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Evaluation of the Increase in Feedwater Accident with Common Cause Failure during Flexible Operation using SPACE Code

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1. Introduction

The flexible operation of nuclear power plants involves continuous output variations. There are various ambiguous issues related to safety verification for flexible operation. Even for nuclear power plants based on baseload, it is difficult to define that existing safety analyses are invalid because they sometimes perform output fluctuations. Moreover, since output evaporation and reduction are performed at a relatively slow speed, existing safety analyses can be evaluated as valid. However, it is difficult to clearly explain the validity of existing safety analyses under continuous system abnormal conditions. Therefore, it is necessary to compare the safety analyses considering the characteristics of flexible operation with the existing safety analyses, and evaluate the validity of the existing safety analyses by reflecting any issues that were not considered in the existing safety analyses.

The Increase in Feedwater with Common Cause Failures (CCF+IFW) can be caused by excessive opening of the flow control valve or an increase in the speed of the pumps. As a result of the increase in feedwater flow, the temperature of the coolant in the primary system decreases, which leads to a decrease in the pressure of the primary system and an increase in the core power due to the moderator and fuel temperature feedback effects. The reactor's power coefficient of reactivity, which dominates the nuclear safety analysis, decreases until the pressure is restored. The power coefficient of reactivity, which is dominant in the CCF+IFW scenario, increases as a result of heat transfer from the primary side to the secondary side due to the increase in feedwater flow rate.

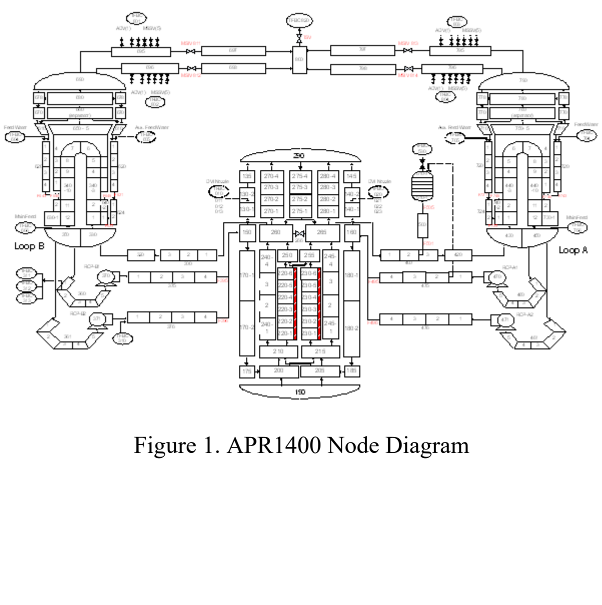

Figure 1. APR1400 Node Diagram

2. Methods and Results

2.1 Modeling & Assumptions

To analyze the CCF+IFW, the existing analysis methodology applied in APR1400 design was reviewed to confirm its appropriateness. The nuclear power plant is modeled as shown in Figure 1. To verify the adequacy of the analysis methodology, the conservativeness applied in the existing methodology was reviewed to examine the appropriateness of the design data, and the applicability of the SPACE computer code, which is to be applied during prompt transients, was examined. When examining the conservativeness of the design data, the nominal settings for thermal-hydraulic design data and instrument settings and control systems can be applied to perform the analysis using the best estimate method, so the adequacy of the nuclear design data conservativeness was examined. The Departure from Nucleate Boiling Ratio of the fuel is evaluated with THALES code.

Table I. Initial Conditions

Parameter	100% Power
Core Power, MWt	3,983
PZR Pressure, MPa	15.5
RCS Flow Rate, kg/s	20,992
Core Inlet Temperature, K	563.8
Secondary Pressure, MPa	7.03
Secondary Steam Flow Rate, kg/s	1,131
PZR Level, %	50.0
Steam Generator Mass, kg	98,911

2.2 Non-Equilibrium Condition

In the qualitative evaluation in the view point of the non-equilibrium conditions of reactivity and power distribution during the flexible operation, a CCF+IFW accident was selected as the representative accident which can be characterized by core operating parameters. The accident was evaluated without any operation action.

In the case of a CCF accident accompanied by an increase in main feedwater flow rate during the flexible operation, the shutdown of the reactor is not assumed to occur for 30 minutes after the reactor trip. Therefore, in the core safety analysis methodology, the output considering the xenon reactivity is taken into account. In addition, during the low-power operation due to the flexible operation, the xenon reactivity caused by the non-equilibrium condition of reactivity exists, and it can be assumed to be around 150 pcm. Based on the existing core safety analysis methodology, assuming the xenon reactivity caused by the CCF+IFW to be 155 pcm, the total reactivity of 305 pcm is considered by adding the xenon reactivity caused by low-power operation, and the analysis is performed.

In the low-power operation for flexible power operation, the power distribution becomes more curved compared to full-power operation. According to the existing standards for nuclear power plants, an ASI value of 0.026 is applied during full-power operation when analyzing CCF+IFW accidents. However, when performing CCF+IFW accident analysis for flexible power operation, an ASI value of 0.3 is applied to the lower part of the core, which is the most conservative assumption due to the increased curvature of the power distribution as shown in Figure 2.

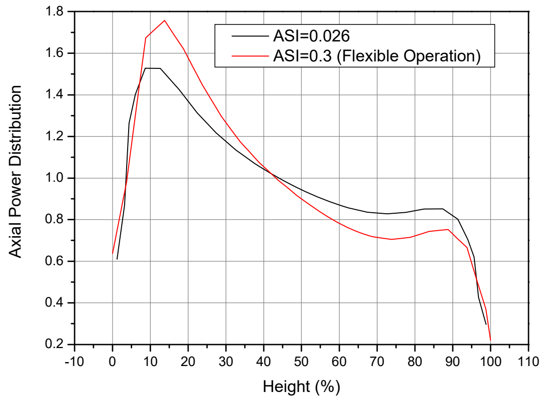

Figure 2. Axial Power Distribution

2.3 Results

According to the given information, in the event of a CCF+IFW accident, the increase in feedwater flow rate is assumed to be 170% of the initial flow rate at time zero as shown in Figure 3. This increase in flow rate affects the behaviors of core and fuel. The steam generator mass is shown to increase due to the sudden increase in feedwater flow rate as shown in Figure 4. This causes an increase in steam production, leading to an increase in turbine flow rate. The increase in secondary side heat removal helps to reduce the reactor coolant system temperature, which compensates for the decrease in power output due to the decreased temperature.

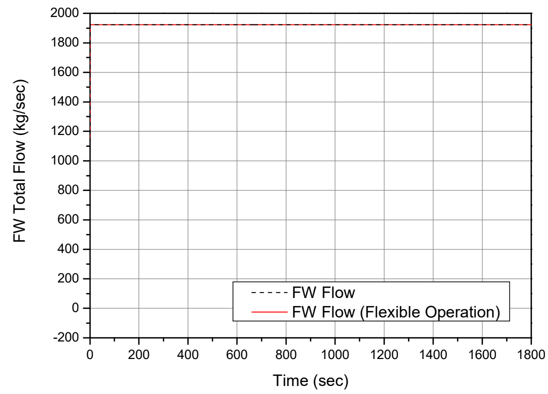

Figure 3. Feedwater Flow

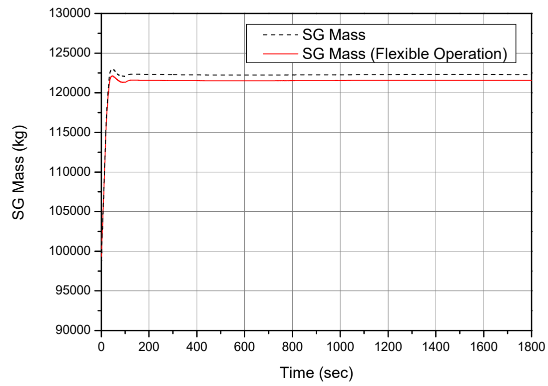

Figure 4. SG Mass

The core power increases by about 113% and then decreases to about 112% without considering the reactor kinetics non-equilibrium effects. However, considering the additional xenon reactivity due to the non-equilibrium condition, the core power increases up to about 115% and maintains at around 114%. Therefore, the difference in core power between considering the non-equilibrium reactor kinetics and not considering them is about 2% as shown in Figure 5.

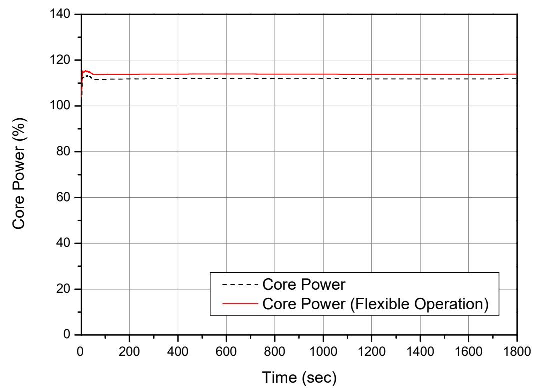

Figure 5. Core Power

The minimum DNBR is 2.52 without considering the non-equilibrium condition. However, considering the reactor kinetics non-equilibrium and output distribution non-equilibrium, it decreases to 2.11, indicating a difference in safety margin. Nevertheless, both cases ensure an adequate safety margin from SAFDL 1.29 as shown in Figure 6.

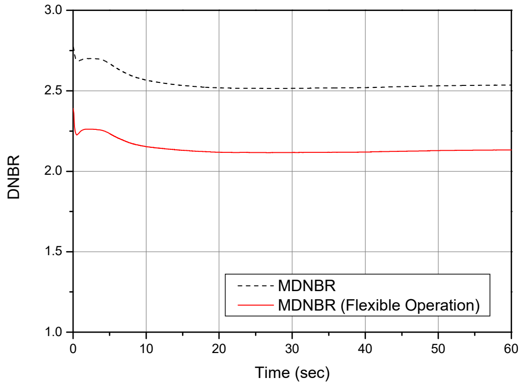

Figure 6. DNBR

3. Conclusions

The changes in the core and system due to flexible operation require an evaluation of the validity of existing safety analysis. Therefore, an analysis of CCF+IFW accidents was performed for non-equilibrium conditions of the core. CCF+IFW accidents are more limited when considering reactivity and axial power distribution imbalance. However, since all of the above events satisfy the allowable criteria, they were evaluated to have no impact on safety.

ACKNOWLEDGEMENTS

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