Journal of the Korean Nuclear Society
Volume 23, Number 1, March 1991

⟨Technical Report⟩

Prediction of Thermal-Hydraulic Phenomena in the LBLOCA Experiment L2-3 Using RELAP5/MOD2

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(Received July 10, 1990)

RELAP5/MOD2 코드에 의한 대형냉각재 상실사고 모사실험 L2-3의 열수력 현상 예측

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(1990. 7. 10 접수)

Abstract

The LOFT LOCE L2-3 was simulated using the RELAP5/MOD2 Cycle 36.04 code to assess its capability in predicting the thermal-hydraulic phenomena in LBLOCA of a PWR. The reactor vessel was simulated with two core channels and split downcomer modeling for a base case calculation using the frozen code. The result of the base calculation showed that the code predicted the hydraulic behavior, and the blowdown thermal response at high power region of the core reasonably and that the code had deficiencies in the critical flow model during subcooled-two-phase transition period, in the CHF correlation at high mass flux and in the blowdown rewet criteria. An overprediction of coolant inventory due to the deficiencies yielded the poor prediction of reflood thermal response. Improvement of the code, RELAP5/MOD2 Cycle 36.04, based on the sensitivity study increased the accuracy of the prediction of the rewet phenomena.

요 약

RELAP5/MOD2 Cycle 36.04코드를 이용하여 LOFT 대형냉각재 상실사고 모사실험 L2-3를 계산함으로써 코드의 대형냉각재상실사고에 관련된 열수력현상 예측능력을 평가하였다. 기본계산에서 원자로 압력용기는 이중노심유로와 분리강수관 모델로 모사되었다. 기본계산의 결과 계통의 전반적인 수력학적 거동과 감압기간동안 노심 고출력 부위에서의 열적 거동은 비교적 타당하게 예측되었다. 한편 과냉각-이상유동의 천이 기간동안 임계유량모델, 고질량유속에서의 임계열유속 상관식, 감압기간중의 재접수(Blowdown Rewet)의 판정기준등 코드의 모델/상관식의 부분적 결함이 발견되었다. 이 결함들에 의해 냉각재 재고량이 과대 평가되어 재관수기간의 노심의 열적 거동 예측의 정확도가 감소되었다. RELAP5/MOD2 Cycle 36.04로 부터 개선된 코드를 사용한 계산 결과 재접수 현상의 예측 정확도를 개선할 수 있었다.

1. Introduction

RELAP5/MOD2[1], a frozen version by US Nuclear Regulatory Commission (USNRC), has been assessed through the International Code Assessment and Application Program(ICAP) for its capability and deficiencies in the prediction of the postulated Large Break Loss of Coolant Accident (LBLOCA) in Pressurized Water Reactor (PWR) [2].

The RELAP5/MOD2 Cycle 36.04 was assessed for the Experiment L2-3[3] in this study. The L2-3 experiment, as one of the Integral Effect Test(IET) conducted in the Loss of Fluid Test (LOFT) facility simulated a postulated LBLOCA in the typical Westinghouse type PWR with four loop.

The objectives of this study are to predict the important thermal-hydraulic behavior of the LOFT system during the L2-3 test and to identify deficiencies of the RELAP5/MOD2 Cycle 36.04 code in simulating LBLOCA by comparing the calculation results with the experimental data. From the findings obtained by ICAP activities [4], it was known that the major deficiencies of RELAP5/MOD2 were summarized as the Critical Heat Flux correlations(CHF), the critical flow model and the interfacial drag correlation. This therefore, is focused on confirming whether the deficiencies as stated above are still found in the Experiment L2-3 simulation. During L2-3 experiment, primary coolant pumps were set to continue to operate, which results in the early quenching during blowdown, called rewet[3]. The rewet phenomena known as the early return to nucleate boiling in the core was not well-predicted by the current version of the code according to Aksan [5]. Therefore an updated code from Cycle 36.04 using the modified rewet criteria by Aksan [5] was tested to investigate the sensitivity of the rewet criteria in the RELAP5/MOD2 in this study.

2. Facility and Test Description

The LOFT facility is an experimental 50 MWt PWR designed to simulate LOCA's and anticipated transients and to provide data on the thermal-hydraulic phenomena occuring throughout the system [3]. It is a scaled representation of a commercial PWR of Westinghouse type having 4 loops with a volume ratio of 1/60. Experiment L2-3 represents a postulate 200% double ended offset shear of the pump discharge piping in the cold leg of commercial PWR.

Prior to Experiment L2-3, LOFT facility was set to be a primary system pressure of 15.06 MPa and a loop mass flow rate of 199.7 kg/s. Table 1 presents a summary of initial conditions of Experiment L2-3. The experiment was initiated by opening the Quick Opening Blowdown Valves (QOBV) both in hot leg and cold leg. Reactor was scrammed at 0.103 sec after initiation of the experiment. Sequence of events in the experiment are listed in Table 2.

3. Code and Modelling Description

RELAP5/MOD2 Cycle 36.04, frozen version of the code by USNRC, was used in the present calculation. A nodalization diagram used in the present calculation is shown in Fig.1. The system was discretized by 128 hydrodynamic volumes, 149 junctions and 27 heat structures. The reactor core was modelled by two flow channels: a hot channel representing the central fuel assembly and an average channel representing the peripheral one. Each flow channel was divided into six volumes of an equal length. The reactor vessel downcomer was also modelled by two split flow channels: an intact side downcomer and a broken side downcomer. Each downcomer has five volumes including a volume for an upper annulus part above the cold leg nozzles. Three heat struc-

Table 1 Comparison of the initial conditions

Parameter	Measured	Calculated
Primary coolant mass flow rate*, kg/s	199.0	199.01
Hot leg pressure*, MPa	15.06	15.07
Cold leg temperature*, K	560.7	560.39
Hot leg temperature, K	592.9	592.75
Total power level, MW	36.0	36.0
Maximum linear heat generation rate, kW/m	39.4	39.5
Pressurizer liquid temperature, K	615.3	613.8
Pressurizer pressure, MPa	15.06	15.066
Pressurizer liquid level*, m	1.19	1.1723
Pressurizer water volume, m3	0.67	0.627
S/G secondary saturation temperature, K	482.1	487.88
S/G secondary pressure, MPa	6.18	6.1
Steam mass flow rate, kg/s	19.5	19.129
S/G secondary liquid level*, m	3.11	3.11

Note * : Setpoint in steady state controllers

Table 2 Comparison of the sequence of events

Event	Measured, sec	Calculated, sec
Experiment initiated	0.	0.
End of subcooled blowdown	0.05	**
Reactor scrammed*	0.103	0.103
First indication of DNB	0.96	0.4
End of subcooled break flow (cold leg)	3.0	3.05
Maximum cladding temperature attained	4.95	5.0
Earliest corewide rewet	8.0	7.8
HPSI initiated*	14.0	14.0
Pressureizer emptied	14.0	14.0
Accumulator injection initiated	16.0	15.66
LPSI initiated*	29.0	29.0
Lower plenum refilled	35.0	35.0
Saturated blowdown ended	40.0	**
Accumulator liquid flow ended	45.0	42.0
Core volume reflooded	55.0	43.0

Note * : specified by input, ** : not predicted

ture components were used to model the LOFT fuel rods : 204 hot rods in the central fuel assembly, 572 average rods and 574 intermediate rods in the peripheral assemblies. And the axial power shapes were determined by the experiment for the hot and average rods and by the interpolation for

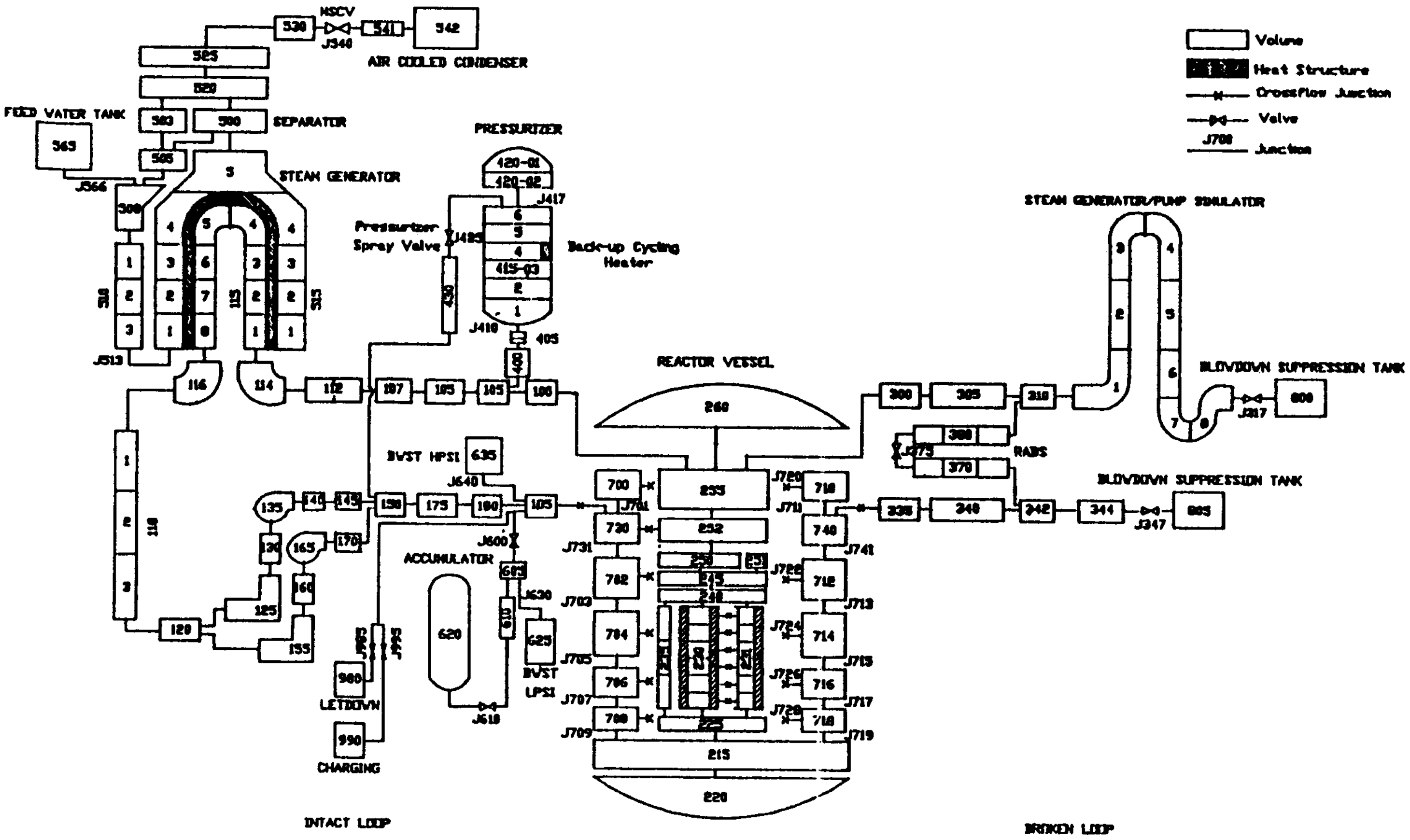

Fig.1 Nodalization diagram for base case calculation of LOFT L2-3 test

the intermediate rods. The power shapes were shown in Fig.2 and Table 3 presents a summary of reactor vessel modelling.

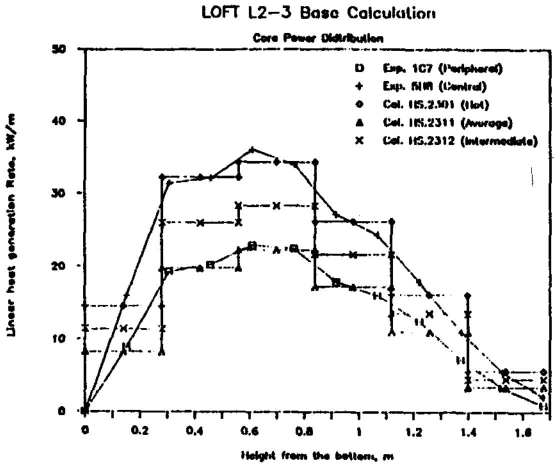

Fig.2 Axial power distribution in base case calculation of L2-3 test

4. Initial and Boundary conditions

To provide all initial conditions of the whole system prior to transient, a steady state run was carried out with seven steady state controllers including two primary pump speed controllers. The result obtained from the steady state run was compared with the measured initial conditions in Table 1. The RELAP5 calculated results generally agree with the experimental conditions.

Among the boundary conditions required to simulate the L2-3 experiment, the PCP speed during the the transient was assumed to be constant to describe the PCP running behavior. The

Table 3 Summary of the important items in nodalization

	Items	Description
Core	Number of flow channel	2
	Channel area ratio (hot/total)	13.7%
	Number of volume per channel	6
	Number of crossflow junction between channels	6
	Loss coefficients at crossflow jucntions	4.2
	Number of heat structure component (hot : average)	1 : 2
	Power ratio(hot/total)	20.5%
	Number of volume per heat structure	6
	Gap conductance model	used
	Reflood option	Pressure*
	Number of mesh point in heat structure	10
	Number of maximum fine mesh	8
	Number of volume in core bypass component	3
	Core Bypass flow ratio	5%
Downcomer	Number of flow channel	2
	Area ratio (intact side channel/broken side channel)	1/1
	Number of volume per flow channel	5
	Number of crossflow junctions for downcomer bypass	5
	Loss coefficients at crossflow junction	18.0–98.0**

Note * : Pressure at core top less than 0.1 MPa

** : Loss coefficients varies with elevation reactor power history and containment pressure were descrihed as time dependent table based on the experiment data [3]. Performance curves for HPSI and LPSI flow rate as function of cold leg pressure were also provided in the input. Feedwater flow rate was reduced to be zero in 2.5 seconds after LOCE initiation using a time–dependent junction. And the steam generator secondary side air–cooled condenser was modelled as a time–dependent volume with a constant pressure of 2.069 MPa during the transient.

5. Base Case Calculation

The L2–3 LOCE was calculated up to 100 sec using all initial conditions obtained from steady state run. The sequence of events found during transient calculation are presented in Table 2 as compared with L2–3 test chronology.

Loop Flow Behavior

Fig.3 shows a comparison of the calculated mass flow rate with the measured one at the broken loop cold leg up to 40 sec. An overall behavall behavior of break flow was well predicted by RELAP5/MOD2 except the underprediction during a short period from 3 sec to 9 sec. This period corresponded to the transition phase from subcooled break flow to two-phase break flow. The underprediction of break flow during the transition period can be considered as a deficiency of the critical flow model in RELAP5/MOD2. This led the reactor vessel to contain more coolant inventory than the experiment and to suppress the core heatup.

Mass flow rate in the intact loop cold leg was shown in Fig.4. The predicted mass flow rate was almost similar to the measured one. However, the calculation did not show several jumps which can be found in the experimental behavior with high frequencies after 20 sec. It was regarded as the effect of void oscillation phenomena induced by the highly subcooled ECC injection water during the reflood phase.

Fig.5 indicates a comparison of the net flow into the core, i.e intact loop cold leg flow minus broken loop cold leg flow. The intact loop cold leg mass flow, driven by the operating pumps, exceeded the broken loop flow from 3 sec to 6 seconds causing an increase in positive core flow, which had reversed after saturation in the lower

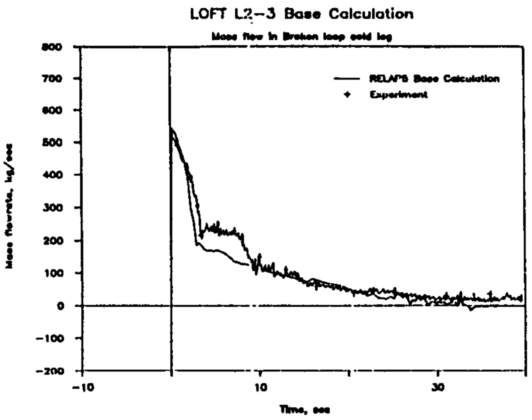

Fig.3 Comparison of mass flow rate at broken loop cold leg between the base calculation and the experiment

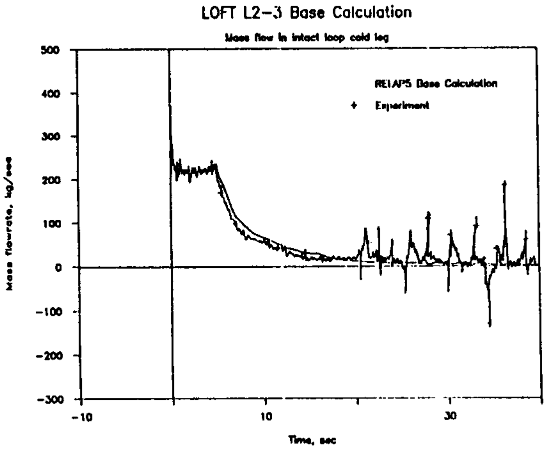

Fig.4 Comparison of mass flow rate at intact loop cold leg between the base case calculation and the experiment

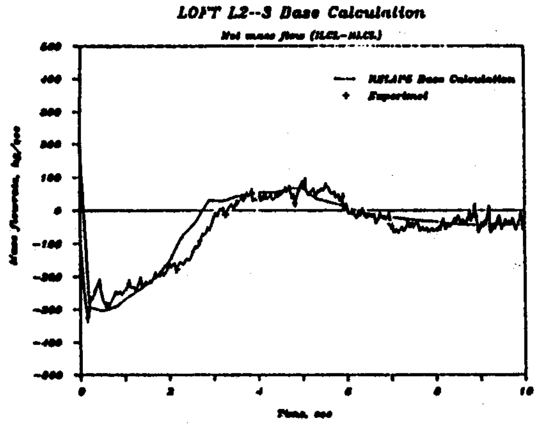

Fig.5 Comparison of net mass flow rate into the core between the base case calculation and the experiment

plenum at about 7 sec. The calculation result was in good agreement with the experimental data.

Vessel Phenomena

Fig.6 shows a comparison of primary system pressure at the upper plenum. The predicted behavior agreed with experiment data very well. A little earlier depressurization was found in calculation than that in the experiment before 10 sec, which resulted partly in an underprediction of cold leg break flow (Fig.3). The calculation also indicated a little lower pressure after 20 sec than the experiment. It resulted in the early high ECC injection flow discussed in the next section.

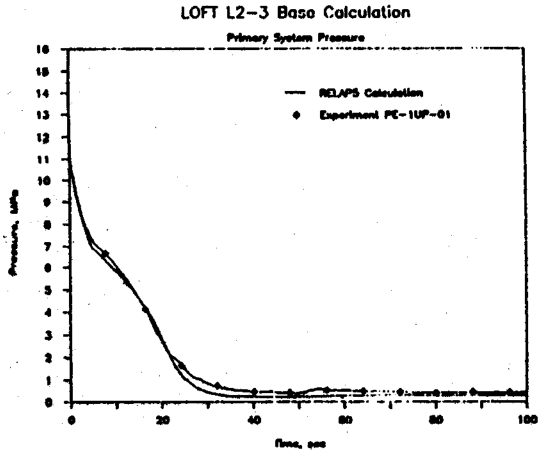

Fig.6 Comparison of primary system pressure between the base case calculation and the experiment

Fig.7 shows the collapsed liquid levels in the two downcomer channels and in the two core channels of reactor vessel. The liquid level behaviors in core were found to be almost identical in both average channel and hot channel. From these level behaviors, it is shown that the LOFT core was almost empty in 5 sec and then filled with liquid up to 2.36m in 8 sec by the positive net flow into the core, i.e rewet. It is also shown that a core re-empty occurred at 14 sec, a lower plenum refill 35 sec and a core-reflood at 43 sec. The end of reflood was predicted to be earlier than the measured time 55 sec in Table 2, which was due to the overestimated coolant inventory, which was mainly caused by under-prediction of cold leg break flow.

The liquid level in the broken side downcomer

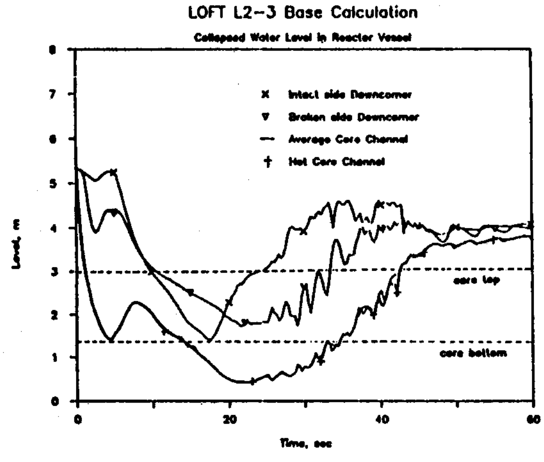

Fig.7 Collapsed liquid levels calculated in the base case calculation for the broken side downcomer, intact side downcomer, hot core channel and average core channel of the reactor vessel

was predicted to be different from that in the intact side one which can be reasoned by a difference in flow resistance. Both levels dropped until 3 sec and then began to increase a little slightly at the time of the transition from subcooled break flow to two-phase break flow. And the incoming flow driven by pumps also increased the downcomer liquid levels during the period. During core rewetting period, downcomer liquid levels decreased again. The slope of the broken side liquid level decrease was slower than that in intact side one, which was due to upward flow from the lower plenum to the broken side downcomer. Downcomer was refilled with ECC injection water from 16 sed, and some time delay and oscillations was found in the level behavior of the broken side downcomer.

ECCS Performance

Fig.8 presents a comparison of acculumator injection flow rate. The predicted injection flow rate was larger before 42 sec and then lower than the experiment. An earlier depressurization of primary system can be a main reason of an high ECC injection flow rate during 20 to 40 sec. The experiment shows that the injection continued until 60 sec, while the calculation shows that the injection was completed at 42 sec. It can be stated that the experimental injection behavior is not a real situation but a measurement error[3].

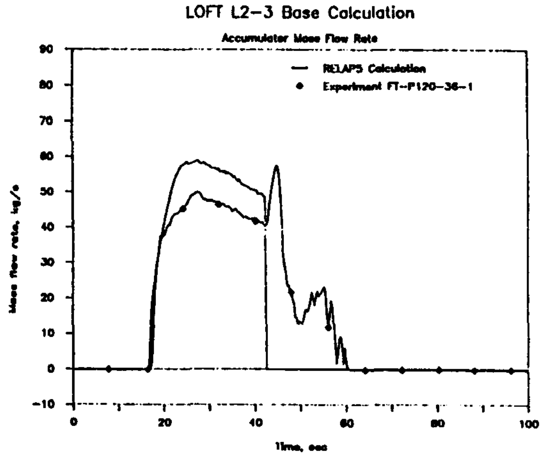

Fig.8 Comparison of accumulator flow rate between the base case calculation and the experiment

Fuel Thermal Response

Fig.9 present a comparison of hot fuel cladding temperature at 27.5 inch from the bottom of core. The PCT was predicted as 950K in blowdown phase, which was higher than the experiment, 891 K. The present calculation also shows no rewet phase and indicates an earlier quench than the experiment.

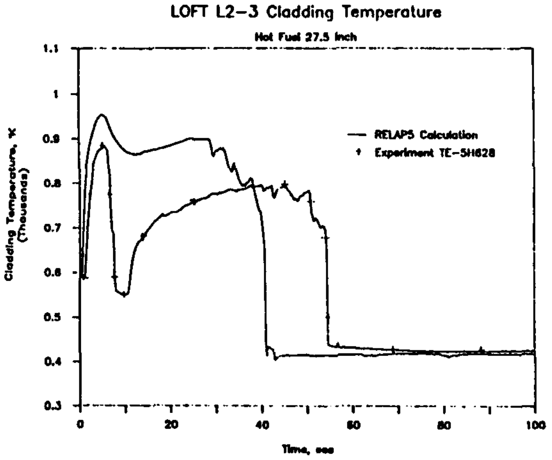

Fig.9 Comparison of cladding temperature at 27.5 inches of hot fuel between the base case calculation and the experiment

Fig.10 shows a comparison of cladding temperature at 16.5 inch elevation. It is shown that the RELAP5/MOD2 calculation led to a later DNB and a less heat up in reflood phase than the experiment. The reason of the discrepancy in the thermal response during the reflood phase was considered as an overpredicted coolant in- ventory, i.e less break flow and more ECC flow than the experiment. The delayed DNB at low power region of the hot fuel was regarded as a deficien-

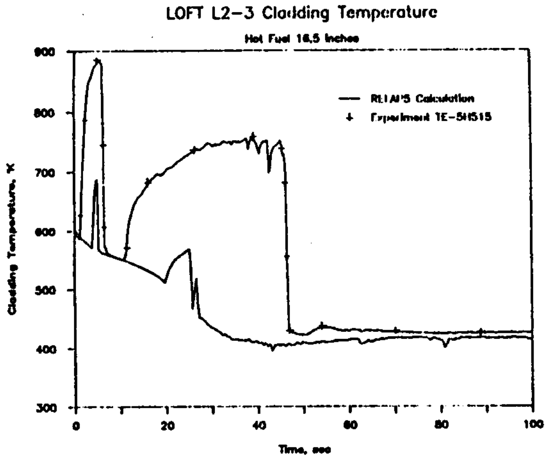

Fig.10 Comparison of cladding temperature at 16.5 inches of hot fuel between the base case calculation and the experiment

cy in Biasi CHF correlation for high mass flux in the frozen code. The absence of rewet at the high power region was due to a severeness of the current rewet criteria.

5. Sensitivity Study

The effect of rewet criteria on the thermal response was tested by recalculating the L2–3 transient with an updated RELAP5/MOD2 code including a modified rewet criteria. This criteria was updated by Aksan, PSI[5] as follows

$T_w + T_{sat} < 1250$ K, and

Equilibrium quality, $X_e < 0.99$

or Mass flux, $G < -100$ kg/ m^2 sec

Figures 11 and 12 show comparisons of the calculated cladding temperatures by the frozen Cycle 36.04 with those by PSI updated version at 27.5 and 39 inch elevation, respectively. The sensitivity calculation shows the blowdown rewet evidently, which were not predicted in the base case calculation. As the result calculated by the PSI–updated version, the core heatup was a little higher than the base case calculation result. It was due to some modifications of heat transfer coefficients in the subcooled or saturated transition film boiling regime included in the PSI updates[4]. It can be, however, stated that an improvement of the rewet criteria did not yield a accurate reflood thermal reponse and that the CHF correlation was the major factor as shown in Fig.12.

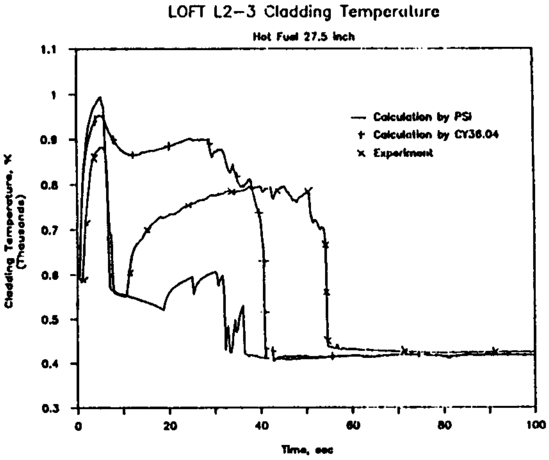

Fig.11 Comparison of cladding temperature at 27.5 inches of hot fuel between the base case calculation and the sensitivity calculation

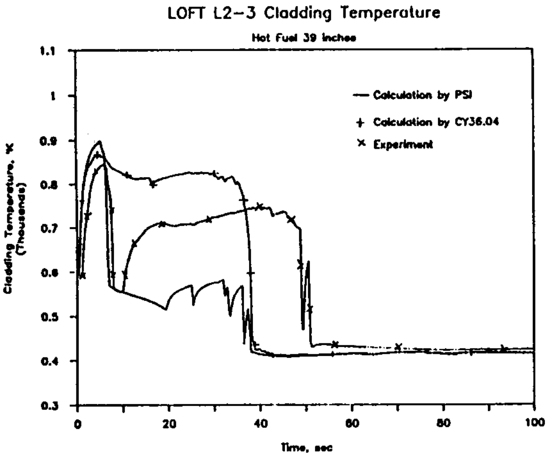

Fig.12 Comparison of cladding temperature at 39 inches of hot fuel between the base case calculation and the sensitivity calculation

7. Conclusions

RELAP5/MOD2 Cycle 36.04 code was assessed using LOFT L2-3 LBLOCE data. A base case calculation was carried out using the original version of the code. And one sensitivity calculation was conducted with an updated version by the PSI modification. As a result of the present calculations, the following conclusions are obtained :

1) The loop flow behavior was well-predicted by the frozen code. The cold leg break flow was underpredicted due to a code deficiency of the critical flow model during the transitional period from subcooled to two-phase flow. The reactor vessel hydraulic behavior was reasonably predicted by the frozen code and ECCS performance were also well simulated in the base calculation.
2) The thermal response of the high power region of the core was well-predicted in the base case calculation during blowdown period, and the predicted PCT was 950 K, a little higher than the measured PCT of 891 K. The rewet phenomena was not predicted by the frozen code. The thermal response of the low power region was poorly predicted due to the deficiency of the Biasi CHF correlation at high mass flux.
3) Generally the thermal reponse during the reflood phase was poorly predicted mainly due to misprediction of coolant inventory and deficiency in CHF correlation of the current frozen code.
4) The modified rewet criteria was effective in predicting the rewet phenomena.

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