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Analysis of Anticipated Transients Without Scram (ATWS) Event for Kori Unit 1 Nuclear Power Plant

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1. Introduction

The ATWS is a transient of moderate frequency (Condition II) with a common mode failure affecting the reactor trip system. The consequence of this failure is that the scram banks don't fall into the core. The ATWS requirement, as stated in 10CFR50.62 released from NRC, is that the ATWS Mitigating System (AMS) [1] has to be installed. The AMS consists of equipments to trip the turbine and initiate auxiliary feedwater diverse from the reactor trip system. Now, the AMS has been installed to all the Korea nuclear power plants of Westinghouse type.

The generic studies of ANSI N.18.2 Condition II transients without scram, performed by Westinghouse for 3-loop plants, were reported in the document, NS-TMA-2182 [2], and led to a criterion of maximum primary pressure equal to 3,200 psig (ASME service level C stress limit proposed in NUREG-0460). As the result of transient analysis, loss of normal feedwater (LONF) event without scram was the most limiting event with respect to the overpressurization. Also, it was shown that the value of moderator temperature coefficient (MTC), key core physics parameter for mitigating ATWS, was more negative than -8 pcm/^{o}F for 95 % of the time when a reactor was critical. Then, if a MTC value is more negative than -8 pcm/^{o}F, the peak primary system pressure does not reach 3,200 psig even though ATWS occurs.

But the MTC value of Kori Unit 1 is -4.3 pcm/^{o}F for 95 % of the core cycle life time at 100 % power level. Because of the value, it came to the conclusion that the result of LONF analysis for Kori Unit 1 did not meet the ATWS rule. So, other method using detailed heat transfer coefficient for delta 60 steam generator of Kori Unit 1 generated by NOTRUMP computer code [3] was considered.

As a part of the development of ATWS analysis method for Kori Unit 1, sponsored by Korea Hydro & Nuclear Power Co. Ltd. (KHNP), analysis of LONF without scram was performed and analysis method and results were presented in the following sections.

2. Methods of Analysis

A loss of normal feedwater flow may occur due to failure or trip of main feedwater pump, spurious closure of the main feedwater isolation & regulating valves, or the loss of offsite power. In addition, it is assumed no reactor trip and that ATWS occurs. Following LONF without scram, the rate of heat generation in the reactor coolant system may exceed the heat removal capability of the secondary system. In this case, there will be much more increase in the RCS pressure, temperature, and pressurizer water level.

The assumptions used to analyze LONF event without scram for Kori Unit 1 are described below.

2.1 Initial Conditions for Analysis

Without any actuation of reactor protection system (RPS), all other components, equipments and systems are assumed to operate normally during LONF event without scram. Because of normal operating conditions, the assumed values of initial plant conditions (power, temperature, pressure, flow and so on) were nominal values without any uncertainty.

The event analysis was performed at initial 100 % power level. In this case, the AMS actuation following ATWS is assumed so as to trip the turbine and initiate auxiliary feedwater. Also, the analysis at initial 40 % power level was performed without AMS actuation because the AMS is designed not to operate at less than 40 % power of nominal power level.

Key plant parameters for Kori Unit 1 assumed in ATWS analysis are shown in Table 1.

Table 1. Initial Conditions

Parameters	Value
NSSS Power, Mwt	1,728.5
Reactor Coolant Pressure, psia	2,250
Reactor Coolant Vessel Average Temperature, ^{o}F	574.0
Zero Load Temperature, ^{o}F	547.0
RCS Flow, gpm	184,200

2.2 Flow Discharge Model for Pressurizer

In case of the LONF event without scram, pressurizer became a solid state (filled with water in pressurizer) due to rapid increase of RCS temperature and pressure. Then, water in pressurizer may be discharged through the nozzle of pressurizer relief & safety valves. To simulate the water discharge through the pressurizer relief & safety valves, a homogeneous equilibrium critical flow model (HEM) was applied.

2.3 Auxiliary Feedwater System Modeling

The most severe single failure in the auxiliary feedwater system was assumed to be occurred (turbine driven auxiliary pump fails to start). And the minimum auxiliary feedwater flow was assumed to be delivered to both steam generators.

2.4 MTC During 95 % of the Core Cycle Life Time

The MTC values were evaluated as -4.3 pcm/^{o}F at 100 % power level and 0.7 pcm/^{o}F at 40 % power level for 95 % of the core cycle life time. In case of 100 % power level, -3.7 pcm/^{o}F was assumed conservatively.

2.5 Detailed Steam Generator Heat Transfer Modeling

Detailed heat transfer coefficients for delta 60 steam generator of Kori Unit 1 were calculated using NOTRUMP computer code and the coefficients were shown in Figure 1.

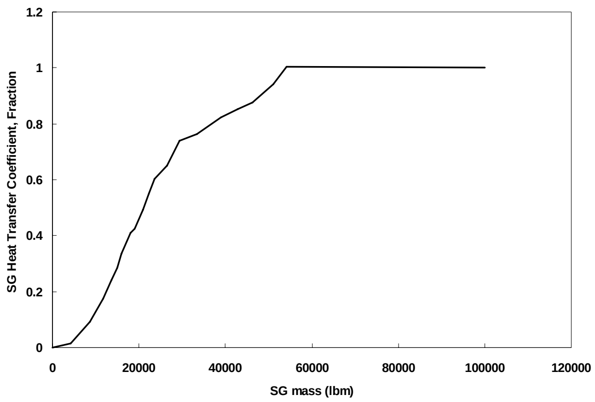

Figure 1. Detailed Heat Transfer Coefficient for Delta 60 SG

2.6 Computer Codes Utilized

LOFTRAN computer code [4] is used for studies of transient response of a pressurized water reactor system following the ATWS. LOFTRAN simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and the pressurizer.

And NOTRUMP computer code is used in the calculations of detailed heat transfer response for steam generator. The code can model one-dimensional thermal hydraulics using control volumes interconnected by flow paths (junctions) and a two-phase horizontal stratified flow.

3. Results

In the result of analysis of LONF event without scram, the peak pressure in the reactor coolant system was maintained below the pressure limit, 3,200 psig, for all cases (100 % power level and 40 % power level) as shown in Table 2. For 100 % power level, the time sequence of event is shown in Table 3 and the transient response for RCS pressure is shown in Figure 2.

Table 2. Result of LONF Event without Scram Analysis

Initial Power Level	Peak RCS Pressure, psia
100 %	3,045.30
40 %	2,755.90

Table 3. Time Sequence of Event

Time, sec	Sequence of event
0.0	Accident was occurred
0.1	Loss of normal feedwater
49.6	AMS actuation signal actuated
55.0	Opening of pressurizer relief valve
78.1	Turbine trip
96.0	Pressurizer water filled
105.0	RCS reached at the peak pressure
137.6	Both steam generator begin to receive auxiliary feed from two motor-driven auxiliary feedwater pumps

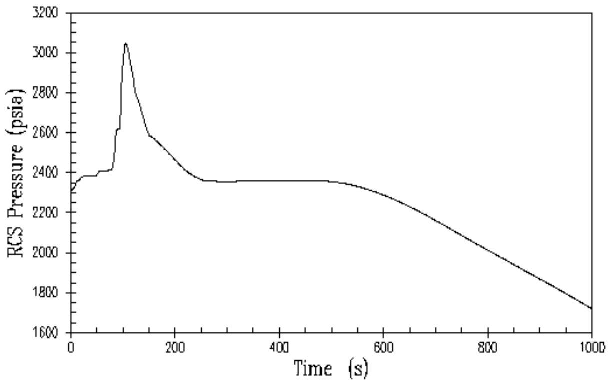

Figure 2. Transient Response for RCS Pressure

4. Conclusions

The analysis of LONF event without scram for Kori Unit 1 was performed using the MTC value during 95 % of the core cycle life time and detailed heat transfer coefficients of delta 60 steam generator. As a result, the peak RCS pressure was maintained below the ATWS overpressure criterion, 3,200 psig.

REFERENCES

[1] WCAP-10858-P-A, Rev. 1, M. R. Adler, AMSAC Generic Design Package, July 1987.
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[4] WCAP-7878, Rev. 6, LOFTRAN Code Description and User's Manual, Westinghouse Electric Corporation, February 2003.