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Plant Cooldown Test Simulation After Steam Generator U-Tube Rupture under Offsite Power Available Without Safety Injection

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증기발생기 세관파열사고 후 소외전원 가용 및 비상냉각수 주입 배제 조건하에서의 발전소냉각에 관한 실험 모사

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Abstract

The objective of the PKL III A 4.4 experiment is to examine that the plant could be controlled by manually operative actions "after Steam Generator Tube Rupture under Offsite Power Available without Safety Injection". In order to verify the limitation and ability of the system code NLOOP in the expeiment simulation, the behaviors of the PKL III facility obtained in the experiment are compared with the results of NLOOP code. NLOOP code, which is originally developed to simulate the transients of the Westinghouse type PWRs by KAERI/SIEMENS, modified properly to simulate the PKL III facility. Particular attention is given to the RCS mass flow rate of the natural circulation in loops and the termination behavior of the natural circulation in the isolated loop. The comparisons between the experimental and calculational results show the simulation ability and problems of the code.

요 약

PKL III A 4.4 실험은 "증기발생기 세관파열사고 후 소외전원 가용의 조건 하에서 발전소가 비상냉각수 주입없이 수작동에 의해 제어될 수 있음을 확인하는 것이다. 실험 모사에 따른 NLOOP Code의 제한이나 능력의 검증을 위해, 실험에서 얻어진 PKL 설비의 거동은 NLOOP의 결과와 상호 비교되었다. NLOOP 코드는 한국원자력연구소와 독일 SIEMENS/KWU사에 의해 Westinghouse 형 발전소의 과도현상 해석용으로 개발되었으며, PKL III 설비모사를 위해 적절히 수정되었다. 자연대류에 의한 RCS Loop의 냉각수 유량과 격리된 RCS Loop에서의 자연대류 중단현상을 특별히 주의깊게 연구하였다. 실험과 계산 결과의 비교는 NLOOP 코드의 모사능과 문제점들을 보여준다.

1. Introduction

The objective of this study is to compare the behaviors of the PKL (Primaer Kreislaeufe) III facility (Figure 1-a) obtained in an experiment with the results of the NLOOP [1] to verify the code. The experiment was performed in the PKL III facility [2] to investigate plant cooldown behavior and examine the manual operating procedure at an event of "Steam Generator Tube Rupture under Offsite Power Available without Safety Injection (Experiment Number: PKL III A 4.4)".

The Steam Generator (SG) U-tube rupture event is initiated by a penetration of the barrier between the primary coolant system and the secondary steam generation system in the U-tube steam generator. When it happens, it causes a depressurization of the Reactor Coolant System (RCS). Because of the pressure difference between primary and secondary side, the coolant flows into the defected SG until the pressures of the both sides are equalized. The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on time in order to minimize contamination of the secondary system. When the recovery procedure provided by this experiment is carried out on a timely manner, the plant can be cooled down without the safety injection. The RCS natural circulation behavior of the affected SG is of a particular interest during the stepwise cooldown phase from the hot shutdown condition (RCS temperature of 244°C) to the cold shutdown condition (that of 50°C) in manual mode to avoid the safety injection into the core.

The complete experimental sequences and details of the initial and boundary conditions are described in Reference 3. The important sequences and operator actions during experiment are listed in Table 1.

The following phenomena in the experiment are particularly investigated:

- o Natural circulation in the primary side,
- o Pressure and water level change in the isolated SG, especially after the isolation of the affected SG until the pressure of the primary side equalizes to that of the secondary side,
- o Pressure drop in the primary side during and after steam evaporation in the primary side SG U-tubes, and
- o Secondary system conditions in the isolated SG during the plant cooldown phase.

Table 1. Important Sequences of Events

Time (s)	Sequences
0.	Leakage opening in SG 10 (size: 2×1.8 mm^2) KBA (volume control system) extraction off
60.	Isolation of feedwater to SG 10
900.	Isolation of steam flow from SG 10
2700.	1st stepdown depressurization of pressure (2.5 bar) in SGs 20, 30, 40
3420.	2nd stepdown depressurization of pressure (2.5 bar) in SGs 20, 30, 40
4680.	KBA extraction on
5100.	Pressurizer heaters on
5640.	3rd stepdown depressurization of pressure (2.5 bar) in SGs 20, 30, 40
6900.	LBA (steam line system) 10 pressure decreased by 2 bar
7500.	Pressurizer water level control (manual)
8100.	LBA 10 pressure decreased by 2 bar
8400.	LBA 20, 30, 40 pressures decreased by 3 bar
10140.	Cooldown by 50 K/hr
12180.	KBA injection with a pump
16680.	JNA (PKL III specific heat removal system) on
19500.	Isolation of steam flow from SG 10
21120.	Isolation of steam flow from SGs 20, 30, 40
23700.	End of experiment

2. Conversion from NLOOP to PKL III Specific Version [4]

NLOOP is one of Non-LOCA safety analysis codes in Korea for PWR design applicactions and used for NSSS transient simulation of Westinghouse plants. The code contains models for major components of the primary and secondary side, for important auxilary systems and for the essential control, limitation, protectional interlocking system. The fluid in the primary coolant system is treated as homogeneous. Temperature non-equilibrium is allowed in the pressurizer, steam generators, feedwater tank and reactor pressure vessel head. The one-dimensional conservation equations of mass and energy are integrated by an explicit numerical method. Node/flow path networks are used to model the flow rates in the primary coolant loops and the main steam and feedwater systems. The one-dimensional, non-steady state channel flow of the primary coolant circuit is approximated by a serial distribution of the homogeneous zones defined by input.

The Fig. 1-b shows by way of example, the nodalization for two computation loops with 19 zones per loop. The main steam piping system consists of a node model. The networks of the line systems could be different from plant to plant. In NLOOP, main steamline system is modelled for Westinghouse type PWR, especially 2 loop and 3 loop plant. However, PKL III facility has 4 loops and its configuration is less complex than the real plants. Therefore, it is necessary to model the 4 loop geometry and accompanying functions for the simulation of PKL III test (see Fig. 1-a).

The NLOOP-PKL specific version is prepared by extending from 3 loop to 4 loop geometry and by simplifying the code due to the less complex configurations than the real nuclear plants. The geometric and thermal-hydraulic data of the PKL III facility for the PKL-Version are prepared in Reference 5. These include primary side volume and elevation, pump characteristics, steam generator geometry, and fuel rod data. Most code modifications are needed to take into account system and size differences between the real plant and the PKL facility, such as volumes and factors related to surfaces or volumes. Most of the instrumentations and control routines of NLOOP are in fact removed because the PKL test facility operates with fundamentally different control equipments. For the verification analysis, missing control variables are replaced by inputting measured variables. Each time dependent action is properly considered in the NLOOP calculation. These changes

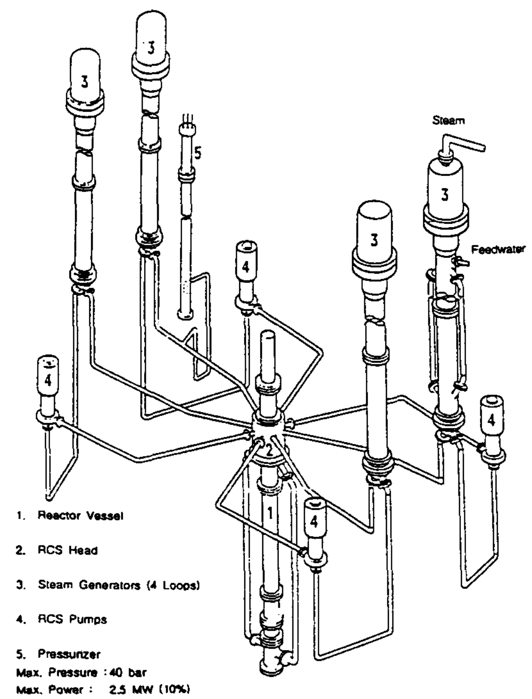

Fig. 1-a. Schematic Diagram of PKl III Facility

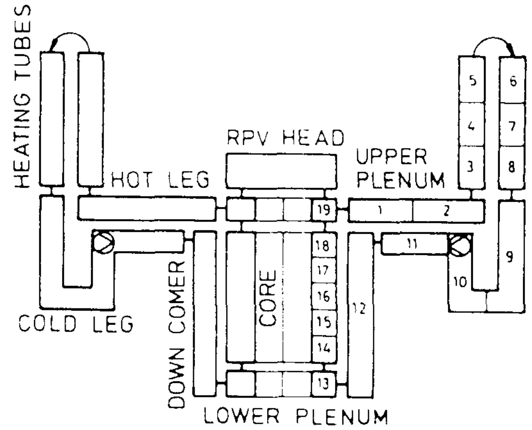

Fig. 1-b. Nloop Nodalization for two Computation Loops With 19 Zone per Loop

bring in no significant loss on the verification purpose, since the main objective is to identify thermal-hydraulic differences between calculation model in the NLOOP and the facility due to the natural circulation.

In PKL III cylindrical rods, which include the heating coil inside, are used for energy generation instead of the real fuel rods. Therefore, geometry and the governing equations of the heat transfer are different from the heat transfer model using real fuel rods. In subroutine BREST, cylidrical heating rods are modelled by the regional lumped parameter method, and the derivatives of temperature of each layer are calculated.

3. Steady State Calculation

The starting conditions of the experiment, namely, conditions at the opening of the U-tube leakage in SG 10, were set during the semi-equilibrium phase. For verification of the transients with the NLOOP, it is necessary to commence with steady-state initial conditions at a pre-set core output of equal heat transfers in all four steam generators with all reactor coolant pumps running. Therefore, an initial constant power output of 440 kW is assumed to obtain the facility parameters at the beginning of the transient.

Recording of plots of the transients (PKL measurements) begins 5874 seconds after the start of the test. At this time the reactor is already tripped and the plant is under natural circulation with SG pressure of about 26 bar and a uniform heat transfer in all 4 SGs. In the NLOOP calculation, this condition is also attained after the 5874 seconds. All subsequent events are therefore proceeded by the initial 5874 seconds in the verification calculation. Up to this time, inlet and outlet temperatures attained in all 4 SG loops are almost identical. These are entered into the NLOOP representing the initial transient conditions. To achieve the same values of pressurizer level and pressure as those of PKL test, the pressurizer level is controlled by the time dependent charging and letdown flow rates in the RCS of the NLOOP, and the NLOOP pressurizer pressure is given by the time dependent PKL III measurement value.

Due to the relatively large heat losses through the wall structure of the PKL III test facility, the thermal energy transferred to the secondary side is much smaller than the core output despite compensatory heating for the steam generator and reactor pressure vessel through the back-up heaters [5]. Since the computer model NLOOP makes no allowance for external heat losses through the wall structure, the core output was set to the effective heat energy transferred to the SGs. To enable extraction of comparable results from the NLOOP calculation, the energy loss involved is estimated from the existing primary mass flow, the temperature drop between reactor pressure vessel (RPV) and SG, and between SG and RPV inlet. To allow for the energy loss, i.e. retain the transfer load in SG from NLOOP, the core output is reduced by a correspondind amount.

$P_{loss} = (m \cdot (H_{RPV\text{-}SG} + H_{SG\text{-}RPV})) +$ other thermal radiation losses $= 37$ kW

Ploss : energy loss
m : primary mass flow
H : enthalpy

From this calculated output levels, the effective thermal energy transferred by SG is found to be 403 kW. The verification calculation is performed for using this effective output of 403 kW.

A main steam and feedwater flow rate of 0.045 kg/s is necessary to remove the effective heat transferred to the steam generators. The heat on the heater rod bundle chamber and the pressure in the pressurizer are held constant until the opening of the U-tube leakage. The important initial and boundary conditions shortly before the onset of the coolant leakage into the SG 10 are listed in Table 2.

4. Transient Simulation and Results

The important plant parameters both in the

Table 2. Important Initial and Boundary Conditions Shortly Before U-Tube Leak

Primary Side	
– Power output	440 kW
– effective core output	403 kW
– Primary system pressure	40 bar
– Core outlet temperature	244°C
– Core outlet subcooling	6°C
– Temperature difference in core	20°C
– Pressurizer water level	4.7 m
– Pressurizer temperature	250°C
– Reactor coolant pumps	Not in operation
– Natural circulation in each loop	1.15 kg/s
Secondary Side	
– Main steam pressure	26 bar
– Main steam temperature	226°C
– Steam generator level	11.2 m
– Feedwater and steam flow rates	0.045 kg/s

NLOOP PKL-Version and their measurements taken in the PKL III test facility are compared in Figures 2 to 10. All measurement data are plotted synchronizingly for comparison. The pressures both in the SGs and RCS are given as time dependent inputs for the NLOOP simulation, because the time dependent pressures are controlled manually in the experiment (see Table 1). The fraction of the pressure vessel head bypass flow rate is changed in the NLOOP calculation in order that the calculated pressure vessel head temperature agrees with the measured value. In order to avoid the numerical discontinuity, a small flow change rate between the head and upper plenum is selected.

Figures 2 through 4 trace the SG pressures and levels measured in the PKL III experiment and calculated by the NLOOP, respectively. Due to the pressure difference between the primary and secondary side, there is a flow of the coolant into the secondary side of the SG 10. This leakage of the RCS flow causes a water level increase from 11.4 m up to 11.7 m in the affected SG 10 at time of 900 seconds. During this period, the pressure of SG 10 is equal to those of other SGs, because the SGs are connected each other through the steam header. After the 900 seconds, the defected SG 10 is isolated both from the feedwater system and steam line system, and the decay power begins to decrease according to DIN 25463 (Deutsche Industrie-Norm, German Industrial Standard). The water level increase in SG 10 continues due to the coolant leakage until the pressures of both sides equalize. Since the defected SG is not capable of removing the heat transferred from the primary side, the SG 10 is heated and its pressure increases. Following isolation of the SG 10, the total thermal energy generated in the core is transferred mainly to the SGs 20, 30 and 40 through the natural circulation mechanism. Because the coolant density differences within among the U-tubes, the SG plenums and the RCS coolant piping connected with loop 10 become smaller, the RCS mass flow rate due to the natural circulation in loop 10 is also reduced.

The SG 10 outlet temperature will be higher than those of the intact loops. As the water temperature of the secondary side in SG 10 equals to the reactor vessel outlet temperature, there is no more RCS mass flow in loop 10, while the mass flow of about 1.15 kg/s is continued throughout in the other loops with the cold leg and hot leg temperature difference of 20°C due to the natural circulation. However, the calculated mass flow in loop 10 was continued longer than measured flow because the pressure difference between primary side and secondary side in SG 10 is still large enough to promote the loop flow under der existing of leak flow (at 4800 seconds in the experiment and at 6800 seconds in the calculation for the loop flow termination in loop 10). This result has no practical effect on the intact loops, because the decay heat generated in core is almost completely removed by the existing natural circulation in loops 20, 30 and 40 (Figures 5 through 9).

As the depressurization of the RCS continues, the primary side of the SG 10 U-tubes approaches a saturated condition and it brings in steam formation. Due to the evaporation process on the primary side of SG 10, the water level in pressurizer increases fast from 4. 7 m up to 9 m in the test, while it occurs at a later time with smaller heights in the calculation. Because the NLOOP model does not take into account for the local steam formation in the primary side of the SG 10 U-tubes, this effect is considered in the NLOOP calculation by the charging flow into RCS (Fig. 10).

After 6900 seconds, the pressure in the isolated SG 10 was decreased by 2 bar by means of the steam blowdown through the actuation of the steam-line system (LBA). The reverse heat flux transferred from the secondary side to the primary side of SG 10 decreases. And the steam bubbles in the U-tubes of SG 10 are condensed and the pressurizer level drops again to 5.5 m. Eventually, the U-tubes in SG

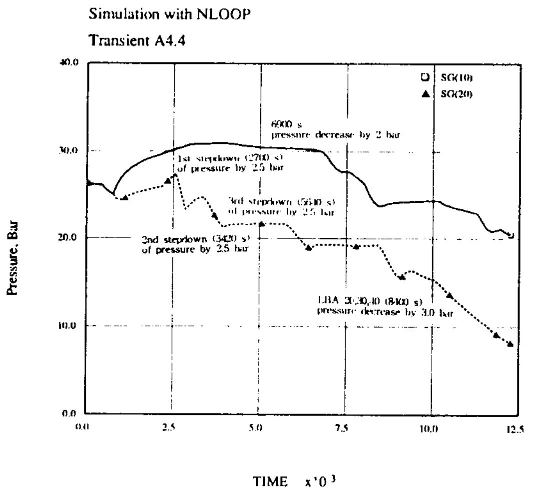

Fig. 2. SG(10)/SG(20) Pressure

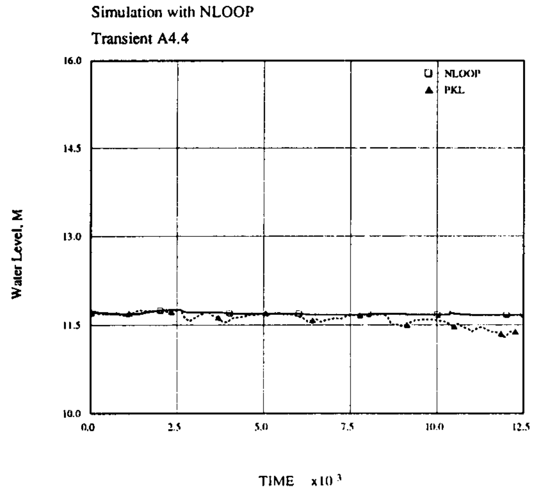

Fig. 4. SG(20) Water Level

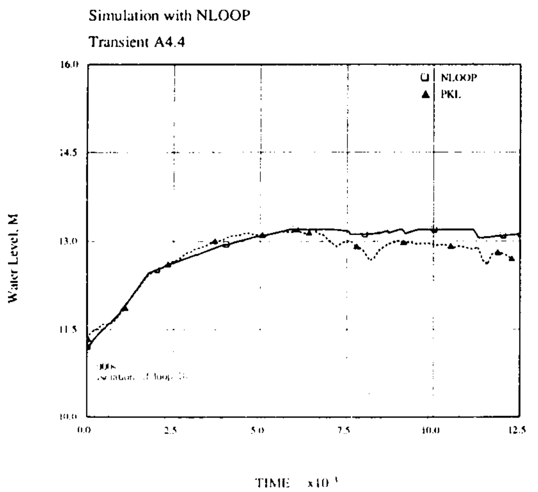

Fig. 3. SG(10) Water Level

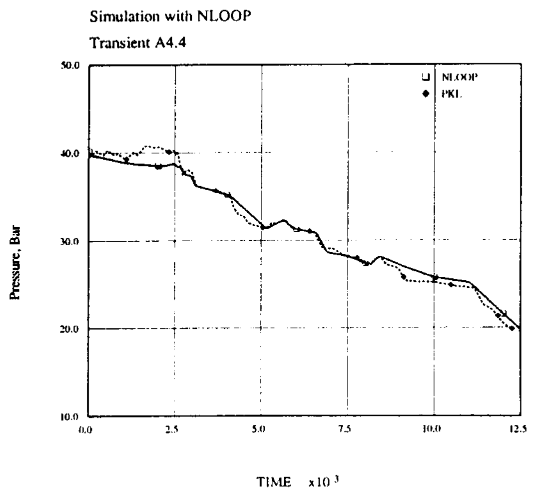

Fig. 5. RCS Pressure

10 are refilled with the primary coolant (Figures 2 and 10). After 8000 seconds, the water level is controlled manually both in the test and calculation. The oscillation in the pressurizer level is caused by the repeated evaporation and condensation processes in loop 10. During the entire transient, the pressurizer level reaches the maximum values of 12 m (measured) and 11. 5 m (calculated). After 10140 seconds, the cooldown process is performed by steam pressure drop with a cooldown gradient of 50 K/hr through the steam system of the intact SGs, which is well simulated by the code.

5. Conclusions

The PKL III test results show a successful manual handling of the plant against the SG U-tube rupture event without ,the safety injection. The plant under

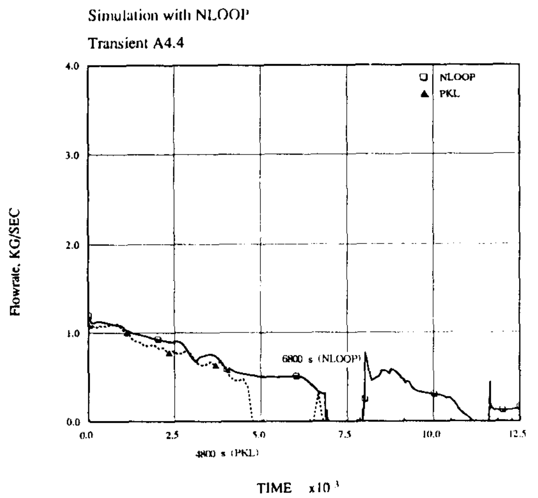

Fig. 6. RCS Flow Rate (Loop 10)

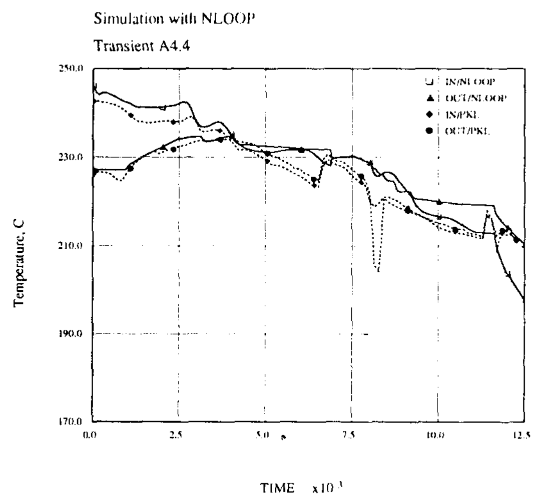

Fig. 8. SG(10) Inlet/Outlet Temperature

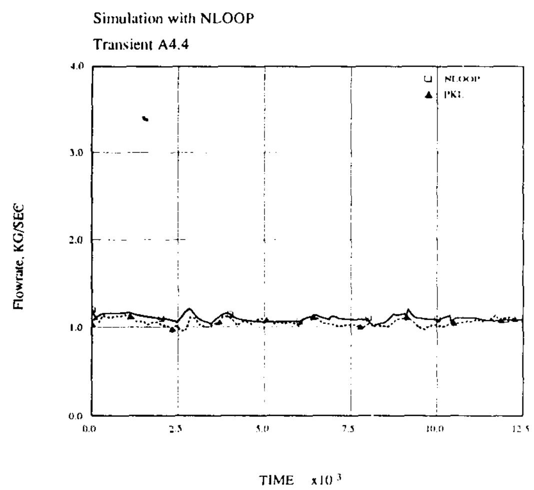

Fig. 7. RCS Flow Rate (Loop 20)

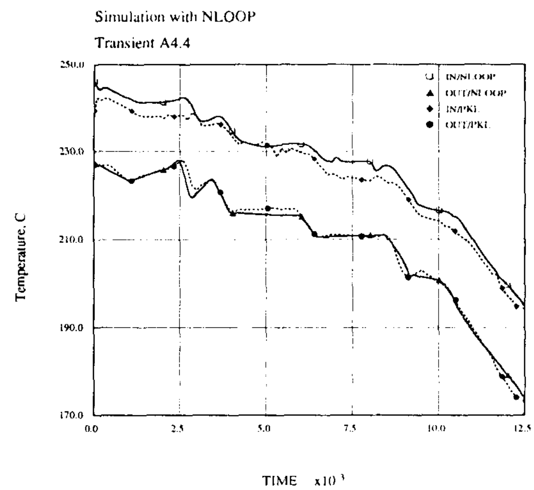

Fig. 9. SG(20) Inlet/Outlet Temperature

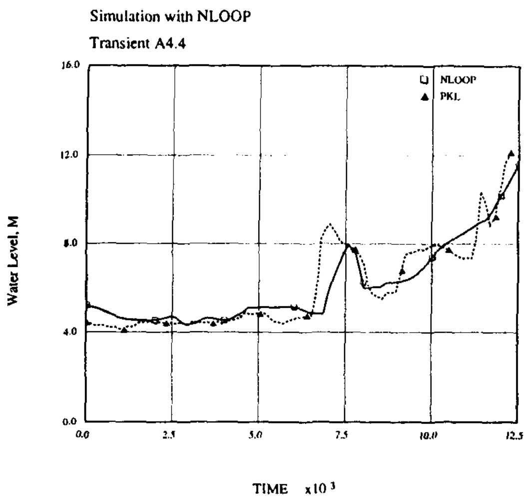

Fig. 10. Pressurizer Water Level

the hot shutdown condition could be cooled down by the natural circulation mechanism until the heat generated in core can be removed by the residual heat removal system. In the NLOOP verification, the RCS mass flow rate caused by natural circulation in the individual loop and the termination of the natural circulation in the isolated loop have to be calculated. In general, good agreements are found between the PKL III experiment and the NLOOP. There is a pronounced deviation in the RCS flow rate of the affected loop and steam generator leak flow because the thermo-hydraulic model in NLOOP is unable to predict the bubble formation in the primary side of the affected steam generator during the event. NLOOP models no phase separation within a node into a separate steam region and a liquid or two-phase region consisting of a continuous liquid phase with dispersed bubbles. It is the limitation of the NLOOP code in this experiment simulation. The PKL III facility is scaled down 1:147 in volume, 1:20 in heat loss surface and 1:75 in steel mass compared with a real plant. Due to the large density at high pressure and thereby great mass of the RCS, the steam generator pressure buildup at pressure of 75 bar in the real plant is quicker than at pressure of 26 bar in the PKL III facility. Since the termination of leak flow from the primary side comes earlier in the real plant, the condensation of steam against the great RCS mass does not matter.

It is recommended for the real plants to reduce the primary pressure by actuation of pressurizer sprays directly after isolation of the defected SG in order to minimize the leak flow and to limit the water level in the defected SG. Since the NLOOP simulation gives good agreement with the experimental data, it enables one to state that the system transient code NLOOP is applicable to simulate the experimental behaviors of the PKL facility for the event of steam generator U-tube rupture.

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