ATWS Analysis for Replacement of Steam Generator using MARS-KS

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1. Introduction

An anticipated transient accompanied by a failure in the Reactor Trip System (RTS) to shut down the reactor is defined as an Anticipated Transient Without Scram (ATWS). Under certain postulated conditions, the ATWS could lead to Reactor Coolant system (RCS) pressure boundary fracture and/or core damage. [1] These ATWS for the safety of nuclear power plants are to be evaluated items. The ATWS are affected by many factors such as reactor power, reactivity feedback and pressure relief capacity. Since replacement of steam generators for Ulchin 1&2 reactors to promote safety, improve operational reliability and expand the utilization of nuclear power plants has an effect on ATWS, the transient analysis as the change of MTC (moderator temperature coefficient) and capability of steam release were performed for Ulchin 1&2 power plant. [1]

2. Methods and Results

A thermal hydraulic analysis code, MARS-KS was used to simulate a ATWS scenario, loss of feedwater without scram, in a Ulchin units 1&2 reactors. Ulchin 1&2 have a unique feature for pressure relief; three Pilot-Operated Safety Relief Valves (POSRV), and steam dump capacity (85 percent of rated turbine flow) much larger than Westinghouse designed three-loop plants (70 percent of rated turbine flow). Table 1 shows the major design parameters of the plants and Figure 1 shows the input nodalization. The primary and secondary systems in the code input deck consist of hydrodynamic volumes including branch, annulus, and pipe, junctions, and heat structures. The various component models such as pumps and valves, and the instrumentation actuation logic and setpoints are also utilized to simulate the transient behavior of the systems. [1]

Table I: Nominal operating condition of Ulchin 1&2 at 100 % full power

Parameter		Ulchin 1&2
Core	Power (MWt)	2775
	MTC upper limit	0
RCS	Total vol. including Prz. and surge line (m^3)	258.83
	Nominal pressure	15.51
	Nominal flow (m^3/h)	65962
	Nominal reactor vessel inlet temp. (°C)	285.4
	Nominal reactor vessel outlet temp. (°C)	323.7
Pressurizer	Total volume of Prz. and surge line (m^3)	41.54
	Nominal steam volume (m^3)	14.54
	POSRV capacity (kg/sec)× No.	47.22×2
	Max. spray rate (kasg/sec)	40.00
Secondary system	SG type	51B
	Design SG tube plugging (%)	10
	Nominal steam pressure (MPa)	5.52
	Nominal steam flow (kg/sec)	504.4/SG
	Nominal feed temp. (°C)	219.4
	Auxiliary feed flow capacity (m^3/h)	165.0/SG
	Rated steam dump capacity (%)	85

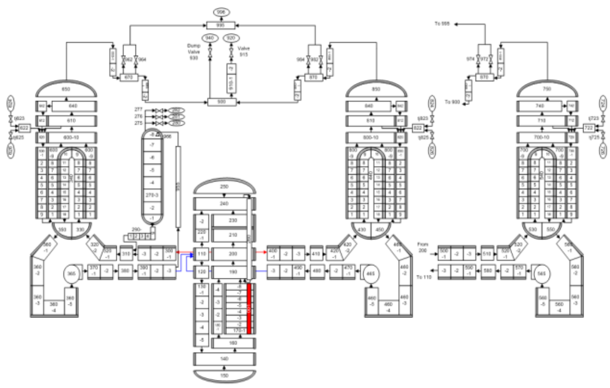

Fig. 1. Nodalization of Ulchin 1&2

The ATWS transient analyses were performed with 99% MTC in the specific cycle designs, and the parametric sensitivity to RCS overpressure was investigated. The results showed that the ATWS overpressure depends largely on the MTC, RCS pressure relief and heat removal capacity of the steam generator. It was also found that the AMSAC(ATWS Mitigating System Actuation Circuitry) in Ulchin units 1&2 was somewhat effective in mitigating the pressure transient. The fraction of MTC Unfavorable Exposure Time(UET) or MTC Over Pressure Factor(OPF) is defined as the fraction that the MTC-OPF in Ulchin units 1&2 is expected to be relatively higher than the other Westinghouse designed plants. [1]

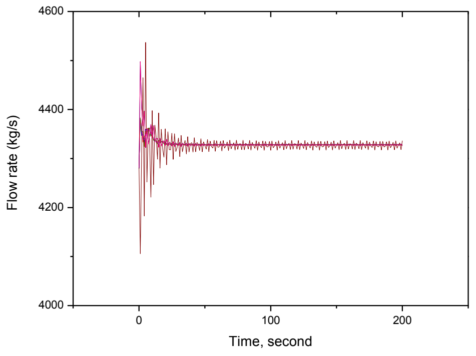

Fig. 2. Steady state of mass flow rates for Ulchin 1&2

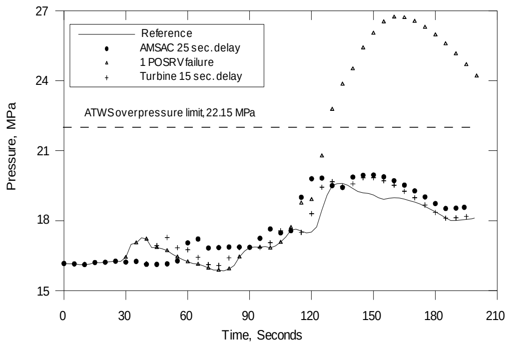

Fig. 3. RCS pressure for Ulchin 1&2 LOF-ATWS

3. Conclusion

To ensure the safety of nuclear power plants, ATWS analysis was performed. MARS-KS (domestic regulatory code) was used. This study was carried out by considering various factors which are affected the RCS pressure. The effect of pressure was also evaluated. The effect of AMSAC was also evaluated. Increased RCS pressure was assessed according to the MTC, pressure relief capacity, depends on the performance of steam generators. Ulchin 1&2 for the effect of the steam generator replacement were assessed. Evaluation of the performance of these ATWS is very important for the nuclear power plant safety.

4. Recommendation & Further Study

A further direction of this study will be to provide the effect of nuclear power plant due to the change of geometry by changing Inconel, the effect of release valve by changing S/G and etc.

5. Acknowledgement

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