Development of SPACE Input Model of OPR 1000 Plant for Non-LOCA Analysis

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1. Introduction

Non-LOCA safety analysis methodology of APR 1400 plant has been developed using SPACE code (Safety & Performance Analysis CodE for nuclear power plants), designed to predict the thermal-hydraulic response of nuclear steam supply system to anticipated transients and postulated accidents. Several nodalization have been tested to find the optimized configuration for Non-LOCA analysis.

In this study, the steady-state initialization is performed to evaluate the capability of SPACE code. Because the initialization is important to analysis the Non-LOCA behavior, a steady-state conditions are compared with design data of OPR 1000 plant for major plant parameters such as core power, core inlet temperature, and primary pressure. In addition, comparative analysis is performed according to different nodalizations of OPR 1000 based on that of APR 1400 for Non-LOCA analysis.

2. Modeling information

Figure 1 shows the schematic nodalization diagram for the OPR1000. The component and heat structure values are adopted to simulate system model for Ulchin 3/4 plant [2]. As shown in this figure, OPR1000 has a reactor pressure vessel, two hot legs, four cold legs, a pressurizer, four reactor coolant pumps (RCPs), and two steam generators (SGs). The plant is modeled with 233 fluid cells, 303 connections between cells and 231 heat structures. In order to make steady-state base input deck for OPR1000 using SPACE code, the input data of MARS-KS is used [5].

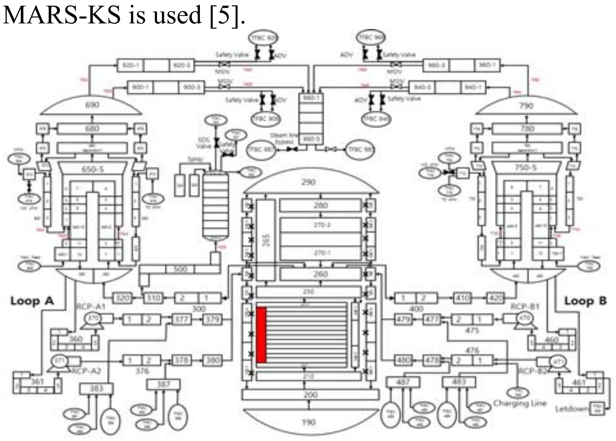


Fig. 1 Nodalization diagram of OPR1000

3. Results and analysis

3.1 Steady-state initialization test

In order to confirm the validity of the input data at steady-state condition, major parameters are compared with 100 % power operation condition for a steady-state. The RCS cold leg temperature is controlled by adjusting the steam generator pressure. The pressurizer pressure and initial SG level are set as 155.1 bar (2,250 psia) and 44% NR. The values are updated until the steady-state condition is obtained.

The major parameters are pressurizer pressure and level, reactor coolant flow rate, core inlet temperature, and steam generator level. The level of steam generator and pressurizer are obtained through control systems. The reactor coolant flow rate is initialized by using the pump control. The core inlet temperature reaches to steady-state using secondary pressure control.

Table 1 shows the design data and calculated values of the major parameters at steady-state condition. As shown in Table 1, the results of SPACE code agree well with the design data.

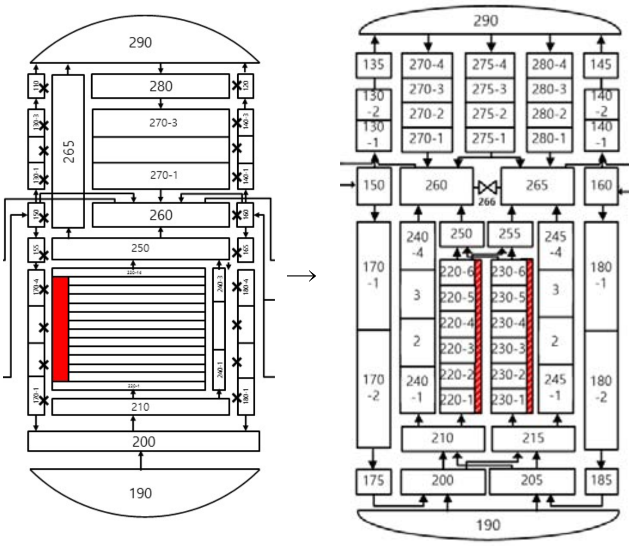
Table 1 Design data vs. Calculation results

	Parameter	Design	SPACE
Reactor Vessel	Core power [MWt]	2815	2815
	Core bypass flow [kg/s]	36.5	36.6
	Core flow [kg/s]	14944.8	14955.0
Primary side	Cold leg flow rate [kg/s]	3824.4	3827.3
	Hot leg temp. [K]	600.48	600.7
	Cold leg temp. [K]	568.98	569.4
	Pressurizer pressure [bar]	155.1	155.1
	Pressurizer water level [%]	52.6	52.6
	Pump head [m]	102.7	104.4
	Pump torque [Nm]	38497.1	37789.5
Secondary side	Stream flow rate [kg/s]	801.3	802.8
	Stream pressure [bar]	73.774	73.782
	SG water level [%]	44	44.0

3.2 Modification of reactor vessel nodalization

Figure 2 shows the nodalization diagram of modified reactor vessel. The core region is split into two separate channels to simulate asymmetric transients like stream line break [3]. The heat structures of core are split into two part and apply to component #220, #230. The lower head (#200) is divided into four parts, and the core inlet (#210) is changed to two parts to make symmetry. The reactor vessel inlet annulus (#155, #165) and downcomer (#170, #180) are combined to component #170, #180. The core bypass (#240) and the upper plenum (#250, #260) are changed to symmetric configuration. The upper guide structure assembly (UGSA, #270), upper portion of the UGSA (#280), and guide tube (#265) are combined and split in to three pipe components.

In order to examine the validity of modified nodalization, heat flux of reactor coolant system is compared to that of previous value using cold leg flow, hot leg temperature, cold leg temperature. The difference between before and after modification case is about 1 percent, which indicate that the modified reactor vessel nodalization is reasonable and consistent with previous input models.

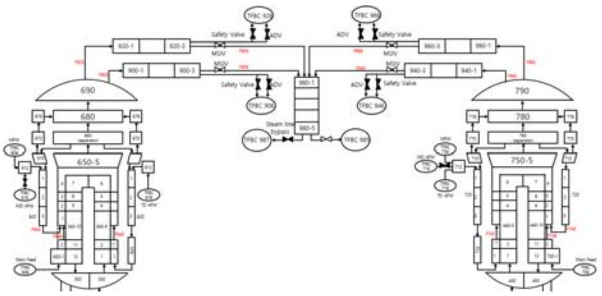


[Before] [After modification]

Fig. 2 Nodalization diagram of modified reactor vessel

3.3 Modification of secondary side nodalization

Figure 3 represents the modified nodalization of secondary side. The steam generator downcomer annulus (#620, #720) and downcomer annulus of hot leg side (#624, #724) is split into four parts and two parts, respectively. The main steam line system is modified with reference to nodalization of APR 1400. The part of main steam line between the MSIVs and turbine stop valve remain the same in terms of volume, flow area, and hydraulic diameter. The heat flux difference of reactor coolant system between reactor vessel modification case and additional secondary side modification case is within less than 1 percent.



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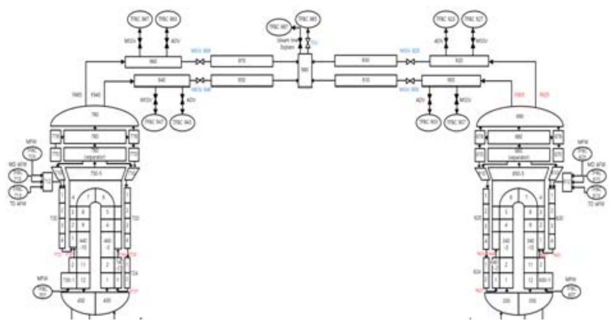

[After modification]

Fig. 3 Nodalization diagram of modified secondary side

4. Conclusions

The steady-state initialization was performed and comparative studies were performed according to nodalization of OPR 1000 for Non-LOCA safety analysis using SPACE code. The calculation results were compared with design data of OPR 1000 for evaluate the capability of SPACE code. The major plant parameters and heat flux of reactor coolant system were compared to examine the validity of modified nodalization.

ACKNOWLEDGMENTs

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