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ATWS Analysis for Kori Unit 2

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1. Introduction

Regarding the evaluation of the PSR(Periodic Safety Review) of life extension operation of Kori Unit 2, the regulatory agency has requested an ATWS(Anticipated Transient Without Scram) analysis of Kori Unit 2 to ensure that the generic analysis of the ATWS case performed by Westinghouse is still valid. This paper describes analysis of the Kori Unit 2 ATWS and verify that the generic analysis of the ATWS performed by the WH is still valid.

2. Methods and Results

2.1 ATWS General Information

The ATWS event is an event in which a reactor trip signal has occurred due to the AOO(Anticipated Operational Occurrence), but the control rod is not inserted at all. ATWS has a very low probability of occurrence because when a certain AOO occurs, multiple failures or common cause failures occur independently in the reactor protection system. If the reactor is not tripped during the transient state, the pressure of the reactor coolant system may rise above the allowable value and damage to the RCS(Reactor coolant system) pressure boundary may occur. Typical AOOs that can have serious consequences when occurring simultaneously without scram include loss of feedwater, turbine trip, control rod assembly withdrawal, loss of AC power, loss of condenser vacuum[1].

The possibility of ATWS occurrence was raised in the late 1960s, and extensive research has been conducted in NRC(Nuclear Regulatory Commission) and industry in the United States. The US NRC announced the ATWS Rule WASH-1270, and since ATWS has a very low probability of occurrence, best estimate initial conditions and system variables were used for ATWS analysis, and the operation of the protection system and the control system were allowed except for reactor shutdown[2]. WH submitted ATWS analysis results to the NRC that could cover all WH-type nuclear power plants through WCAP-8330[3]. After the ATWS event on the US Salem Unit 1 in 1983, the US NRC laid the foundation for the ATWS final rule through SECY-83-293, and published 10CFR50.62 in June 1984[4][5].

According to 10CFR50.62, the installation of AMSAC(ATWS Mitigation System Actuation Circuitry), a diversity ATWS mitigation system separate from the reactor protection system, was required for WH-type nuclear power plants. AMSAC is a separate system from the protection system and mitigates the results of ATWS by supplying auxiliary feed water and turbine trip in the event of ATWS.

When ATWS occurs, the temperature of the reactor coolant rises sharply. Negative MTC(Moderator Temperature Coefficient) is important in mitigating this phenomenon. As the temperature of the reactor coolant rises, the reactor coolant expands, generating level increase of the pressurizer, and the pressure of the primary coolant rises sharply from the time the fluid on the secondary side of the steam generator is depleted due to the loss of the main feed water supply flow rate. WH has submitted analysis results to NRC that don't exceed the reactor coolant system pressure limit of 3200 psig when ATWS occurs under operating conditions with MTC below -8 pcm/°F for 95% of the period with a core power of 80% or more through NS-TMA-2182[6].

Kori Unit 2 installed AMSAC in 2003 in accordance with the ATWS Rule of 10CFR50.62, and when ATWS occurs, the turbine is tripped and auxiliary feedwater supply is started when the secondary side water level of the steam generator reaches the low-low water level setpoint[7].

2.2 Overview of the ATWS event

LONF(Loss of Normal Feedwater event) is the most limiting accident in terms of overpressure in the primary coolant system during ATWS events. LONF is caused by failure of the main feedwater pump, valve malfunction, or loss of offsite power, resulting in a decrease in the heat removal ability of the secondary system to remove heat generated in the reactor. In case of LONF, without scram is accompanied, the power of the reactor is maintained, while the temperature of the reactor coolant rises sharply due to the degradation of the heat removal capacity of the steam generator secondary system. The reactor power decreases due to the insertion of the negative reactivity by the negative MTC as the increase of the reactor coolant temperature. An increase in the temperature of the reactor coolant cause coolant expansion, which causes the reactor coolant pressure to gradually increase to reach the pressurizer full condition, and the reactor coolant pressure rises sharply when the fluid on the secondary side of the steam generator is depleted. The reactor coolant pressure remains high and the reactor coolant is released through the pressurizer relief valve or the pressurizer safety valve as there is no reduction in the reactor power due to the insertion of the control rod or boron injection. Therefore, the mitigating ability of ATWS results depends on the insertion of negative reactivity by MTC. In addition to MTC, the mitigation of additional ATWS results is achieved by turbine trip by the operation of AMSAC and supply of auxiliary feedwater. In the case of Kori Unit 2, when the steam generator level reaches the low-low water level set point, turbine trip and auxiliary feedwater supply start after the delay time of AMSAC. Delay time for turbine trip and auxiliary feedwater supply are within 30s and 90s, respectively.

2.3 Input variables and assumptions

ATWS is not classified as a DBA(design basis accident) because the probability of occurrence is very low. Therefore, the ATWS analysis consider the initial conditions and system variables of the plant's normal conditions and assumes the operation of the control and protection system except for reactor trip.

Table 1. Event analysis initial conditions

Initial conditions	Values
Reactor power	1882 MWt
Reactor coolant AVG temperature	583 °F
Reactor coolant pressure	2250 psia
Reactor coolant flow	186,200 gpm
PZR fluid volume	587 ft^3
Main feedwater temperature	408.4 Btu/lbm
SG secondary side fluid mass	102,617 lbm/SG

- Protection system model

• Pressurizer safety valve: the steam discharge before the pressurizer full water level is modeled according to the steam discharge capacity of the pressurizer safety valve, and the liquid is released according to the HEM(Homogeneous Equilibrium Model) after the pressurizer full water level. The pressurizer safety valve is modeled to start opening at the normal opening set point and fully open after 3% accumulation.

• Main steam safety valve: It is modeled that the valve opening starts at the highest pressure among the main steam safety valves and is fully opened after 3% accumulation.

• AMSAC: It is modeled that turbine trip occurs and auxiliary feedwater is supplied by the low-low water level signal of the steam generator of AMSAC.

- Control system model

• Pressurizer relief valve: It is opened at the normal setpoint of the pressurizer relief valve, and the discharge flow rate is modeled in the same way as the pressurizer safety valve.

• Steam dump: It is modeled that 40% of the steam is dumped into the condenser.

• Pressurizer spray and heater: It is modeled that the pressurizer spray and the heater are operable.

- Reactor kinetics model

• MTC: Assume MTC corresponding to 95%/95% of the current operating cycle. (-8 pcm/°F)

• Doppler coefficient: Assume doppler coefficient corresponding to minimum reactivity feedback.

- Single failure

• No single failure shall be considered in the reactor protection system other than without scram.

2.4 Analysis code

Kori Unit 2 ATWS is analyzed using LOFTRAN and NOTRUMP codes used in DBA in Chapter 15 of FSAR(Final Safety Analysis Report)[8], [9]. The LOFTRAN code simulates the behavior of the system and uses the NOTRUMP code for fluid mass analysis according to the steam generator heat transfer coefficient and secondary water level.

The LOFTRAN code is an analysis code that simulates the systematic behavior of a multi-loop system, including a reactor vessel, hot leg and cold leg, steam generator and a pressurizer, and includes a pressurizer heater, a spray, and a pressurizer safety valve model. It also includes a point kinetics model and simulates the moderator, nuclear fuel, boron and the reactivity effect of control rod operation. The secondary side of the steam generator is simulated with homogeneous saturated mixed steam in a thermal transient state. The reactor protection system has the function of reactor trip due to high neutron flux, overtemperature and overpower ΔT, high and low pressure of the pressurizer, low flow rate of coolant, and high water level of the pressurizer, and the control system includes the control rod control, steam dump, and pressurizer pressure control model. It also includes a safety injection system model including accumulator.

NOTRUMP is a common one-dimensional network code used to analyze thermodynamic transient behavior. This code is used not only for primary and secondary transient accidents, but also for the analysis of SBLOCA(Small Break Loss Of Coolant Accident). The modeling concept of this code uses multiple independent target volumes. These volumes are represented by mass and energy, and are connected by appropriate flow paths between each volume to enable mass and energy exchange. The mass and energy associated with the target volumes are homogeneously present in the volumes, and the flow rate and pressure drop associated with the flow path are connected to the center point of the target volumes by a one-dimensional flow path.

2.5 Acceptance criteria

ATWS, which is limiting from the perspective of reactor coolant system pressure, is a loss of feedwater without scram. Kori Unit 2 satisfies the requirements of the ATWS Rule 10CFR50.62 with the installation of AMSAC that supplies turbine trips and auxiliary feedwater to the diversity ATWS mitigation system for WH-type nuclear power plants. The ATWS Rule and AMSAC design are based on WH's generic ATWS analysis.

According to WASH-1270, NRC's technical report, the maximum primary stress of the reactor coolant pressure during ATWS should be maintained below the emergency condition of the ASME Nuclear Power Plant Component Code, Section III. Currently, this condition is ASME Boiler and Pressure Vessel Code Service Limit C, and the allowable pressure of the reactor coolant system is 3,200 psig.

The basis for the ATWS Rule and analysis is that the maximum pressure of the reactor coolant system does not exceed the allowable pressure of 3,200 psig during 95% of the plant operating cycle period.

2.6 Analysis results

As a result of the ATWS analysis Kori Unit 2, the maximum pressure of the reactor coolant system is 2,756 psia, which satisfies the acceptance criteria of 3,200 psig. The event sequence of ATWS are shown in Table 2, and a graph of the main system variable is presented in Figure 1 to Figure 8.

Table 2. Kori Unit 2 ATWS event sequence

Time (sec)	Event sequence	Value or setpoint
0.0	Event start (LOF)	-
0.1	Total loss of normal feedwater	-
26.4	AMSAC signal occurred (SG low-low water level)	13.2 NR%
56.4	Turbine trip	-
82.4	Peak pressure in RCS	2,756 psia
86.4	Aux. feedwater injection	-

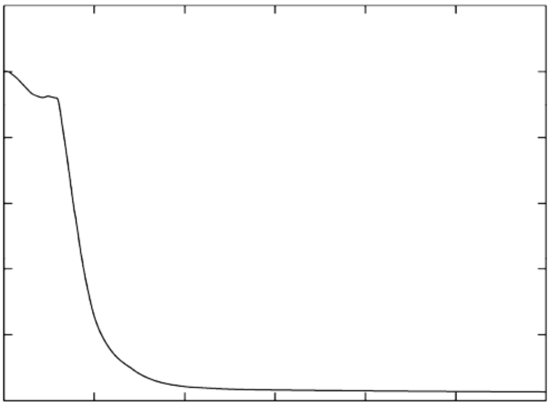

Fig. 1. Core power over time.

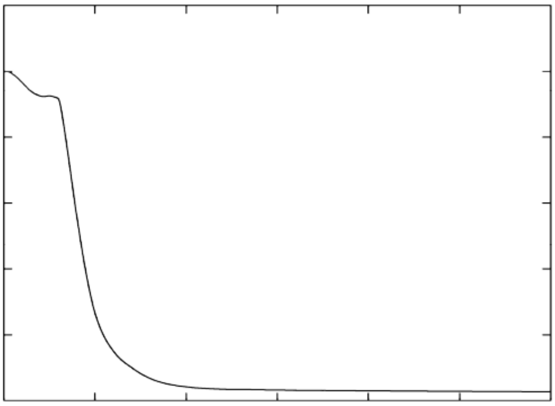

Fig. 2. Core heat flux over time.

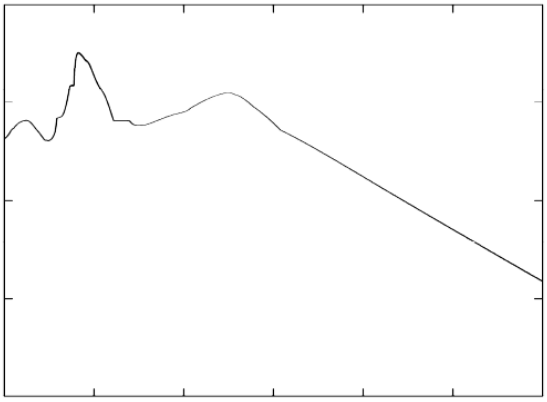

Fig. 3. RCS pressure over time.

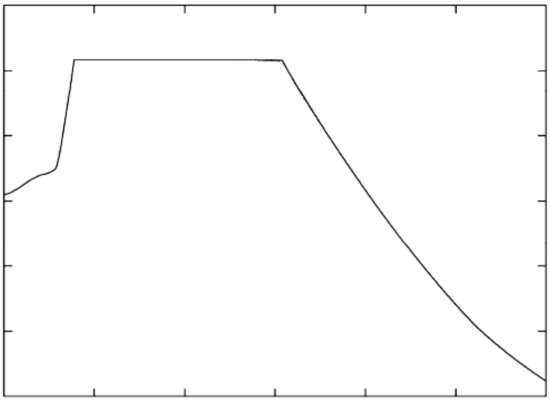

Fig. 4. PZR water volume over time.

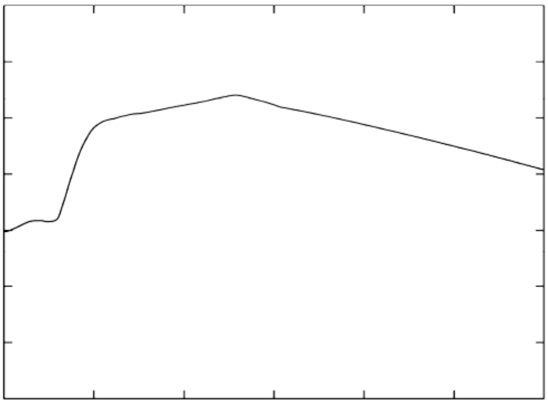

Fig. 5. Vessel inlet temperature over time.

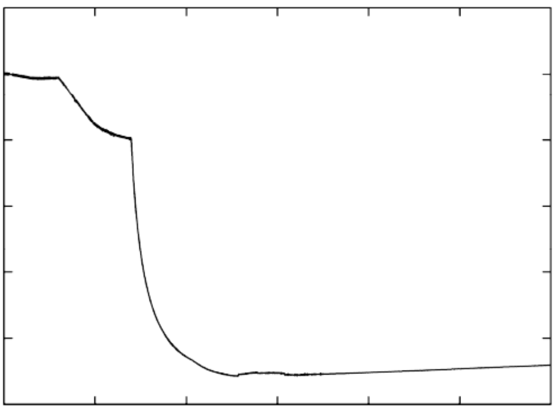

Fig. 6. RCS flow over time.

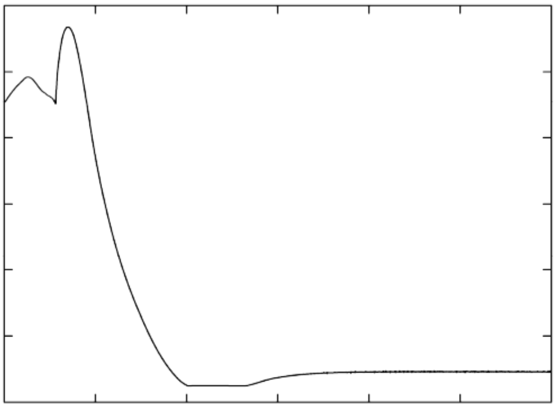

Fig. 7. SG pressure over time.

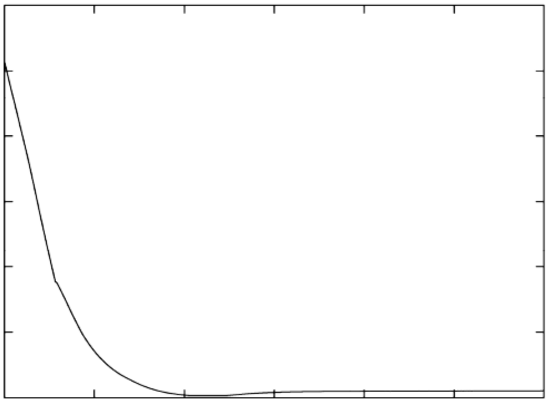

Fig. 8. SG mass over time.

3. Conclusions

As a result of the ATWS analysis of Kori Unit 2, the maximum pressure of the reactor coolant system is 2,756 psia, which satisfies the acceptance criteria of 3,200 psig, and the generic analysis performed by WH is still valid.

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