Modeling of Control System and Protection System for the SKN1 Simulator V&V

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1. Introduction

The ShinKori nuclear power plant unit 1(SKN1) simulator has been developed, and the thermal-hydraulic model of the nuclear steam supply system has been developed using the RELAP5 R/T code, which is a real-time version of RELAP5/MOD3.2 [1]. To evaluate the model performance, the validation and verification (V&V) calculations are needed using the reference code.

In this study, the RELAP5 R/T models of control system and protection system for the SKN1 were developed to simulate various NPP transient conditions provided by ANSI/ANS-3.5 [2]. And the test results of manual reactor trip from an initial condition of 100% power steady state were presented. The transient results were compared with those of the RELAP5/MOD3.3.

2. Modeling of Control System and Protection System

The RELAP5/MOD3.2 input deck for the Ulchin 3&4 NPP(UCN3&4) was used as a base input model, and the difference between the SKN1 and the UCN3&4 was analyzed. We found no difference in thermal-hydraulic component, only the aux. feedwater flow rate was 50% of the reference plant.

The control systems perform automatic mitigation of transient events which can occur during power operation of NPP, and the protection systems shutdown the reactor promptly and surely to prevent exceeding safety limits when the anticipated transient events occur. However, the SKN1 Simulator uses those models made by simulation tools. Thus the modeling of those systems using RELAP5 R/T code is required for the V&V.

The control systems are composed of Pressurizer Pressure Control System(PPCS), Pressurizer Level Control System(PLCS), Reactor Regulating System (RRS), FeedWater Control System(FWCS), Steam Bypass Control System(SBCS) and Reactor Power Cutback System(RPCS). The Control Element Deriving Mechanism Control System(CEDMCS) was modeled as the same system to RRS.

The reactor protection systems are composed of reactor trip system, turbine trip system and steam generator(SG) isolation system. These systems have the detailed trip signals respectively, and the time delay was considered in the modeling.

The detailed model descriptions of the control system and protection system could be found in the previous study [3]. Fig. 1 shows the RELAP5 R/T nodalization for the SKN1 which includes all the above control systems.

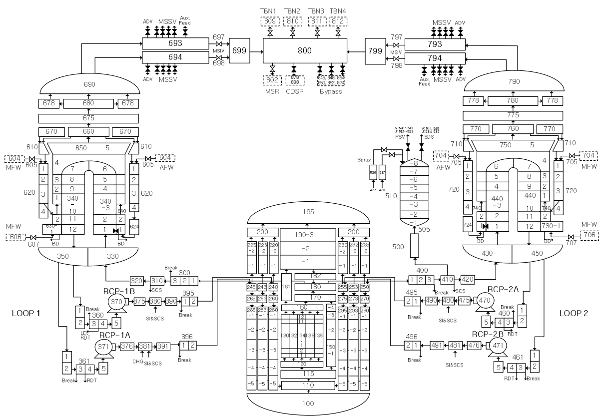

Fig. 1. RELAP5 R/T nodalization for the SKN1

3. Steady State Test

The steady state of full power was calculated in the auto-actuating conditions of the control system and protection system using the RELAP5 R/T and the RELAP5/MOD3.2. We modified such parameters as friction factor and heat transfer rate in the RELAP5 R/T input deck to meet the allowed error.

Table 1 shows the parameters calculated at the steady state of full power. The initial thermal hydraulic conditions were in good agreement with design ones within the allowed errors.

In addition, we calculate the steady state at the power of 75%, 50%, 25% and no load condition(2%). The calculated reactor coolant temperature and the pressurizer level at various powers are shown in Table 2. The parameters stabilized at the programmed values.

Table 1: Comparison of steady state parameters

Parameter	Unit	Design	R5/RT	R5/Mod3.3
Rx power	MWt	2815	2815	2815
HL Temp.	K	600.48	600.35	600.21
CL Temp.	K	568.98	569.29	569.1
PRZ P	bar	155	155.13	155.13
PRZ level	%	52.6	52.6	52.6
S/G P	bar	73.77	73.48	73.44
S/G level	%	44	44	44
CL flow	Kg/s	3827.7	3848.2	3845.6
MF flow	kg/s	801.1	800.9	800.6
S/G Stm. flow	kg/s	400.55	400.51	400.29

Table 2: Steady state parameters at various powers

Condition	Parameters	R5/RT	R5/Mod3.3
75%	HL Temp(K)	592.77	592.85
	CL Temp(K)	568.96	569.03
	PZR Level(%)	47.83	47.92
50%	HL Temp(K)	584.76	584.94
	CL Temp(K)	568.61	568.83
	PZR Level(%)	41.52	41.72
25%	HL Temp(K)	576.74	576.71
	CL Temp(K)	568.63	568.58
	PZR Level(%)	35.38	35.29
no load (2%)	HL Temp(K)	569.25	569.01
	CL Temp(K)	568.73	568.48
	PZR Level(%)	33.16	35.57

4. Transient Performance Test

We performed 15 sets of transients from an initial condition of approximately 100% power with no operator follow-up action. The test sets are as follows.

- Manual reactor trip
- Simultaneous trip of all feedwater pumps
- Simultaneous closure of all MSIVs
- Simultaneous trip of all reactor coolant pumps
- Trip of any single reactor coolant pump
- Main turbine trip
- Maximum rate power ramp from 100% down to approximately 75 % and back up to 100%
- Maximum size reactor coolant system rupture combined with loss of all offsite power
- Maximum size unisolable main steam line rupture
- Slow primary system depressurization to saturated condition with PRV stuck open
- Maximum design load rejection.
- Loss of all AC power
- Loss of all feedwater and ATWS
- SG tube rupture(3 tubes)
- Small loss of coolant accident

The results of manual reactor trip test using RELAP5 R/T and RELAP5/MOD3.3 are shown in Fig. 2. The thermal hydraulic parameters after the manual reactor trip were stabilized to designed setpoints by auto-actuation of control systems.

5. Conclusions

The models of control system and protection system for the SKN1 were developed using RELAP5 R/T and RELAP5/MOD3.3. The steady states at various powers were calculated in the auto-actuating conditions of the control system and protection system. The initial thermal hydraulic conditions were in good agreement with design ones within the allowed errors.

15 sets of transients from an initial condition of approximately 100% power were performed with no operator follow-up action. The results of manual reactor trip test showed reasonable plant responses as the thermal hydraulic parameters after the manual reactor trip were stabilized to designed setpoints by auto-actuation of control systems.

By developing the models of control system and protection system for the SKN1 using RELAP5 R/T and RELAP5/MOD3.3, the V&V materials for the SKN1 simulator thermal hydraulic model could be provided.

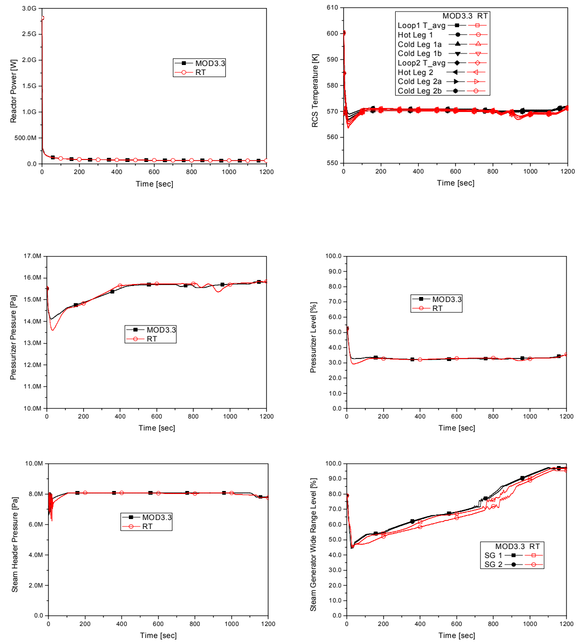

Fig. 2. Results of manual reactor trip test

REFERENCES

[1] Do Hyun Hwang, et al., "Interface between Core/TH Model and Simulator for OPR1000", Transactions of the KNS Spring Meeting, Cheju, Korea, May 22, 2009.

[2] "American National Standard for Nuclear Power Plant Simulators for Use in Operator Training and License Examination", ANSI/ANS 3.5, ANS, 1998.

[3] Jeong Kwan Suh, et al., "Modeling of Protection System and Control System for the APR1400 Using RELAP5 R/T", Transactions of the KNS Autumn Meeting, Gyeongju, Korea, Nov. 2-3, 2006.

[4] The RELAP5-3D Code Development Team, "RELAP5-3D Code Manual", INEEL-EXT-98-00834, Revision 2.2, INEEL, October 2003.