Development of APR1400 Nuclear Plant Analyzer using RETRAN-3D

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1. Introduction

A Nuclear Plant Analyzer (NPA) has been developed to simulate the thermal-hydraulic phenomena for Pressurized Water Reactors (PWRs). In this study, the RETRAN input model is developed for APR1400 NPA. It is used for the small-LOCA with broken pipe below 3inch and Non-LOCA transient conditions. Control/Protection systems are established to accept the operator's action related to the NSSS during the accidents. In order to confirm the validity of the input data, the calculations are performed for a steady-state at 100% power operation conditions, manual reactor trip and RCP trip.

2. Development of APR1400 NPA

The RETRAN input model is developed for APR1400 NPA. It is used for the small-LOCA with broken pipe below 3inch and Non-LOCA transient conditions.

2.1 Physical input model

The model includes 196 Trip control cards, 125 control volumes, 186 flow junctions, 4 reactor coolant pumps, 86 valves, 39 general data tables, 52 fill junctions , point kinetics data for reactor core and 18 heat conductors. Figure 1 shows the nodalization of APR1400.

o Thermal-hydraulic model

- Reactor coolant system(RCS)

Reactor and reactor coolant system(RCS) issimulated with reactor vessel, cold legs for each loop, hot leg for each loop, reactor coolant pumps, steam generator primary side, pressurize, surge line, reactor drain tank, spray line and direct vessel injection line(DVI).

- Secondary system

Secondary system is simulated with steam generator secondary side, main steam isolation valves(MSIV) and main steam line (each steam generator, upstream of MSIV, common line up to turbine governor valve). Other components simulated are pressurizer pilot operated safety and relief valve(POSRV), safety depressurization system (SDS), steam line safety valves, steam line power-operated relief valves, dump valves and governor valves.

o Core model

Point kinetics model is used in the reactor core which is applied with the characteristics of KSNP. The reactor core data for APR1400 will be applied later.

o Heat conductor model

Heat conductor model simulated are fuel rod, steam generator tube, pressurizer heater, reactor vessel and its internal structure, hot leg, cold leg, pressurizer vessel, steam generator vessel and other components.

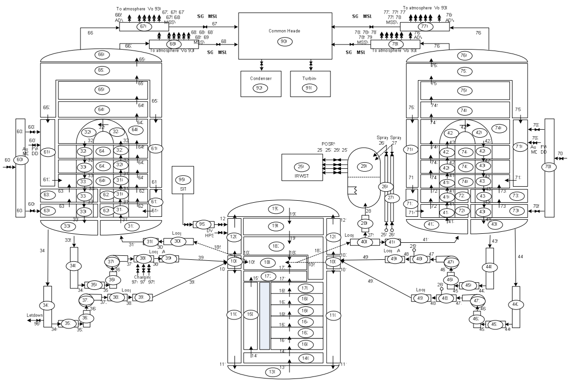


Figure 1. RETRAN nodalization of APR1400

2.2 Control /protection system

Control/protection system input is necessary to simulate the transient phenomena, operators' action and plant behavior in the actual way. The control/protection systems are modeled properly according to the scenarios. Figure 2 shows the logic diagram of feedwater control system. Some setpoints in control/protection system are applied with the characteristics of KSNP because APR1400 plant data is not available yet. The data for APR1400 will be applied later.

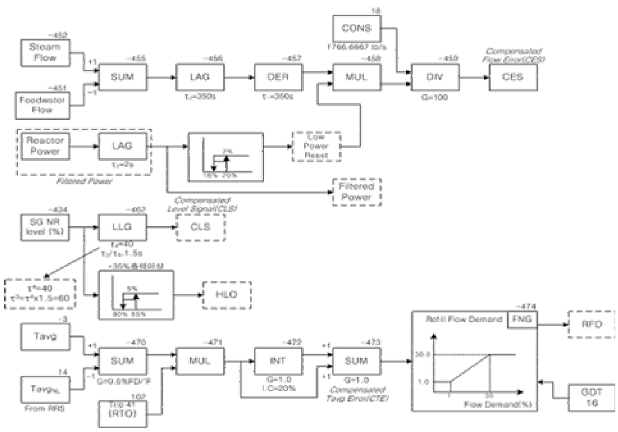
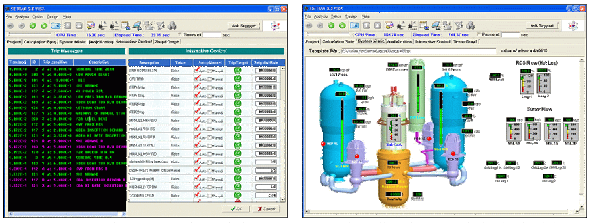


Figure 2. Feedwater control system

2.3 Graphic user interface of NPA

The NPA consists of the pre-processor, execution control and visualization modules. The NPA graphic user interface(GUI) of APR1400 is designed for convenient use. Figure 3 shows the modules of APR1400 NPA.



Execution control module   Visualization module (Mimic)

Figure 3. Modules of APR1400 NPA

3. Performance Test

In order to confirm the validity of the input data, calculations are performed for a steady-state at 100% power operation conditions with manual reactor trip and RCP trip.

3.1 Steady- state condition

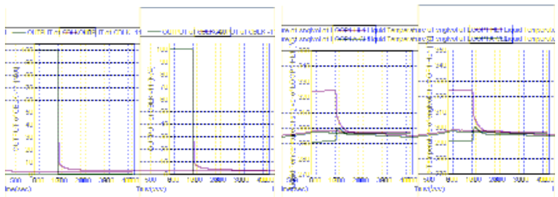
The steady-state of full power is calculated under the auto-actuation conditions of protection and control systems. Table 1 shows the initial thermal hydraulic conditions in good agreement with design data.

Table 1 . Steady-state calculations at 100% power operation conditions

	Parameter	Design	Calculation	Error(%)	Reference
Reactor	Core Power[Mwt]	3983	3983	0.0000	PSAR, Chapter 4, Table 4.4-1
Primary Side	Cold Leg 1A Flowrate [kg/sec]	5247.6	5247.6	0.0000	PSAR, Chapter 4, Table 4.4-7
	Hot Leg Temp.[◦C]	323.9	324.0	0.0309	PSAR, Chapter 4, Table 4.4-1
	Cold Leg Temp.[◦C]	290.6	291.3	0.2409	PSAR, Chapter 4, Table 4.4-1
	PZR Level(%)	52.6	52.7	0.1901	FS-DD012, App.A,pA102
	PZR Pressure[kg/cm^2]	158.2	157.8	-0.2528	PSAR, Chapter 4, Table 4.4-1
	Pump Speed[rpm]	1190	1193	0.2521	FS-DD012, App.A,pA10
Secondary Side	Downcomer FW Flowrate [kg/sec]	113.4	113.1	-0.2646	PSAR, Chapter 10, Table 10.1-1
	Economizer FW Flowrate [kg/sec]	1019.7	1017.6	-0.2059	PSAR, Chapter 10, Table 10.1-1
	Steam Flowrate (each line) [kg/sec]	565.4	565.4	0.0000	PSAR, Chapter 10, Table 10.1-1
	Steam Dome Pressure [kgf/cm^2]	70.3	70.5	0.2845	PSAR, Chapter 10, Table 10.1-1

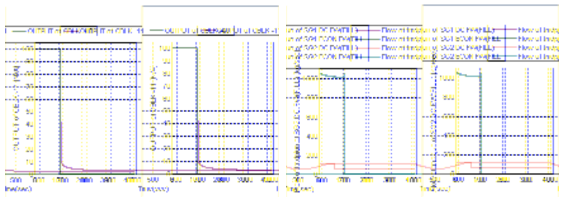
3.2 Transient analysis

The thermal hydraulic parameters after manual reactor trip and RCP trip are stabilized to designe setpoints by auto-actuation of control and protection systems. Figures 4 and 5 show the calculation results.



Rx power & TBN load   Hot leg & cold leg temperature

Figure 4. Results of manual reactor trip test



Rx power & TBN load   RCP speed

Figure 5. Results of RCP trip test

4. Conclusion

The APR1400 NPA is developed to simulate thermal-hydraulic transient phenomena in APR1400. It can be used for various transient analyses as well as accident analysis. The RETRAN input model is developed. Performance test calculations are done. The results of performance test calculations are reasonable and consistent with those of other best-estimate calculations. It is expected that the APR1400 NPA can aid operators to understand plant transient analysis, improve the APR1400 safety and operation through nuclear safety evaluation and resolve the safety issues.

REFERENCES

[1] APR1400 Preliminary Safety Analysis Report, KHNP
[2] APR1400 Nuclear Plant Functional Diagram, KHNP
[3] Uljin-3&4 Nuclear Plant PLS (Precautions, Limitations and Setpoints), KHNP
[4] Fluid Systems and Engineering and Architect Engineer design data for plant safety, containment and performance analyses for KNGR, N0797-FS-DD012 Rev.01
[5] Fluid Systems and Component Engineering Design Data for Plant Safety, Containment and Performance Analysis for KNGR, N0797-SA-DD140 Rev.0, 1999
[6] Design Data for Safety Analysis for KNGR, N0797-SA-DD140, Rev.00, 1999
[7] Design Data for Reactor Internals for KNGR (Water Volumes, Masses, Thicknesses, Surface Areas, and Elevations, N0797-ME-DD210-00, Rev.01, 1998