS temperature behavior at BOC and EOC

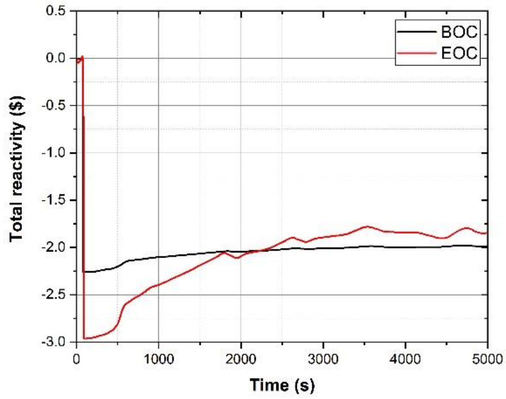

Fig. 4. Total reactivity behavior at BOC and EOC

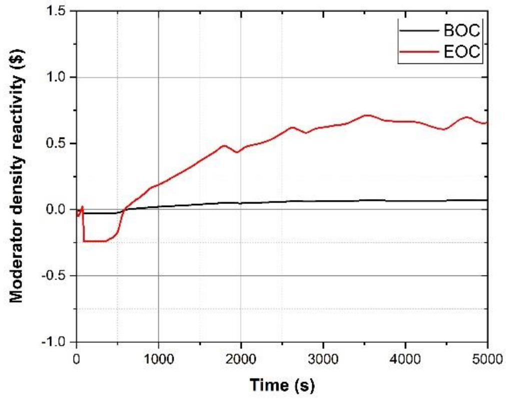

Fig. 5. Reactivity due to the MTC feedback at BOC and EOC

4. Conclusion

As the analysis results, the peak pressure at BOC is slightly higher than in case of EOC. Nevertheless, the RCS pressure of both case are within acceptance criteria limits of 3,200 psig due to the DPS operation and it is confirmed that the analysis result meets the requirement.

REFERENCES

[1] U.S. NRC, Amendments to 10 CFR 50 related to ATWS Events, SECY-83-293, 1983.
[2] KEPCO NF Co., Ltd., The Nuclear Design Report for Shin-Kori Nuclear Power Plant Unit 1 Cycle 2, Rev.0.