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Analysis of Loss of Normal Feedwater Transient Using RELAP5/MOD1/NSC; KNU1 Plant Simulation

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RELAP5/MOD1/NSC를 이용한 원자력 1호기 주급수 상실 사고 해석

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Abstract

Simulation of the system thermal-hydraulic parameters was carried out following the KNU1 (Korea Nuclear Unit-1) loss of normal feedwater transient sequence occurred on November 14, 1984. Results were compared with the plant transient data, and good agreements were obtained. Some deviations were found in the parameters such as the steam flowrate and the RCS (Reactor Coolant System) average temperature, around the time of reactor trip. It can be expected since the thermal-hydraulic parameters encounter rapid transitions due to the large reduction of the reactor *thermal power in a short period of time and, thereby, the plant data involve transient uncertainties.* The analysis was performed using the RELAP5/MOD1/NSC developed through some modifications of the interphase drag and the wall heat transfer modeling routines of the RELAP5/MOD1/CY018.

요 약

1984년 11월 14일 원자력 1호기에서 발생된 주급수 상실사고에 대한 계통의 열수력학적인 거동을 모의 · 해석하고, 발전소 실측자료와의 비교를 통하여 사용된 전산코드의 신뢰도를 평가하였다.
모의된 열수력학적 변수들은 발전소 실측자료와 비교적 잘 일치하였으나 원자로 트립시에 증기발생기 증기유량과 주 냉각재 계통 평균온도에 있어서 약간의 차이를 보였다. 이는 원자로 트립시 짧은 시간에 급격한 노심 출력의 감소로 인하여 열 · 수력학적 변수들에 큰변화를 야기하여 발전소 실측자료가 과도상태에서의 불확실성을 내포하기 때문으로 예측되었다.
해석에 사용된 전산코드는 RELAP5/MOD1/CY018로 부터 불합리한 oscillation을 일으키는 interphase drag 및 wall heat transfer model의 수정을 통하여 개발된 RELAP5/MOD1/NSC이다.

1. Introduction

Recent concerns and interests are of the full understanding and the prediction of the system thermal-hydraulic performances during plant transients, that are beyond the scope of the design-base-accidents. It is ascribed to the fact that the multi-component failure and/or the operator's mis-operation result in severe accidents, as TMI experience showed. It has led to the introduction of the best-estimate methodology in the accident analyses, in the effort to quantitatively evaluate the system thermal-hydraulic performances, and to assess the impact of the component failures and the operator's misoperation on the progression of the accident. This trend has been realized in the many international research plans and one of the prime examples of this is the international joint research on Severe Fuel Damage Accident coordinated by USNRC (Larkins & Cunningham; 1983).

This study follows up this trend and deals with the best-estimate calculation method in transient analysis, using the RELAP5/MOD1/NSC developed through some modifications of the interphase drag and the wall heat transfer modeling routines of the RELAP5/MOD1/CY 018 (Kim, et al.; 1985).

Recently, statistical information on the reactor trip frequency in the Korea Nuclear Power Plants was provided in the report of Han, et al. (1985). The report indicated that the reactor trips due to the main feedwater system malfuntion cover up to 23% of total reactor trips (for KNU1; 19%). In addition, one should be reminded of the TMI accident (Rogovin; 1980), where initiating event was the feedwater system malfunction.

In the sense, the present study focuses on the analysis of the loss of normal feedwater transients. System thermal-hydraulic parameters were simulated following the KNU1 loss of normal feedwater transient sequence that had occurred on November 14, 1984 and compared with the plant transient data. The main objectives of the analysis are first, to assess the best-estimate system code, RELAP5/MOD1/NSC, and second, to evaluate the effects of the actuation and the functioning of the safety and/or non-safety related components.

2. Plant and Sequence Description

Plant Description

The KNU1 is a 587 MWe two-loop pressurized light water reactor. It consists of Westinghouse Nuclear Steam Supply System and GEC turbine-generator. The reactor coolant system consists of a reactor vessel, two inverted U-tube steam generators, two water-sealed reactor coolant pumps, an electrically heated pressurizer and various interconnecting pipings.

The two heat transport loops of the system are designated loop-A and loop-B. Each loop is made up of one coolant pump, one steam generator, a hot leg and a cold leg, and the pressurizer is connected to the loop-A.

Plant Transient Sequence

Transient sequence described here is based on the sequence of events record, which is obtained from the plant. On November 14, 1984 at around 10:00 AM, while operating at 100% power and 570 MWe generator power, the connecting line to one of the terminals at inverter-B transformer broke off due to overheating. As a consequence, steam/feedwater flowmeter, to which power is supplied by the failed transformer, lost its power, and spurious steam/feedwater mismatch signal was generated. At the same time, steam flowmeter, also due to power failure, gave spurious signal indicating reduction in steam flow. This signal activated the feedwater flow controller(FC-466A) to close the main feedwater control valve. Closing the valve caused rapid reduction in the steam generator water-level. Upon perceiving the rapid reduction of water-level, the operator promptly switched to manual mode and started to control the feedwater control valve. However, after a few seconds, the steam/feedwater flow mismatch coupled with the low water-level in steam generator-A caused the reactor trip and then the turbine was tripped.

Afterwards the reactor remained in Hot Zero Power condition.

3. Code and Input Model Description

Code Description

The RELAP5/MOD1/NSC used in the analysis is a best-estimate system code for the transient analysis of the Pressurized Water Reactor. The code is originated from the RELAP5/MOD1/CY018 (Ransom, et al.; 1981) and developed by modifying the thermal-hydraulic models to avoid an unphysical flow oscillations (Kim, et al; 1985). These modifications involve two-phase flow regime map, interphase drag and wall heat transfer models.

The RELAP5/MOD1/NSC classifies the flow regimes to bubbly, slug, and annular-mist flow. The transition from the bubbly to the slug flow is based on that, for the bubbly flow in small-diameter tubes, the bubble rise velocity could not exceed that of the Taylor bubble (Taitel, et al.; 1976, 1980). The transition from the slug to the annular-mist flow is based on the critical vapor velocity required to suspend a liquid drop (Taitel, et al.; 1980). The interphase drag coefficient for the bubbly and the slug flow regimes is based on the result of Ishii & Chawla (1979). In the annular-mist flow regimes, the entrainment relation deduced from Ishii & Mishima (1980) is chosen to calculate the interfacial area, and the interphase drag coefficient is calculated from the Bharathan, et al. (1978) correlation. The comparison of models in the RELAP5/MOD1/NSC with the RELA5/MOD1/CY 018 is shown in Fig. 1. Significant variations are found in the interphase drag, especially at annular-mist region.

In calculating the heat fluxes for mass flux below 200kg/m² sec, the RELAP5/MOD1/CY018 adopts the scheme whereby the heat fluxes are calculated from both the convective boiling and the pool boiling modes and then the maximum value is taken. However, this unreasonable scheme results in an unphysical flow oscillation in steady-state calculation at a relatively low reactor power, that is, at a lower mass flux. RELAP5/MOD1/NSC interpolates linearly the heat fluxes, which are calculated from both boiling modes, according to the mass flux between 50~200kg/m² sec, as shown in Fig. 2.

Input Model

The nodalization method used in RELAP5 for the system transient analysis is to divide the fluid system into a system of subcomponents consisting of volumes and junctions. The solid

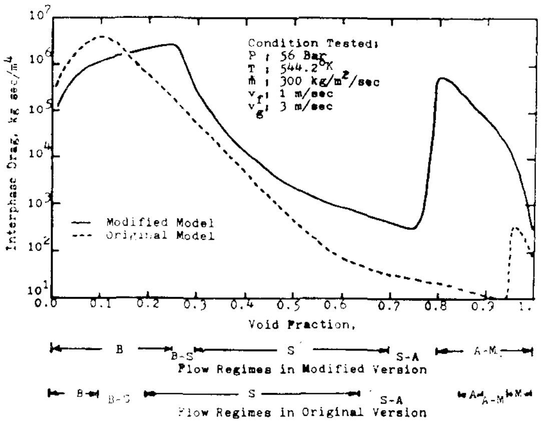

Fig. 1. Comparison of the Original Model with the Modified in Interphase Drag Calculation(B; Bubbly, S; slug, A; Annular, M; Mist)

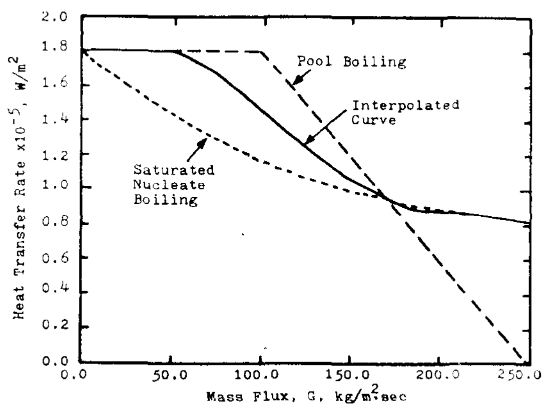

Fig. 2. Comparison of the Original Version with the Modified in Heat Transfer Rate, 50<G<200 (kg/m²·sec)

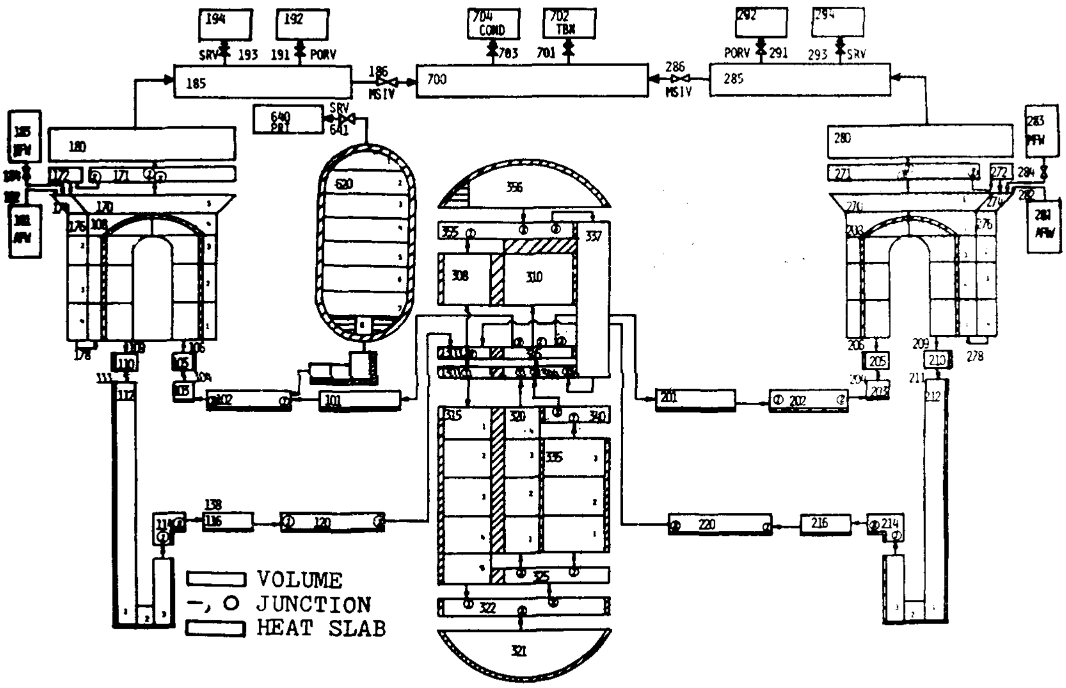

Fig. 3. RELAP5/MOD1/NSC Nodalization

structures are modeled using a sub-component named heat-slab for the thermal considerations and solid-fluid interactions. Nodalizing the system involves the consideration of the computer memory size, computing time and the accuracy of the calculation.

The nodalization of KNU1 in the present simulation divides the whole system into 113 volumes including 11 boundary volumes, 117 junctions and 79 heat-slabs. The nodalization scheme is shown in Fig. 3. Each steam generator was modeled with 13 volumes including a steam separator and 8 heat-slabs for U-tubes. The outlets of both steam generators are connected to form a single volume, steam-head, which is then connected to two time-dependent volumes to act as the pressure boundary conditions of the steam generator.

Initial Conditions

A RELAP5/MOD1/NSC calculation of steady state at full power operation was carried out to provide the initial conditions for the transient simulation. The simulated initial conditions are summarized in Table 1, with the desired plant steady-state data. In general, the simulated values were in excellent agreement with the desired values.

The simulated reactor power is close to the desired value, but the total heat transfer rate in the steam generators is about 2.9MW higher. This is because the reactor coolant pumps gene-

Table 1. Initial Conditions

Parameters		Simulated	Desired
Core Thermal Power	(MW)	1,723.5	1,723.5
PZR Pressure	(MPa)	15.5	15.5
PZR Level, Narrow Range	(%)	46.73	47.6
Hot Leg Temperature	(K)	589.41	589.36
Cold Leg Temperature	(K)	555.94	555.89
Loop Coolant Flow	(kg/s)	4,687.5	4,687.5
Main Feedwater Flow	(kg/s)	473.1	473.1
Feedwater Temperature	(K)	496.3	496.3
Steam Flow	(kg/s)	473.9	473.1
S/G Pressure	(MPa)	5.55	5.55
S/G Narrow Range Level	(%)	43.9	44.0
U-tube Heat Transfer Rate	(MW)	1,731.4	1,728.5

rate heat approximately 2.9MW more than desired. However, this increase in the total heat transfer rate has no significant effect on the initial thermal-hydraulic parameters of the secondary side as can be seen in Table 1, except for the steam generation rate which shows some (0.8 kg/sec) increase as expected. Following the reactor trip, the decay heat becomes an important parameter in transient analyses. The decay heat is strongly dependent on the initial core thermal power and hence the slight increase in total heat transfer rate is expected to give little effect on the thermal-hydraulic parameters throughout the whole transient period.

Two percent deviation in the pressurizer level was found to have no significant effect on the transient events or timing.

4. Results and Discussion

Present analysis was performed following the plant transient sequence described in Section 2. The initiating events and the major simulated events during the progression of the transient are summarized in Table 2. Major thermal hydraulic parameters are compared with the plant

Table 2. Sequence of Events

Time (sec)	Initiating Events	Simulated Even
0.0	—100% Power Operation	—Steady-State Calculation
50.0*	—Malfunction of Inverter 'B'	
	—S/G 'A' MFWCV Start to Close	—Use Plant Feedwater Data
82.0	—S/G 'A' Low Level and Steam Flow Mismatch	—Reactor Trip
	—Reactor and Turbine Trip	—Turbine Stop Valve Start to Close
85.5		—Steam Dumping Start
87.0		—S/G 'A' Low-Low Level
		—Aux. Feedwater Actuation
96.95		—S/G 'A' Low Level and $T_{avg}<563K$
		—S/G 'B' MFWCV Start to Close
107.0		—Aux. Feedwater Begin to Feed
400.0		—End of Calculation

*; Actual Time is 10H 11M 27S

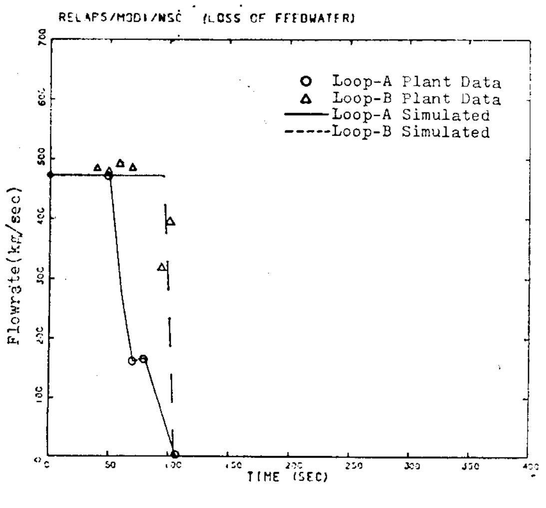

Fig. 4. S/G Feedwater Flowrate vs. Time

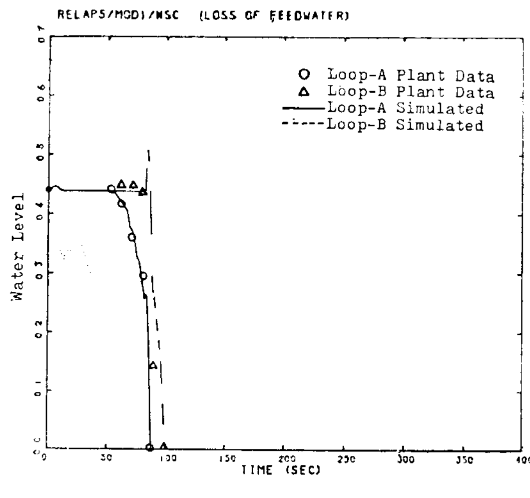

Fig. 5. S/G Water Level vs. Time

transient data, which are deduced from the computer daily log and trip review sheets, as shown in Fig. 3-11. The plant data between 170 sec. and 250 sec. are missing due to the failure of the computer logging process.

The feedwater flowrate, shown in Fig. 4, is taken from the plant data and used to define a boundary condition, since the feedwater flowrate was modulated by the operator's action during the initial stages of the transient. Fig. 5 shows that the calculated steam generator water-level is in excellent agreement with the plant data. It indicates that the code responds very well to the variation in feedwater flowrate, leading to the same reactor trip time.

The steam generator water-level can be calculated from the pressure difference between the upper and the lower tap locations, which is identical to the actual measurement method. This Δp method resulted in a doubtful oscillation. Thus, the water level, in the present simulation, is estimated based on the collapsed water volume which is deduced from the volume void fraction.

In Fig. 6, the steam generator pressure variation shows similar trends, especially around the reactor trip time, however, the first peak pressure is quite lower in the present analysis. This is expected since the steam dump valve is simulated to fully open in 3 seconds (design value), which may be considerably shorter than the actual valve performance. This points out that the occurrence and the magnitude of the first peak pressure is quite sensitive to the valve performance. It then becomes important to obtain the actual valve performance data. This trend is also recognized in the steam flowrate, as shown in Fig. 7. The steam dump flowrate, shown in Fig. 8, well explains the trend.

Fig. 7 shows that some deviations exist in the steam flowrates after the reactor trip. It may be partly because the steam flowrate encounters great variations at a short period due to the rapid reduction of the reactor thermal power, from 100% to approximately 4% within 10 seconds, and thereby the plant data may involve the transient uncertainties, such as measurement errors and the delay in response times. Thus, this deviation seems to be within a reasonable range in that the measurement in steam flowrates involves ±5% uncertainties (KNU1 Final Safety Analysis Report). In addition, the steam flowrate experiences a little unphysical oscillation, near the time of the reactor trip. It may come from the fact that the hysteresis effect is not accurately considered in the interphase drag calculation and the separator model may not

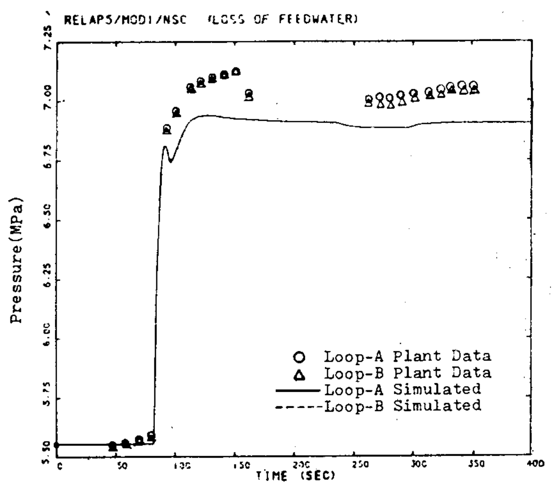

Fig. 6. S/G Steam Pressure vs. Time

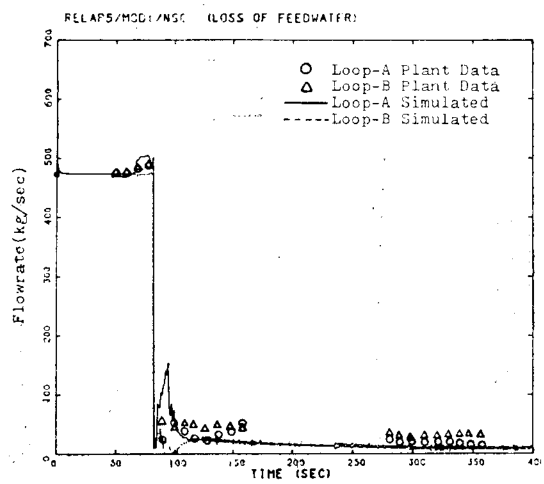

Fig. 7. S/G Steam Flowrate vs. Time

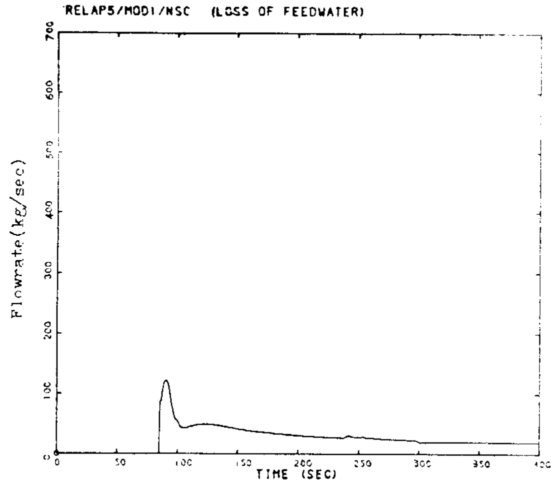

Fig. 8. Steam Dump Flowrate vs. Time

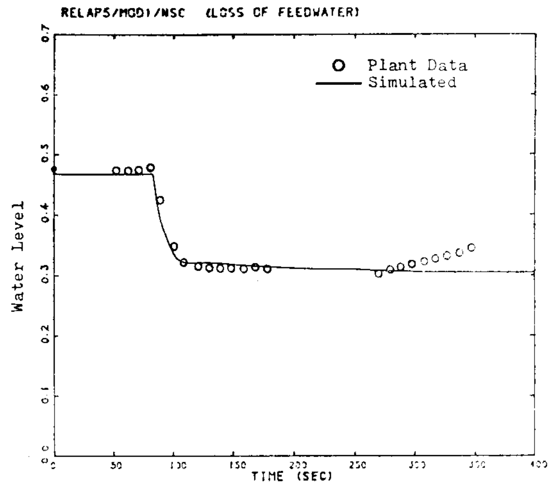

Fig. 9. Pressurizer Water Level vs. Time

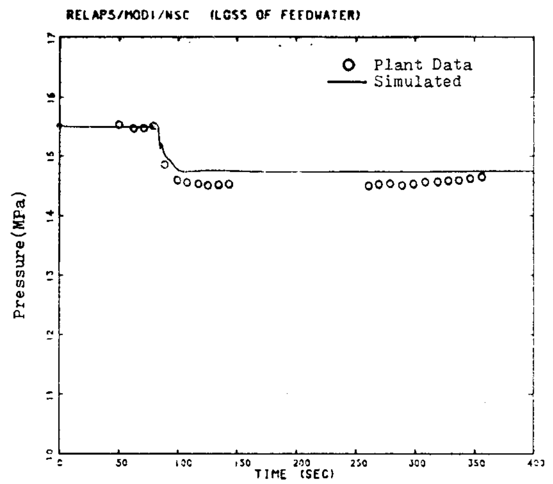

Fig. 10. Pressurizer Pressure vs. Time

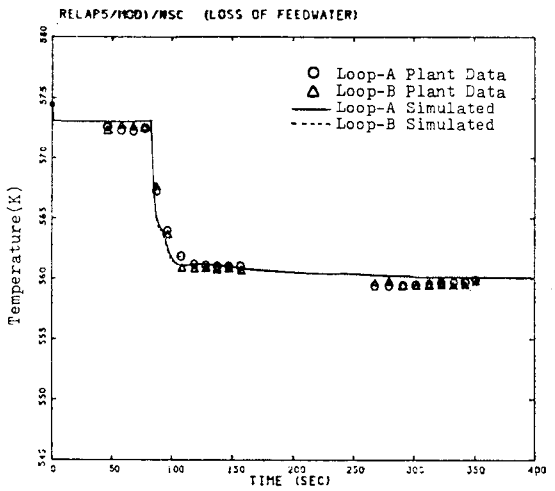

Fig. 11. Coolant Average Temperature vs. Time

comply with the actual phenomena.

The thermal-hydraulic parameters in the primary side, such as the pressurizer level, pressure and the RCS average temperature, show good response to the variations in the secondary side, resulting in a good agreement with the plant data, as shown in Fig. 9-11 respectively. The pressurizer water level is estimated using the same method as described in the steam generator level calculation. The actual pressurizer pressure and level start to increase slowly after 250 seconds, while, in the present analysis, the parameters reach the equilibrium state around 150 seconds and remain in that state. Possible causes of this discrepancy could be that the operator may have taken some actions and/or the automatic control system may have been actuated.

5. Conclusions

An analysis of KNU1 loss of feedwater transient was carried out using the RELAP5/MOD-1/NSC developed through some modifications of the interphase drag and the wall heat transfer modeling routines of the RELAP5/MOD1/CY 018. The code gave a stable steady-state and excellent predictions of the plant behaviour during the transient, pointing to the good capability of the code in transient analyses.

Some deviations in the steam flowrate were found since the thermal-hydraulic parameter encounters great variations due to the rapid reduction of the reactor power and, thereby, the plant data may involve the transient uncertainties.

The characteristics of the non-safety related components such as the steam dump valves,etc. are recognized to be important in the transient analyses on a best-estimate basis. In addition, the simulation of the control system is essential in this type of analyses.

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