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Assessment of RELAP5/MOD2 Code using Loss of Offsite Power Transient of Kori Unit 1

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고리 1호기 외부 전원 상실사고에 의한 RELAP5/MOD2 코드 모델 평가

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Abstract

The Loss of Offsite Power Transient at 77.5% power which occurred on June 9, 1981 at the Kori Unit 1 PWR (Pressurized Water Reactor) is simulated using the RELAP5/MOD2 system thermal-hydraulics computer code. Major thermal-hydraulic parameters are compared with the available plant data. The comparison of the analysis results with the plant data demonstrates that the RELAP5/MOD2 code has the capability to simulate the thermal-hydraulic behaviour of PWRs under accident conditions of this type with accuracy, except the pressurizer pressure and level. The pressurizer pressure increase is sensitive to the insurge flow. It is believed that the interfacial heat transfer in a horizontal stratified flow regime may be estimated low and the compression effect due to insurge flow may be high. In the nodalization sensitivity study it is found that S/G noding with junctions between bypass plenum and steam dome is preferred to simulate the S/G water level decreasing and avoid the spurious level peak at trubine trip.

요 약

1981년 6월 9일 고리 1호기 원자력발전소에서 발생한 외부 전원 상실사고 자료를 근거로 RELAP5/MOD2 코드모델 평가를 하였다. 계산된 주요 열·수력학 변수를 실측자료와 비교 분석하였으며 증기발생기의 Nodalization 민감도 분석이 수행되었다. 계산된 열·수력학 변

수는 실측치와 비교적 잘 일치하고 있으며, 이러한 유형의 사고 분석에 RELAP5/MOD2가 적합하다는 것을 보였다. 그러나 가압기 압력과 수위변동에서는 상당한 차이를 보였으며 높게 계산되었다. 이러한 사실은 RELAP5의 수직관에서의 층류 열전달 모델에 기인하는 것으로 해당모델의 개선을 요하고 있다는 것을 알았다. 그리고 증기발생기의 Nodalization 연구를 통하여 수위변동을 잘 예측하기 위해서는 증기발생기 증기 Dome와 Downcomer 사이에 압력을 전달시켜주는 유로를 모델링 하여야 한다는 것을 알았다.

I. Introduction

Recent concerns and interests are of the full understanding and the prediction of the system thermal-hydraulic performances during plant transient in the efforts to quantitatively evaluate the performances during the progression of the transients. Therefore, the use of an advanced T/H codes has been promoted by the increasing trend to perform the transient analysis on a best-estimate basis. However, the use of an advanced, best-estimate codes for safety analysis requires that its uncertainties be identified by various ways of assessment, and eliminated through relevant updated technology. In other words, the capability of the code to accurately predict the plant behaviour should be quantified and confirmed. The present study follows up this trend and deals with the best-estimate calculation method in transient analysis, using RELAP5/MOD2. System thermal-hydraulic parameters are simulated based upon the sequence of events for the Kori 1 Loss of Offsite Power transient at 77.5% power which occurred on June 9, 1981 and compared with the plant transient data. Main objectives of the analysis are first, to assess the best-estimate system code, RELAP5/MOD2, and second, to evaluate the effects of the actuation and the func-

Table 1. Sequence of Events for Plant Transient (1981. 6. 9)

Time (sec)	Initiating Event	Simulated Event
0.0	— 77.5% Power Operation	— Steady State Calculation
50.0	— Mal-function of I/I converter S/G-A MFWCV statrts to close	— Accident sequence starts
100.0	— S/G-A low level & S/W mismatch Reactor/TBN trip	— Reactor trip (100.31 sec) Turbine stop valve close S/G-A low level & Tavg < 563.0 K
105.13		— S/G-B MFWCV starts to close
105.94		— S/G low level & Tavg < 563.0 K
125.94		— Aux. Feedwater Starts to feed
130.00	— Electric generator trip 2 Emergency D/G in operation	
131.00	— 154KV Bus-A fail to transfer — 154KV Bus-B suceed to transfer	
135.00	— RCP-A trip	— RCP-A trip (135.32 sec)
137.00	— Safeguard Bus-A in operation	
161.00	— 154KV Bus-B fail	
163.00	— RCP-B trip	— RCP-B trip (163.32 sec)
165.00	— Safeguard Bus-B in operation	

tioning of the safety and/or non-safety related components on the system transient.

Important thermal-hydraulic parameters such as reactor coolant system(RCS) average temperature, steam generator (S/G) level and pressurizer (PZR) water volume are compared with the plant data.

II. Sequence Description

Plant transient sequence is based upon the sequence of events record.[1] At around 11:00 AM on June 9, 1981, while operating at 77.5% reactor power and 447 MWe generator power, the I/I converter (LM-461A) of the S/G-A level control system mal-functioned generating a spurious signal that indicated high S/G-A water level. Major sequence of events of the transient is summarized in Table 1 together with the simulated boundary conditions of the sequence.

III. Input Model Description

The Kori 1 nodalization is shown in Fig. 1. The nodalization divides the whole systme into 113 volumes including 11 boundary volumes, 117 junctions and 79 heat slabs. Each steam generator is modeled with 8 heat slabs for U-tubes and 13 volumes having a steam separator. The reactor and RCP trip is modeled as a input trip time and the decay power model is ANS-79-1 using the previous plant power history data as shown in Fig. 2.

The nodalization sensitivity studies have been performed regarding to Steam Generator Noding with junctions between bypass plenum and steam dome, shown in Fig. 3. As shown in Figure, there was no junctions between bypass plenum (volume 172) and steam dome (volume 180) in the base case (Case 1). So there was no steam pass for feedback of pressure spike at

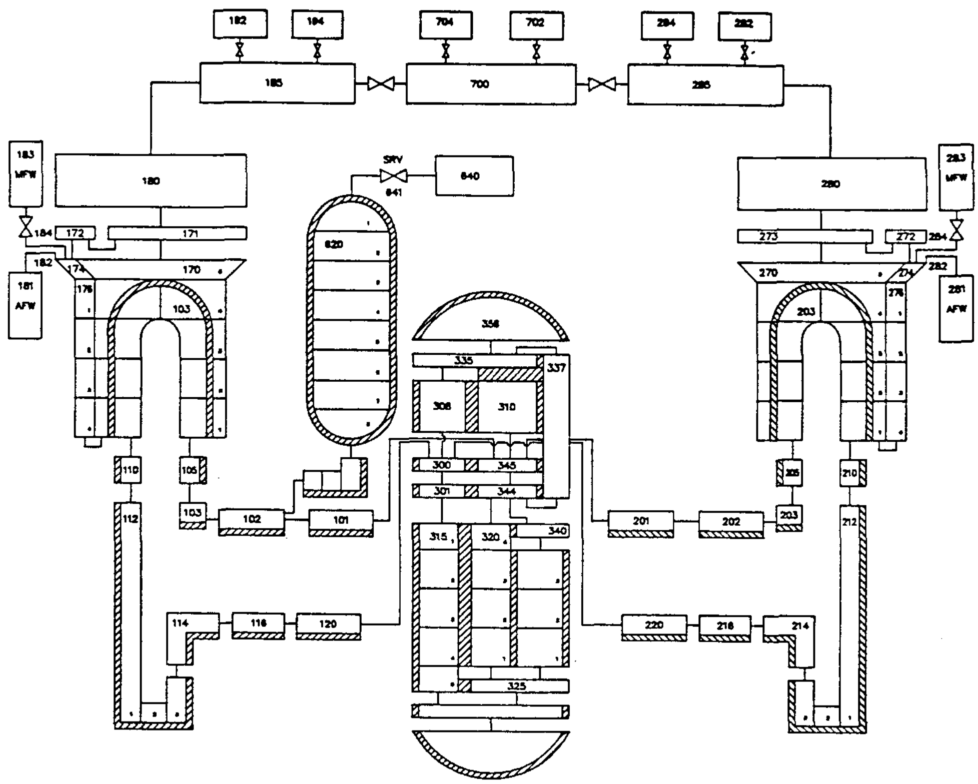

Fig. 1. Nodalization Diagram of Kori #1 Plant

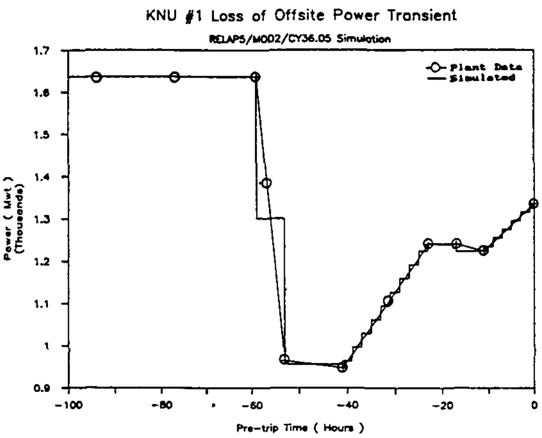

Fig. 2. Power History prior Reactor Trip

the turbine trip, and the pressure buildup at steam dome (volume 180) drove the flow of liquid into the volume 172 through separator (volume 171). This resulted in a spurious level peak. The water level decrease following decrease of feedwater flow was not simulated well because the flow stagnation occurred in volume 172. Three different nodalizations (Case 2, Case 3, and Case 4) were tested to evaluate the effect of junction orientation. The effect of the junction

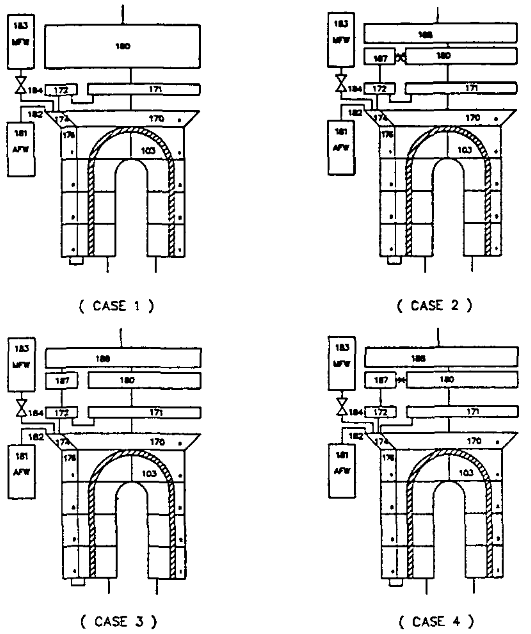

Fig. 3. Various S/G Nodalizations for Sensitivity Study

orientation is negligible, and thus the comparisons of case 2 with the base case (Case 1) were done in this paper. More details can be found in the previous work.[3].

Table 2. Initial Conditions (77.5%)

Parameters	Unit	Simulated	Plant Data
Core Thermal Power	(MW)	1,334.8	1,334.8
PZR Pressure	(MPa)	15.51	15.41
PZR Level	(%)	41.61	41.60
Hot Leg Temperature	(K)	583.06	583.10
Cold Leg Temperature	(K)	556.88	556.80
Loop Coolant Flowrate	(kg/sec)	4,688.28	4,686.50
Main Feedwater Flowrate	(kg/sec)	356.75	356.70
Feedwater Temperature	(K)	484.80	484.80
Steam Flowrate	(kg/sec)	357.00	356.70
S/G Pressure	(MPa)	5.896	—
S/G Narrow Range Level	(%)	44.02	44.00
S/G Mass Inventory	(kg)	49,783.4	—
U-Tube Heat Transfer Area	(m^2)	5,214.4	4,784.5
U-Tube Heat Transfer Rate	(MW)	1,340.9	1,339.8
Recirculation Ratio		3.36	—

The steady-state calculations for 77.5% power were carried out to provide the initial conditions for the transient analyses. The simulated initial conditions along with the plant steady-state data and design values are summarized in Table 2. Generally the simulated values are in excellent agreement with the plant values.

IV. Results and Discussions

Analyses were performed following the sequence described above. The initiating and the major simulated events during the progression of the transients are summarized in Table 1. As can be seen in the table, the sequence of events is identical in both cases but the timing of major events differs somewhat as expected. The simulated thermal-hydraulic parameters for 77.5% power are compared with the plant transient data, which are deduced from the computer daily log sheet[4] and trip review sheet.[5]

The plant transient occurred during power escalation, and hence most parameters were not in stabilized conditions which led to difficulties in deciding the appropriate initial values. Hence, the unreasonable plant data were ignored and the initial values were chosen either by an averaging process, or in some cases, from the design values specified in the FSAR (Final Safety Analysis Report.[6] In the initial stages of the plant transient, the main feedwater flow rate was automatically controlled by the MFWCV (Main Feed Water Control Valve) following the malfunction of the S/G level indicator. Since the actual automatic operation of the MFWCV is difficult to identify, the feedwater flowrate shown in Fig. 4 is assumed based on the plant data so as to correctly simulate reactor trip time.

As Shown in Fig. 5, the code cannot simulate the S/G level decreasing caused by feedwater decreasing and results sudden water level rise at the time of turbine trip. From nodalization sensitivity study, it is shown that the deficiency can eliminate by simulating the S/G noding with junctions between bypass plenum and steam dome. The S/G water level was obtained

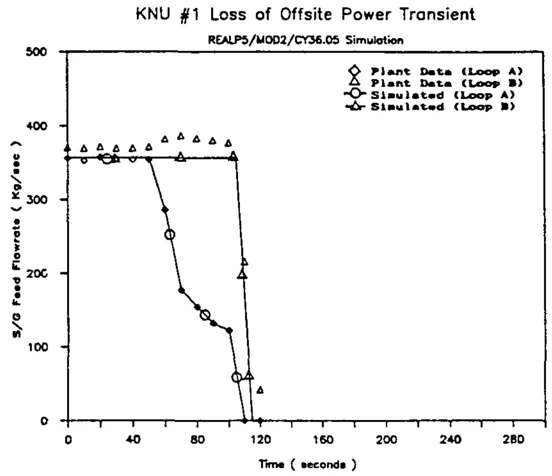

Fig. 4. Change of S/G Main Feedwater Flowrate according Mulfunction of Level Guage

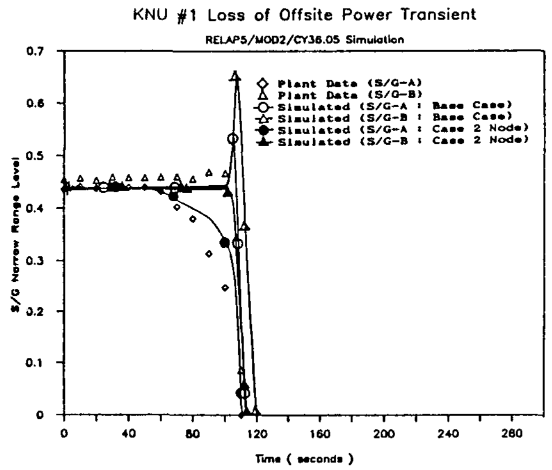

Fig. 5. Change of S/G Narrow Range Level

not by the pressure difference method, as used in the actual plant measurement, but by calculating the collapsed water volume deduced from the void fraction. This is because the pressure difference method, when used in the simulation, often gives rise to a doubtful level oscillation.[7].

Since the two loops of the S/G secondary side are connected via a single common head and because the main steam isolation valve(MSIV) does not operate in these analyses, the pressure variations in S/G-A and B are identical as shown in Fig. 6. Following the reactor/turbine trip, the S/G pressure rapidly increases as

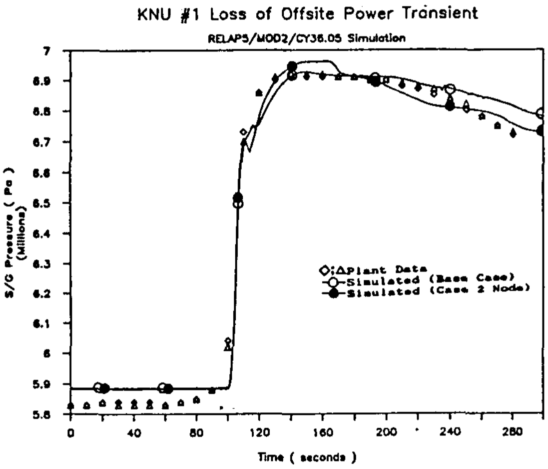

Fig. 6. Change of S/G Pressure

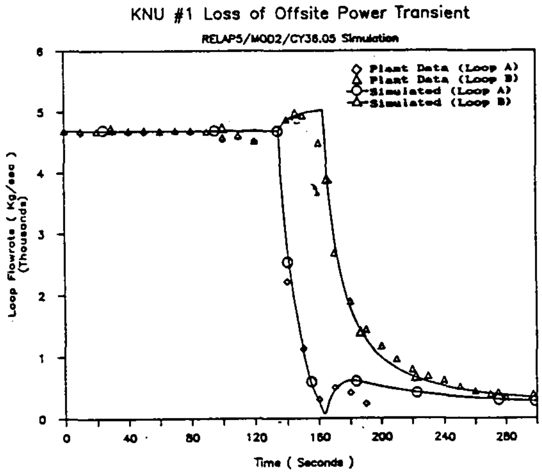

Fig. 7. Coastdown of Loop Flowrates followed by Pump Trip

the turbine stop valve closes. Normally, the turbine trip causes the steam dump valve to open, but in this transient, due to the loss of offsite power it remains closed. In the plant transient analysis, the S/G pressure starts to decrease as the supply of the auxiliary feedwater, actuated by the S/G low-low level signal, reaches its maximum capacity (155.94 sec) so that the secondary heat removal capability begins to overcome the reactor decay power. The calculated S/G pressure variation up to the peak pressure agrees quite well with the plant data. In the analyses, PORVs (Power Operated Relief Valves) were simulated to open at 7.033 MPa (1020 psia). But the supply of the auxiliary feedwater alone secures sufficient secondary heat removal capability without the operation of the PORVs.

The primary loop coolant flowrate versus time is shown in Fig. 7, and as shown in Table 1, the reactor coolant pump-A(RCP-A) tripped at 135sec and the RCP-B at 163sec. The rapid reduction in the RCS flowrate in loop-A due to the pump trip caused a decrease in the frictional resistance of the reactor vessel. Consequently the loop-B coolant flowrate increases until the subsequent trip of the RCP-B leading to the rapid reduction of loop-B flowrate. Meanwhile, the loop-A flowrate increases, due to the same reason as described above, just before flow reversal occurs and then decreases slowly. After 200 second, both trend in the calculated loop flowrates is in excellent agreement with the plant data.
loops show identical trend in the flow coastdown and the natural circulation begins to be established due to the hot-cold leg temperature difference. The overall

Fig. 8 shows the RCS temperature variations and one can note that the hot leg temperature variations for both loops are identical in spite of different RCP trip times. This is reasonable since the present simulation deals with a single channel model for the core, allowing complete liquid mixing. Immediately following the reactor/turbine trips the hot leg temperature decreases rapidly, whereas the cold leg temperature increases due to the reduction in the heat removal capability caused by the increase in the secondary side S/G pressure as described above. After the RCP-A trip, loop-A hot leg temperature has little effect on the cold leg temperature due to the delay in fluid transport and hence the cold leg temperature stays at the saturation temperature corresponding to the S/G-A pressure. Simiarly, the loop-B cold leg temperature also ceases its increase following the RCP-B trip. Afterwards, the cold leg temperatures slowly decrease as the S/G pressure decreases. The flow coastdown due to both RCP trips and the decay heat increase the hot-cold leg temperature difference until the establishment of the natural circulation in the primary side due to this temperature difference gives rise to sufficient heat transfer capability from primary to the secondary side (500 sec). Recognizing above trend in the temperature variations, one can note that the hot leg temperature increases until the stable natural circulation is fully established, and afterwards it decreases as the cold leg temperature. The simulated cold leg temperature agrees well with the plant data whereas the hot leg temperature behaviour is rapid. It is caused by the plant measurement method which use RTD (Resistance Thermal Detector) system. The RTD measurement depends strongly on the coolant flowrate, thus measurement was delayed by the pump coastdown. For comparison purpose the lag measure unit was simulated and the output from this unit was good agreements.

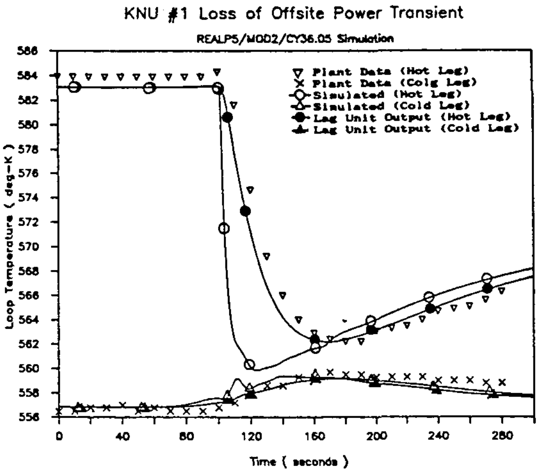

Fig. 8. Change of Loop Temperatures

The pressurizer level, shown in Fig. 9, is higher compared with the plant data. One of the causes may be a difficulty with calculating the accurate volume of the PZR bottom, in which the complex structures and PZR heaters are located. Another cause is a difficulty with predicting upper head temperature of reactor vessel. It is believed that the upper head temperature is between the hot leg temperature and the cold leg temperature due to 3 dimensional flow distribution in the upper part of the core. Using one dimensional code such as RELAP5, it is impossible to predict the upper head temperature correctly. If we consider the bypass flow thorugh upper head nozzles, the predicted temperature should be same as the cold leg temperature. Thus the contraction effect due to RCS cool down may be underpredicted and it results in the overprediction of pressurizer level.

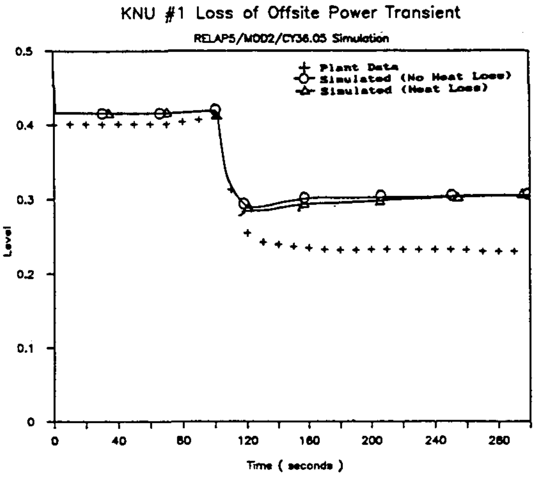

Fig. 9. Change of Collapsed PZR Water Level

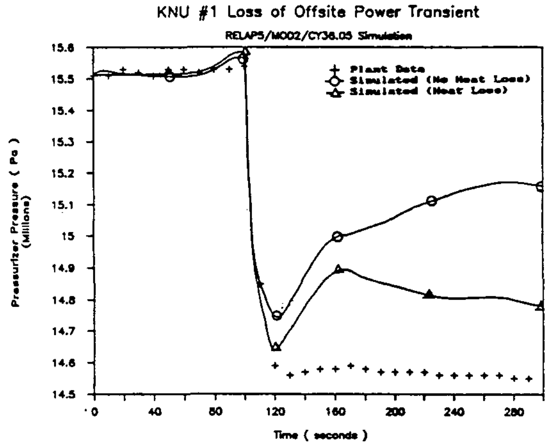

Fig. 10. Change of PZR Pressure

The pressurizer pressure, shown in Fig. 10, has a similar trend as the pressurizer level and is higher also as compared with plant data. But the slope of pressure increase in the heatup phase is much higher. It may be due to the treatment of pressurizer vessel wall. At the normal operating condition, the proportional heaters are partially kept on to compensate the heat losses. The amount is estimated as 167 KW. The pressurizer heater is simulated as heat slab and shut off at the time of loss of offsite power. The heat loss through wall is simulated as boundary condition of the wall heat slab and remains constant during the whole transient. As in Figure, the pressure increase due to the insurge flow is much higher than the plant data, although the improved result was obtained compared with the base case. It is believed that the interface heat transfer in horizontal stratified flow regime may be estimated low, and thus the compression effect may be high.

V. Conclusion

An analysis of Kori #1 Loss of Offsite Power transient was carried out using the RELAP5/MOD2. It is found that the code gives stable steady-state results and accurate predictions for most of the plant behavior associated with the transient, indicating the excellent capability of the code for this type of transients. The establishment of stable natural circulation due to the hot-cold leg temperature difference after both reactor pump trips is confirmed. In particular, the calculated primary thermal behavior closely follows the plant data and this validates that the relevant thermal-hydraulic and decay power model in the RELAP5/MOD2 correctly describes the actual phenomena.

In the nodalization sensitivity study it is found that S/G noding with junctions between bypass plenum and steam dome is preferred. This nodalization allowed the simulation of the S/G water level decrease and avoided the spurious level peak at turbine trip.

The pressurizer pressure increase is sensitive to the insurge flow. It is believed that the interfacial heat transfer in a horizontal stratified flow regime may be estimated low and that the compression effect due to insurge flow may be high.

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