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The Establishment of RELAP5/SNAP Model for Kuosheng Nuclear Power Plant

C. Shih, J. R. Wang, H. C. Chang, S. W. Chen, S. C. Chiang, T. Y. Yu

***Abstract*—**After the measurement uncertainty recapture (MUR) power uprates, Kuosheng nuclear power plant (NPP) was uprated the power from 2894 MWt to 2943 MWt. For power upgrade, several codes (e.g., TRACE, RELAP5, etc.) were applied to assess the safety of Kuosheng NPP. Hence, the main work of this research is to establish a RELAP5/MOD3.3 model of Kuosheng NPP with SNAP interface. The establishment of RELAP5/SNAP model was referred to the FSAR, training documents, and TRACE model which has been developed and verified before. After completing the model establishment, the startup test scenarios would be applied to the RELAP5/SNAP model. With comparing the startup test data and TRACE analysis results, the applicability of RELAP5/SNAP model would be assessed.

***Keywords*—**RELAP5, TRACE, SNAP, BWR.

I. Introduction

KUOSHENG NPP is located on the northern coast of Taiwan. Its nuclear steam supply system is a type of BWR/6 designed and built by General Electric on a twin unit concept. Each unit includes two loops of recirculation piping and four main steam lines, with the thermal rated power of 2894 MWt. After the project of MUR for Kuosheng NPP, the operating power is 2943 MWt [1], [2]. To uprate the power, the assessments of NPP transients should be analyzed. In the past, a TRACE model of Kuosheng NPP has been developed. The analysis and simulation of the TRACE model for the startup tests and hypothetical transients has been done. In this research, a RELAP5/MOD3.3 model of Kuosheng NPP with SNAP interface is developed referred to FSAR [1], training documents [2], and the TRACE model [3]-[5].

RELAP5/MOD3.3 Patch04 code, which was developed for light water reactor (LWR) transient analysis at Idaho National Engineering Laboratory (INEL) for U.S. NRC, is applied in this research. This code is often performed to support rulemaking, licensing audit calculations, evaluation of accident, mitigation strategies, evaluation of operator guidelines, and experiment planning analysis [6]. Same as other thermal hydraulic analysis codes, RELAP5/MOD3.3 is based on nonhomogeneous and nonequilibrium model for the two-phase system. However, the calculations in this code will be solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. It can produce accurate transient analysis results in relatively short time, which means large amounts of sensitivity or uncertainty analysis might be possible.

SNAP is an interface of NPP analysis codes which is developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researchers comprehend the whole power plant and control system more easily. Due to these advantages, the RELAP5/MOD3.3 model of Kuosheng NPP was developed with SNAP interface.

Based on the above description, to ensure the applicability of the RELAP5/SNAP model, the startup tests including feedwater pumps trip (FWPT) and load rejection with bypass (LRWB) would be analyzed in this research.

II. Methodology

Different from typical thermal hydraulic model establishment with ASCII file, the RELAP model in this research was developed in the SNAP interface. With SNAP interface, the components of RELAP5/SNAP model are visible as shown in Fig. 1. As a result, the users could set up component parameters and nodding diagram at same time. In this model, the situation of NSSS in transient was mainly concerned. Hence, the turbines and feedwater pumps of Kuosheng NPP were assumed to be boundaries which were simulated by components Time dependent volume (TMDVOL). Four main steam pipe lines, which were consistent with the configuration of Kuosheng NPP, were developed. On these pipe lines, three important valves including main steam line isolation valves, turbine stop valves and turbine control valves were established. Further, there are totally 16 safety/relief valves connected on the main steam pipe lines. All the opening and closing setpoints were also set up according to the arrangement of Kuosheng NPP.

The reactor vessel was developed by several kinds of components including Branch, Pipe, Single junction and single volume. Four pipes which were established to simulate fuel assemblies were connected to heat structures inside the reactor vessel. Source data of these heat structures were referred to the total reactor power. To simplified the model and save the computational time, Point Kinetics and Separable feedback types were chosen for the reactor kinetics. Two recirculation loops and recirculation pumps were set up according to the configuration of the NPP. Further, there are two control valves on two recirculation loops respectively to adjust the recirculation flow rate. 20 jet pumps in the NPP were merged

C. Shih, J. R. Wang, H.C. Chang and S. W. Chen are with the Institute of Nuclear Engineering and Science, National Tsing Hua University; Nuclear and New Energy Education and Research Foundation; Nuclear Science and Technology Development Center, R.O.C., Taiwan (e-mail: ckshih@ess.nthu.edu.tw, jongrongwang@gmail.com, ms9831117@iner.gov.tw, chensw@mx.nthu.edu.tw).

S. C. Chiang and T. Y. Yu are with Department of Nuclear Safety, Taiwan Power Company, R.O.C., Taiwan (e-mail: u805630@taipower.com.tw, u069601@taipower.com.tw).

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into two jet pump components to save the computational time.

In addition, because of developing analysis model with SNAP interface, analysis data results could be transferred into animations which could illustrate the situation of NPP during transients more easily and clearly (shown in Fig. 2).

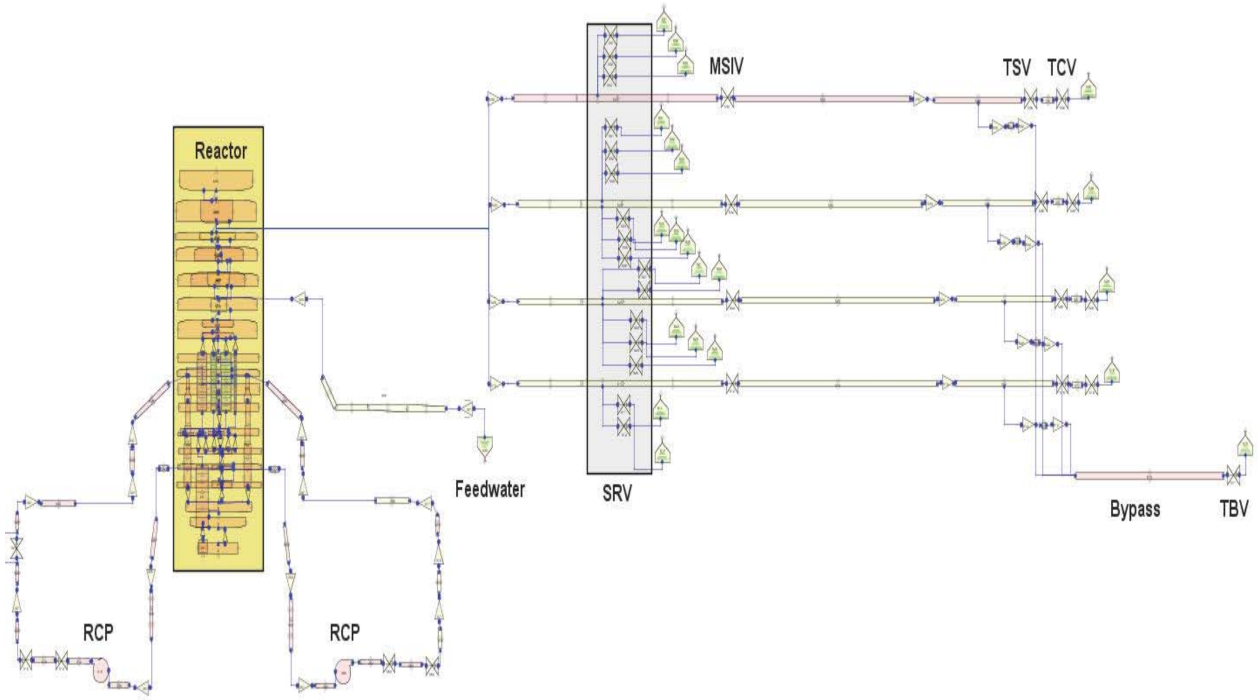

Fig. 1 RELAP5/SNAP model

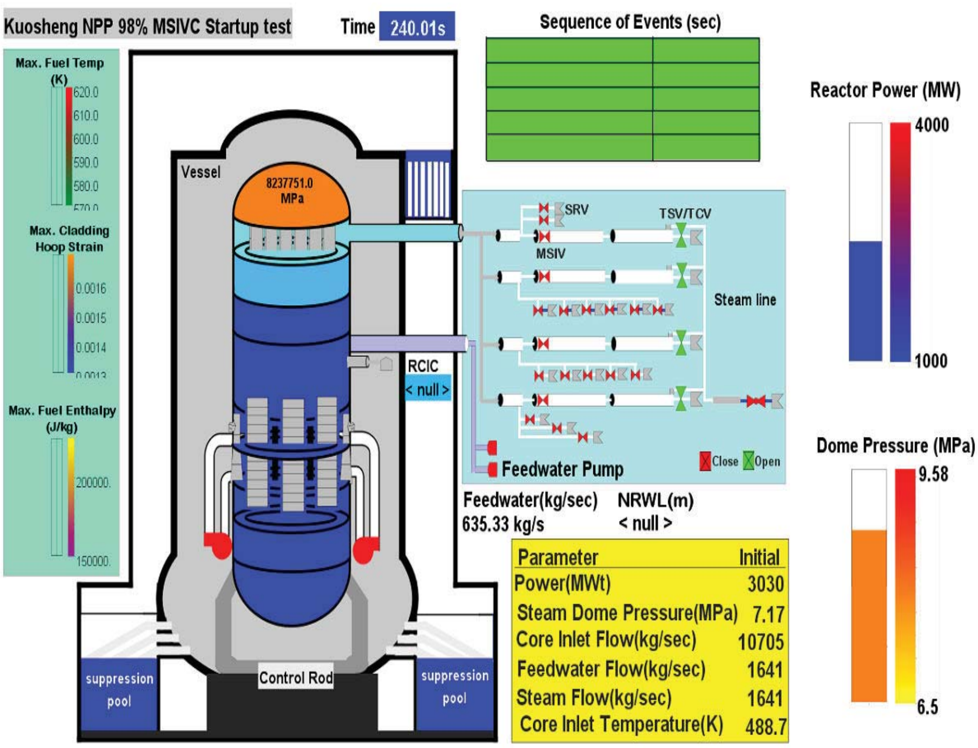

Fig. 2 Animation model

III. Results

The LRWB test was held in November 11th, 1981. The operator triggered the breaker of the main generator as the reactor was operated in 100% power. Then, the turbine control valves were closed, and turbine bypass valve was opened, which caused the transient start and reactor scram. The objective of this test is to ensure whether the turbine control valves could be closed immediately or not. In addition, the responsibility of the turbine bypass valve, safety/relief valves and the reactor protection system should also be assessed. The acceptance criteria are shown as follows:

- Turbine bypass valve should open in 0.1 second after the turbine control valves closure. Further, the bypass valve should reach to 80% open of the design capacity in 0.3 second.
- The feedwater setting should avoid the main steam pipe lines being flooded.
- The feedwater control system should maintain the water level which would not trigger the MSIVs closure.
- The dome pressure increasing should not be higher than 25 psi in 30 seconds and the heat flux ratio should not be over than 2%.

Due to the stability, a 210-second steady state was performed for ensuring that all the parameters matched the operating conditions as shown in Table I. However, to compare the results with the startup test and the TRACE model data more easily, the steady state will be eliminated in the figures.

At 210.2 second, the TCVs started to close. After closure of TCVs for 0.011 second, the TBV start to close. Further, when the dome pressure was lower than 940 psia (6.48 MPa), the TBV would start to close. The recirculation pumps were tripped as the dome pressure reached to 1134.7 psia (7.82 MPa). Once the generator was tripped, the turbine control valves started to close and fully close in 0.1 second. As the TCVs started to close, the steam flow decreased immediately as shown in Fig. 3. Fig. 4 shows great consistency between RELAP5 model and startup test result. As the TCVs started to close, the reactor scram signal was initiated. Due to the large negative reactivity feedback, the power decreased rapidly at first second as shown in Fig. 4. Nonetheless, the reactor core still generated some decay heat by the fission product. Hence, the steam was still produced inside the reactor vessel. Even though the turbine bypass valve started to open at 0.1 second according to the acceptance criteria mentioned above, the dome pressure still increased as shown in Fig. 5.

The FWPT was held in November 6th, 1981. The transient started as the operator tripped one of three operating feedwater pumps at 94% power. The objective of this test is to examine whether the flow control valves (FCV) run back in the recirculation loops could lower the core power or not when a feedwater pump failed. If the core power could be lowered by the reduction of the recirculation flow, the water level would not be below L3, which was the setpoint of the reactor scram.

Table II shows the event sequence of this transient. Same as previous two cases, a 210-second steady state analysis was performed to ensure all the parameters matched the operating conditions. At 214.9 second, one of the three feedwater pumps was tripped. Further, the L4 water level signal would initiate FCV run back to lower the core power which would maintain the water level. According to the configuration of KS NPP, the FCV area of recirculation loop A would decline from 80% to 50%, and the FCV area of recirculation loop B would decline from 69% to 46%.

Fig. 6 shows the comparisons of feedwater flow among startup test, TRACE, and RELAP models. At 4.9 seconds, one of three feedwater pumps was tripped. The feedwater flow decreased for about 5 seconds. About 17 seconds later, as shown in Fig. 7, the NRWL reached to 0.848 m, which is L4 for Kuosheng NPP. The recirculation flow control valves ran back. Hence, core inlet flow rate decreased to 80%, as shown in Fig. 8. Decreasing of the core flow would increase the void fraction, which means that the density reactivity feedback would also decrease. The core power dropped down at about 17 seconds as shown in Fig. 9. In addition, because the water level declination in TRACE model was a little slower than the other two cases, it can be noticed that the core power decreasing in the TRACE model was later than that in the other two data results.

TABLE I
EVENT SEQUENCE OF 100% LRWB TRANSIENT

Event(sec)	Test data	RELAP5
TCV Start to Close	0.2	210.2
Reactor Scram	0.236	210.29
BPV Fully Open	0.329	210.32
TCV Fully Close	0.394	210.35
Peak Dome Pressure Time (value)	3.9 (7.43 MPa)	212.84 (7.55 MPa)
End of Analysis	-	230

TABLE II
EVENT SEQUENCE OF 94% FWPT TRANSIENT

Event (sec.)	Test data	RELAP5
Feedwater pump Trip	4.9	214.9
L4 Signal	15.1	217.01
Minimum Power Value	18.5 (57.1%)	218.90 (54.2%)
Minimum Core Flow	19.4 (79.3%)	218.31 (78.76%)
END of Analysis	-	240

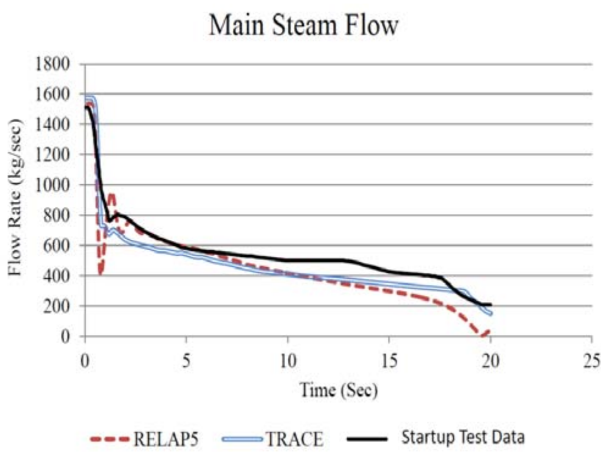

Fig. 3 Steam flow variation during the 100% LRWB transient

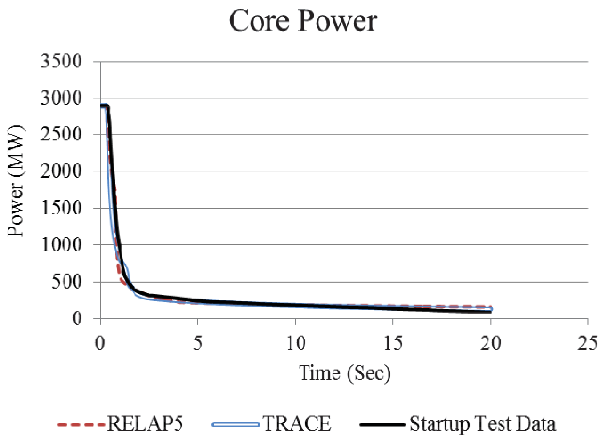

Fig. 4 Core power variation during the 100% LRWB transient

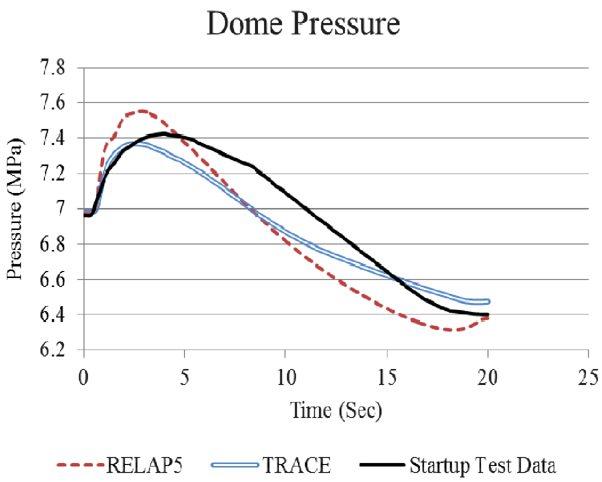

Fig. 5 Dome pressure variation during the 100% LRWB transient

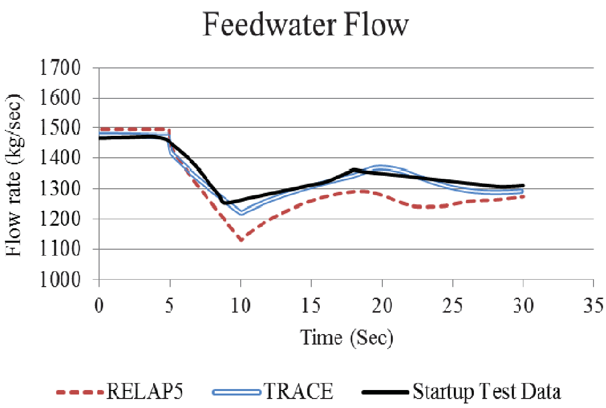

Fig. 6 Feedwater flow variation during the 94% FWPT transient

Due to the declination of feedwater flow rate and water level, the feedwater control system adjusted the other two available feedwater junctions and increased the flow rate as shown in Fig. 6. However, at the end of the transient, core power of computational data results was different from that of startup test since the water level might be controlled in startup test as shown in Fig. 7. The density reactivity feedback would be different from the computational models. Nonetheless, due to lack of some other transient scenario or information, the water level followed the computational results rather than matched the startup test data in both TRACE and RELAP5 models.

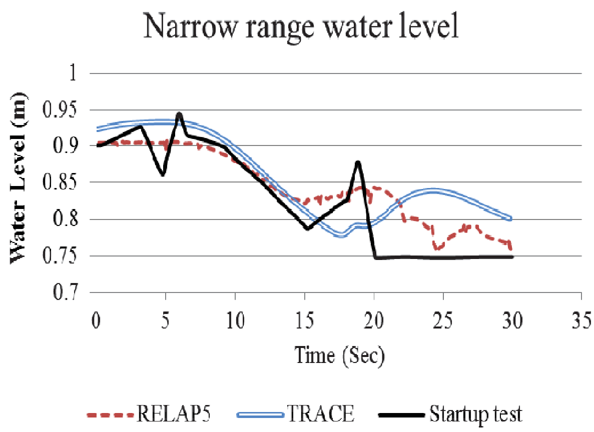

Fig. 7 Water level variation during the 94% FWPT transient

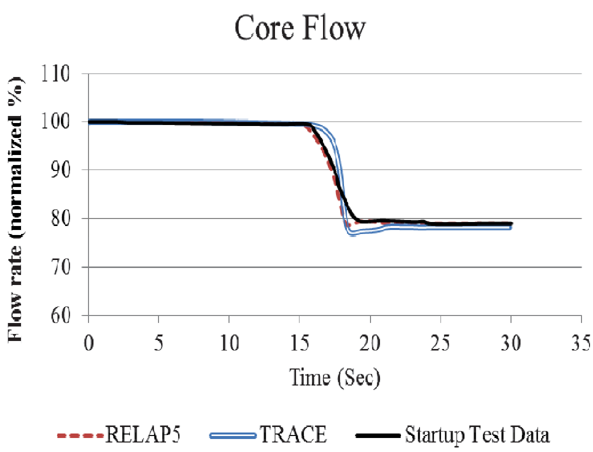

Fig. 8 Core flow variation during the 94% FWPT transient

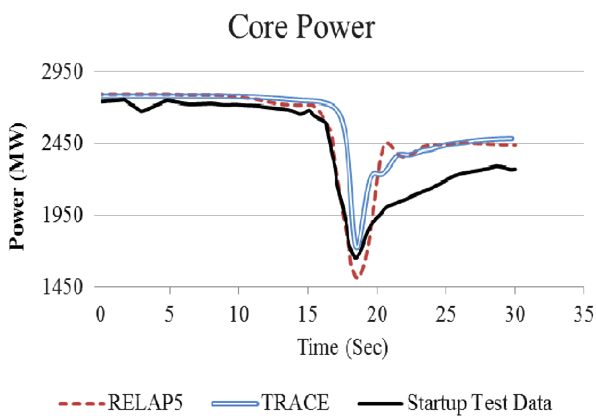

Fig. 9 Core power variation during the 94% FWPT transient

IV. Conclusion

In this research, the RELAP5/SNAP model of Kuosheng NPP was successfully developed. The startup test data and TRACE analysis results were used to compare with the results of RELAP5/SNAP model. The predictions of RELAP5 were consistent with the data of startup test and TRACE roughly. It indicates that there is a respectable accuracy in Kuosheng NPP RELAP5/SNAP model. Additionally, the RELAP5/SNAP model of Kuosheng NPP will be used in the safety analysis of the power upgrade in the future.

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