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Analysis of Total Loss of Feedwater Event for the Determination of Safety Depressurization Bleed Capacity

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안전감압계통의 방출유량을 결정하기 위한 완전급수상실사고 해석

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Abstract

The Ulchin 3&4, which are 2825 MWt PWRs, adopted Safety Depressurization System (SDS) to mitigate the beyond design basis event of Total Loss of Feedwater(TLOFW). In this study the results and methodology of the analyses for the determination of SDS bleed capacity are discussed.

The SDS design bleed capacity has been determined from the CEFLASH-4AS/REM simulation according to the following design criteria : 1) Each SDS flow path, in conjunction with one of two High Pressure Safety Injection (HPSI) pumps, is designed to have a sufficient capacity to prevent core uncovery if one SDS path is opened simultaneously with the opening of the Pressurizer Safety Valves (PSVs). 2) Both SDS bleed paths are designed to have sufficient total capacity with both HPSI pumps operating to prevent core uncovery if the Feed and Bleed (F&B) initiation is delayed up to thirty minutes from the time of the PSVs lift.

To verify the results of CEFLASH-4AS/REM simulation a comparative analysis has also been performed by more sophisticated computer code, RELAP5/MOD3. The TLOFW event without operator recovery and TLOFW event with F&B are analyzed. The predictions by the CEFLASH-4AS/REM of the transient two phase system behavior are in good qualitative and quantitative agreement with those by the RELAP5/MOD3 simulation. Both of the results of analyses by CEFLASH-4AS/REM and RELAP5/MOD3 have demonstrated that decay heat removal and core inventory make-up can be successfully accomplished by F&B operation during TLOFW event for the Ulchin 3&4.

요 약

2825 MWt 가압경수로인 울진 3, 4호기에는 설계기준초과사고인 완전급수상실사고를 완화하기 위하여 안전감압계통이 채택되었다. 본 논문은 울진 3, 4호기의 안전감압계통의 방출유량을 결정하기 위한

해석방법 및 결과에 대하여 논의하였다.

안전감압계통의 방출용량을 다음과 같은 두가지의 설계요건에 따라 결정하였다 : 1) 두 개의 고압안전주입펌프 중 하나의 펌프만이 작동하고 운전원이 안전감압계통의 한 계열의 감압경로를 가압기안전밸브가 열리자마자 개방하였을 경우 노심노출을 방지하여야 한다. 2) 두 개의 고압안전주입펌프가 모두 작동하고 두 계열의 안전감압경로를 가압기안전밸브가 열린 후 30분 뒤에 개방하였을 경우 노심노출을 방지하여야 한다.

CEFLASH-4AS/REM 전산코드의 모델 및 해석 결과의 타당성을 검토하기 위하여 RELAP5/MOD3를 이용한 해석을 수행하였다. 운전원의 복구과정이 없을 경우와 운전원이 충전 및 유출운전에 의해 사고를 완화하는 경우의 완전급수상실사고 경위에 대해 수치모사를 수행하였다. 두 사고 경위에 대해 CEFLASH-4AS/REM에 의해 예측된 원자로계통의 주요 열수력학적 거동이 RELAP5/MOD3에 의한 결과와 정성 · 정량적으로 잘 일치하는 것을 알 수 있었다. 결론적으로 울진 3, 4호기에 대해 완전급수상실사고시 안전감압계통을 이용한 충전 및 유출운전에 의해 잔열제거 및 일차계통 냉각재 재고량 유지가 성공적으로 이루어짐을 수치모사를 통해 확인 할 수 있었다.

1. Introduction

Following the Three Mile Island accident, the potential ability of Power Operated Relief Valves (PORVs) to provide an alternate method to remove decay heat from the primary system was identified and considered to be beneficial in dealing with severe accidents. Recent studies [1, 2, 3, 4] have concluded that F & B can be a viable alternate means of decay heat removal, but successful use of F & B is contingent upon the implementation of proper procedures, as well as upon the specific plant design. ABB-CE's latest plant design, System 80+[5], includes manual bleed valves to provide the F & B capability according to the USNRC's Severe Accident Policy.

The Korea Atomic Energy Research Institute (KAERI) is performing detailed design of the SDS similar to that of System 80+. In particular, manually-actuated bleed valves are designed to provide a capability to rapidly depressurize the Reactor Coolant System (RCS) for TLOFW event. Presented in Reference 6 are the preliminary results of thermal-hydraulic analyses of TLOFW for the Ulchin 3&4, which were performed by CEFLASH-4AS/REM [7]. In this study provided are the final results and methodology of the thermal-hydraulic analyses to determine the Ulchin 3&4 SDS bleed capacity. Also an alternate analysis by more sophisticated computer code RELAP5/MOD3[8] is performed to verify the results and methodology used for Ulchin 3&4 SDS design.

2. Plant Description and Initial Conditions

The Ulchin 3&4 are two loop 2825 MWt PWRs. The plant nodal diagrams for CEFLASH-4AS/REM and RELAP5/MOD3 are provided in Figures 1 and 2. The Ulchin 3&4 are designed with two cold legs per loop and thus contain four reactor coolant pumps. The SDS consists of two separate lines connected to the top head of the pressurizer and the flow through each line discharges to the containment atmosphere through a rupture disc. The two bleed paths consist of an isolation valve and control valve in series per path, and provide redundant paths.

The plant initial conditions are assumed at full power steady state nominal conditions. Table 1 provides major plant parameters. Also provided are steady state initial conditions obtained by two computer codes. The results of initialization indicate that the two initial conditions are essentially same.

3. Analyses Methodology

3.1. Design Criteria

The use of F & B is a trade-off between allowable time before operator action and the bleed capacity of the system. The longer the time, the larger the system capacity must be. A shorter allowable time before operator action increases the possibility of inadvertent actuation and resultant containment contamination. Therefore, appropriate design criteria are requiured. Followings are design criteria selected for the Ulchin 3&4 : 1) Each SDS flow path, in conjunction

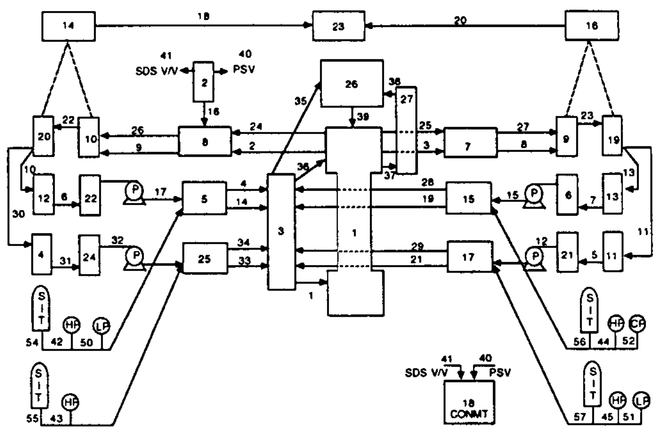

Fig. 1. The Nodalization Scheme of CEFLASH-4AS/REM for UCN 3&4

Table 1. Plant Initial Conditions and Major Plant Parameters

a. Plant Initial Conditions and Steady State Value		
Parameter	Design Value	Steady State : CEFLASH/RELAP5
Core power (MWt)	2815	2815/2815
RCS pressure (MPa)	15.5	15.5/15.48
RCS flowrate (ton/hr)	55113	55113/55113
Cold leg temperature (°C)	295.8	295.8/296.7
Hot leg temperature (°C)	327.3	327.3/327.7
SG pressure (MPa)	7.5	7.2/7.27
RCS Inventory (Kg)	N/A	211000/216200
SG Inventory at Rx trip(Kg)	N/A	41400/41100

b. Major System Parameters	
Primary side volume (m^3)	329.4
Pressurizer volume, liquid/total (m^3)	25.5/51.4
Low SG level reactor trip setpoint (% WR)	38.5
SIAS setpoint (MPa)	12.6
HPSI pump shutoff head (MPa)	12.65
PSV setpoint (MPa)	17.2
PSV capacity (steam at 17.2 MPa), per valve (kg/hr)	247517
Number of PSVs	3
Analytical bleed area (m^2)	0.0026

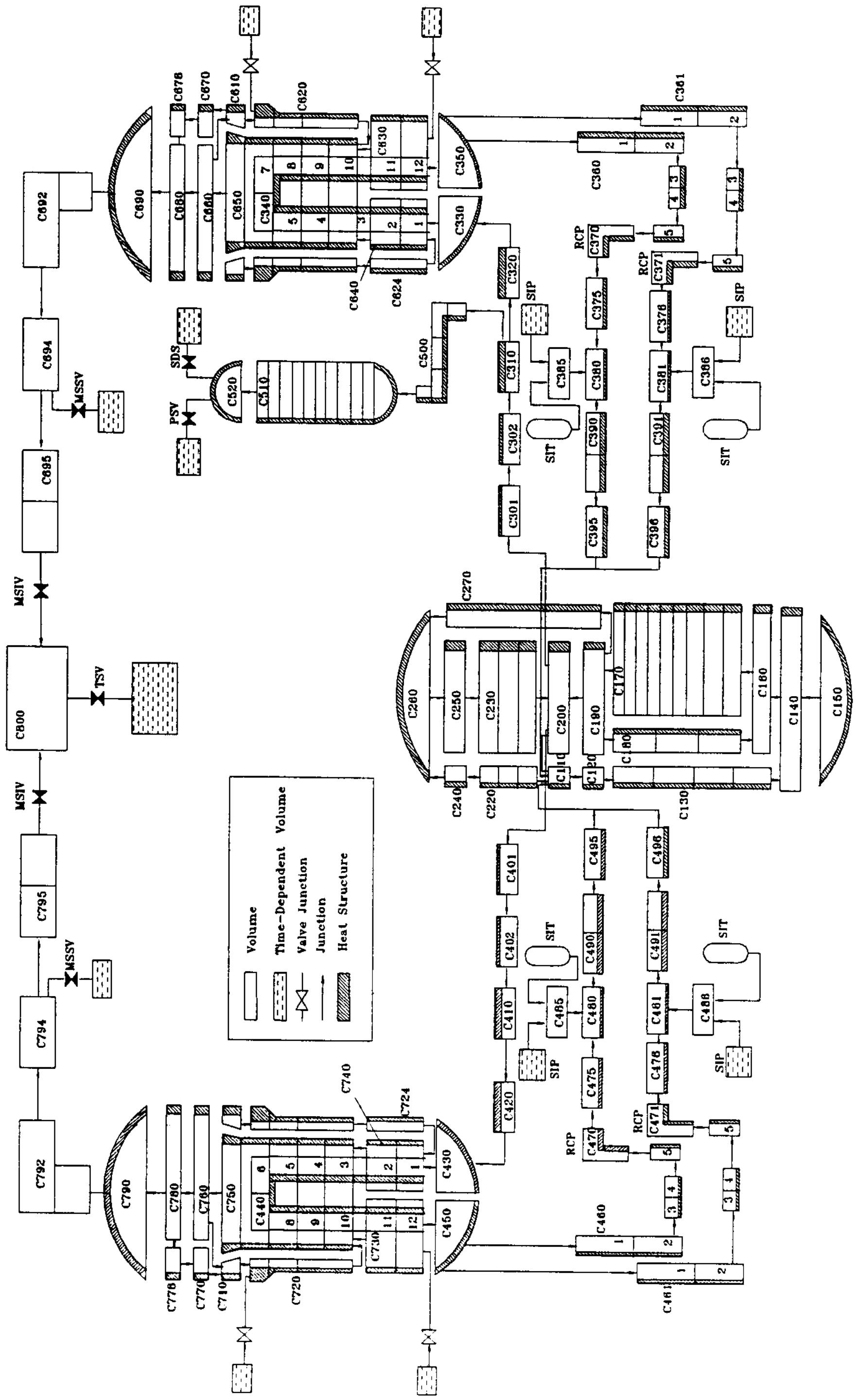

Fig. 2. The Nodalization Scheme of RELAP5/MOD3 for UCN 3&4

with one of two HPSI pumps, is designed to have a sufficient capacity to prevent core uncovery following a TLOFW if one SDS path is opened simultaneously with the opening of the PSVs. 2) Both SDS bleed paths are designed to have sufficient total capacity with both HPSI pumps operating to prevent core uncovery following a TLOFW event if the F&B initiation is delayed up to thirty minutes from the time of the PSVs lift.

The analysis procedure for the bleed capacity starts with a base case in which the bleed paths are not available, i.e., no operator action is assumed. This base case yields the time of PSVs lift and core uncovery. The duration between PSVs lift and core uncovery is the maximum theoretical allowable time for the operator to open the bleed paths to prevent the core uncovery. All subsequent cases are analyzed with F&B operation. The analytical bleed path area required to prevent core uncovery were investigated in conjunction with operator action time for each F & B cases.

3.2. Differences in Analytical Models

This analysis employs two analytical models, CEFLASH-4AS/REM computer code developed by ABB-CE and RELAP5/MOD3 computer code version 3.1 developed by INEL. CEFLASH-4AS/REM has been improved from the CEFLASH-4AS[9] which is used for licensing analysis of small break LOCAs. Reference 10 provides the validation of the CEFLASH-4AS/REM against experimental data to verify the capability of the code for use in the analysis of a TLOFW event with F & B. The CEFLASH-4AS/REM (simply, CEFLASH) employs two mass, two energy, and one mixture momentum equations. Since the CEFLASH solves only mixture momentum equation, various constitutive relations using drift flux model are employed. RELAP5/MOD3 (simply, RELAP5) employs two-fluid, nonequilibrium, nonhomogeneous, hydrodynamic model (six equations) for the transient simulation of the two-phase system behavior.

Like all other computer codes, RELAP5 and CEFLASH are limited by the phenomenological models built into the codes. In addition, RELAP5 and CEFLASH have different nodalization scheme; RELAP5 permits the user to vary the nodalization. On the other hand, the CEFLASH has a customized nodalization scheme as shown in Fig. 1. To provide the basis of comparison and to assess the uncertainties embodied in the simulation results, discussed in the following are the major models which will affect the simulation of TLOFW scenario.

Break Flow Model: The CEFLASH critical flow model is based on the Henry-Fauske (H-F) correlation, Homogeneous Equilibrium Model (HEM), and Murdock-Bauman correlation for subcooled liquid, two phase, and steam flow, respectively. The RELAP5 employs two-phase choking criteria based on the characteristic analysis of two-fluid model equations [11]. With the same analytical bleed area the discharge coefficients of RELAP5 are adjusted to predict the same discharge flow rate as CEFLASH depending on the upstream condition to provide the equal basis of comparison.

Phase Separation Model: The accurate prediction of phase separation is of particular importance in the analysis of TLOFW transients. The rate at which steam disengages from the two-phase region affects the degree and duration of core uncovery. The phase separation model also affects the upstream condition of pressurizer safety valve and bleed valve. In the CEFLASH the drift flux model is used for determining the steam flow rate for a given void fraction and flow regime. Especially, the drift velocity for inner vessel is determined based on the correlations developed with available test data run at the Thermal-Hydraulic Test Facility of the Oak Ridge National Laboratory[12]. On the other hand, the liquid and vapor phases are mixed together in each node in the RELAP5.

Countercurrent Flow and Countercurrent Flow Limit(CCFL) Model : During the steam generator draining phase of TLOFW event, countercurrent flow is expected such that the steam exiting the core flows through the hot leg toward the steam generator and the low quality two-phase mixture from the steam generator U-tubes drains toward the reactor vessel. In the CEFLASH two flow paths are provided for the reactor vessel to hot leg, hot leg to steam generator, and cold leg to reactor vessel to account for the counter-current flow. Also provided are Wallis-based CCFL models for vertical and horizontal flow regimes[13]. The countercurrent flow is not considered in the surge line path in the CEFLASH. On the other hand, the countercurrent flow situation is mechanistically modeled in RELAP5 by the separate momentum equations for the steam and liquid phases.

Entrainment Model for Surge Line and Bleed Path : When a small break (SDS bleed path and surge line are treated as breaks) is located above the horizontal surface, liquid can be entrained in the break flow as a consequence of flow regime transitions, or due to vapor acceleration in the vicinity of the break. In the CEFLASH a simple entrainment model based on the liquid level and a criterion for the stability of a small disturbance is provided and used for the surge line and SDS bleed path. The RELAP5 has a horizontal stratification model similar to that of CEFLASH, which is applied to horizontal pipe such as cold leg and hot leg. Specific entrainment model is not provided for the vertical geometry such as pressurizer. Therefore, the break discharge flow quality in the pressurizer will depend on the condition of donor cell in the RELAP5 simulation.

4. Simulation Results and Discussions

4.1. Determination of Bleed Capacity

CEFLASH is used for the simulation of TLOFW event without operator recovery and TLOFW event with F&B. The assumptions used in the simulation of these transients are : 1) The plant initial conditions are at steady state full power condition. 2) A reactor trip occurs due to low steam generator level 30 seconds after event initiation. 3) The Reactor Coolant Pumps (RCPs) are tripped 10 minutes after the reactor trip per Emergency Procedure Guidelines[14]. The 10 minutes operator action time is based on the fact that the operator should trip all the RCPs in the Optimal Recovery Guideline (ORG) for the Loss of All Feedwater. For the TLOFW event like scenarios the operator can diagnose the event as Loss of All Feedwater easily, since the system pressure will increase rapidly in couple of minutes after reactor trip due to heat transfer degradation. Since the operator can trip the RCPs in the main control room, the 10 minutes operator action time can be justified. 4) The operator actions are considered according to the design criteria discussed in section 3.1.

For the F&B cases the single failure case and no failure case are considered, where single failure implies operation of only one HPSI pump and opening of one SDS bleed path and no failure means operation of two HPSI pumps and opening of two bleed

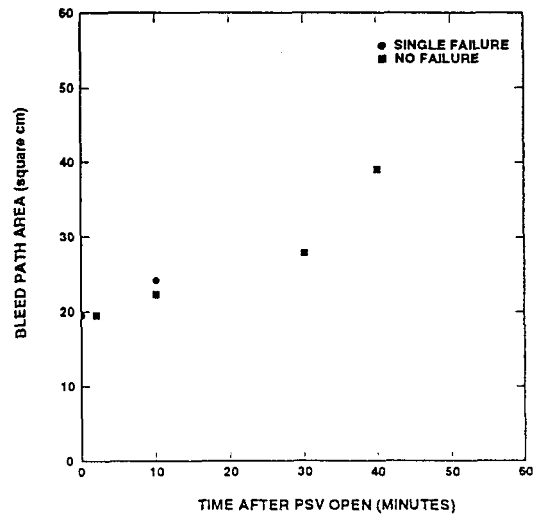

Fig. 3. Analytical Bleed Areas Required to Prevent Core Uncovery for Various Operator Times

paths. The analytical bleed capacity to prevent core uncovery are investigated by varying the analytical bleed area in TLOFW simulations. The operator actions coincident with PSVs lift and 10 minutes after the PSVs lift are considered for the single failure case. For the no failure case 2 minutes, 10 minutes, 30 minutes, and 40 minutes are considered. Figure 3 shows analytical bleed area required to prevent core uncovery for various operator action times for the single failure case and no failure case determined by CEFLASH.

Since the analytical bleed area to meet the second design criterion (28 cm^2) is smaller than twice the analytical area to satisfy the first design criterion (40 cm^2) as can be seen in Figure 3, the first design criterion is more restrictive with respect to bleed capacity. Theses results are in the same trend as the preliminary analyses results presented in Reference 6. However, current analysis does not assume charging pumps operation, since Chemical and Volume Control System (CVCS) including charging pumps is not credited as safety system. The design bleed capacity is selected as 26 cm^2 by accounting for the various uncertainties, such as, valve stroke time, decay heat curve, and code uncertainties.

To verify the results and methodology of CEFLASH analyses, comparative analyses have also been performed by RELAP5. The cases selected for presentations are TLOFW without recovery and single failure case. In the next sections discussions are focused on the results of simulations using 26 cm^2 bleed capacity.

4.2. TLOFW Without Recovery

Table 2 provides major chronology of the event predicted by CEFLASH and RELAP5. Fig. 4 shows pressurizer pressures predicted by the CEFLASH and RELAP5. Following reactor trip the RCS pressure drops due to a sudden decrease in heat generation from the core. After a short time period, the RCS pressure starts to rise in response to the power-to-flow mismatch and reaches to a new steady state. After the RCP trip at 630 seconds, the pressurizer pressure increases more rapidly due to RCP coast down. Fig. 5 shows steam generator inventory. When both steam generators dry out at about

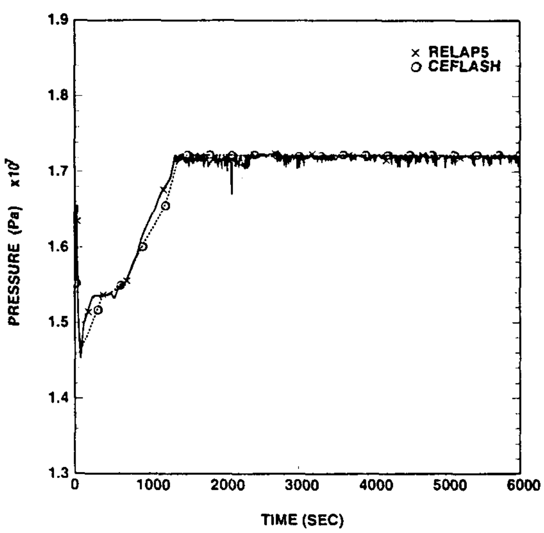

Fig. 4. Pressurizer Pressure (TLOFW w/o Recovery)

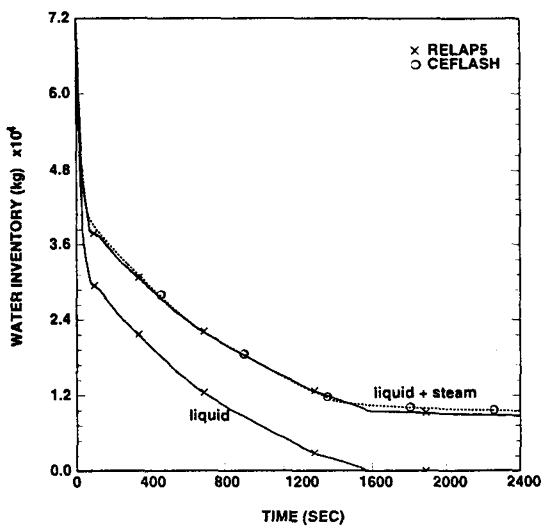

Fig. 5. Steam Generator Inventory (TLOFW w/o Recovery)

Table 2. Chronology of the TLOFW Event

	TLOFW w/o Recovery (CEFLASH/RELAP5)	TLOFW with F&B (CEFLASH/RELAP5)
Bleed Area (m^2)	0	0.0026
Feed flow	No HPSI	1 HPSI
	Time (seconds)	
Event		
Total loss of feedwater	0/0	0/0
Reactor trip	30/30	30/30
RCP trip, manual	630/630	630/630
Steam generator dryout	1360/1600	1360/1600
PSVs open	1389/1345	1389/1345
SDS bleed path(s) opens	N/A	1389/1345
HPSI flow on	N/A	1511/1385
Hot leg saturation	2923/2875	1510/1420
Core uncovery begins	5296/–	N/A
or		
Minimum RV inventory, kg		47300/44300
occurred at, sec		3280/3600

1360/1600 seconds (The dryout time can be determined from the liquid inventory in the RELAP5 simulation. Since only mixture inventory has physical meaning in the CEFLASH simulation, the time when the mixture inventory flattens out corresponds to dry out time in the CEFLASH simulation), the RCS volume expansion and pressurization is accelerated. Then the pressurizer pressure reaches the PSVs opening setpoint. Since the PSVs have enough capacity to accommodate the increased volumetric expansion, the pressurizer pressure is maintained around PSVs setpoint during the whole transient. It is shown that the pressurizer pressures are in good agreement between two simulations.

The primary temperature rises until it reaches the saturation temperature corresponding to the pressurizer safety valve setpoint as shown in Fig. 6. From that time on, the primary temperature stays constant while void is generated in the RCS.

Pressurizer goes solid at about 2500 seconds as shown in Fig. 7, and the discharge flow becomes single phase liquid. After the RCS reaches saturation condition around 3000 seconds, steam generated in the core due to decay heat flows from the core to pressurizer via hot leg and surge line. Once the surge line begins to draw vapor, the net inventory in the pressurizer drops rapidly because low quality mixture is still flowing out of safety valves. Fig. 8 shows the integrated surge flow and PSV discharge flow. The integral surge flows predicted by CEFLASH and RELAP5 are in good agreement before hot leg is highly voided. After that, two predictions deviate slightly, which might be due to the absence of countercurrent flow model for the surge line in the CEFLASH.

Core uncovery begins at around 5300 seconds into the transient, when the RCS inventory becomes so low that the void fractions in the top three nodes of the core reache 1. 0 in the RELAP5(refer to Fig. 9 for Reactor Vessel (RV) liquid inventory and Fig. 10 for core void fraction). At that time, the collapsed water level in the core begins to decrease rapidly and the cladding and core outlet temperatures begin to rise. It is observed that the vapor fractions in the core region tend to hang up in the 25~40% range for extended period of time as shown in Fig. 10. When core inventory decreases, it frequently takes place in such a manner that the local vapor fraction jumps from 0.4 to 1.0 over a very brief time interval as shown in Fig. 10. This is purely a result of the flow regime map change in RELAP5. As shown in Fig. 9 the RV inventory behavior is quite different between two predictions after the RCS starts to void. This difference in RV inventories is judged to be from the difference in inventory distribution among various RCS components including RV upper head, pressurizer, hot legs, cold legs and SG U-tubes. It is observed that the drainage of RV upper head inventory after hot leg voiding predicted by RELAP5 is much faster than that predicted by CEFLASH.

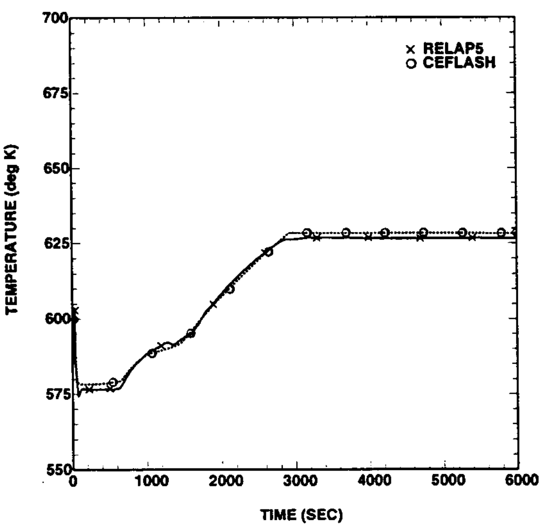

Fig. 6. Hot Leg Temperature (TLOFW w/o Recovery)

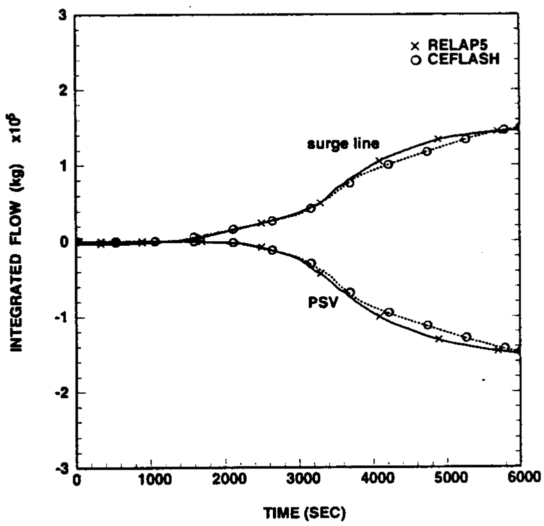

Fig. 8. Integrated Surge and PSV Flows (TLOFW w/o Recovery)

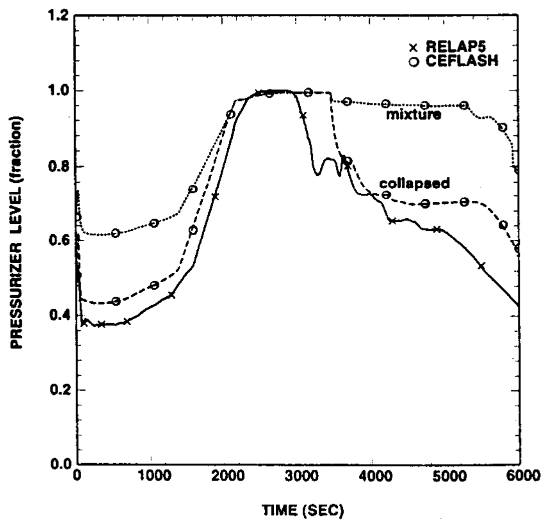

Fig. 7. Normalized Pressurizer Level (TLOFW w/o Recovery)

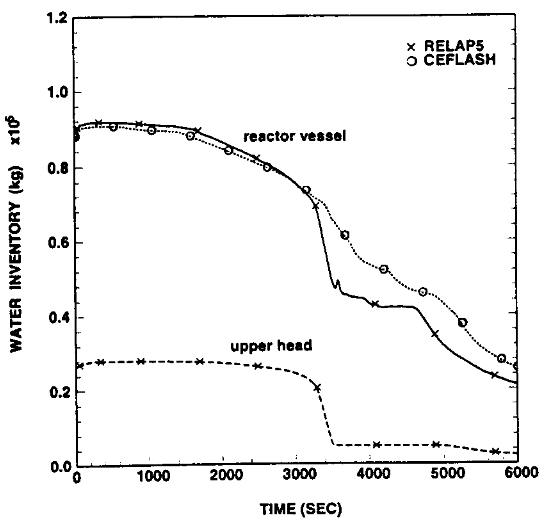

Fig. 9. Reactor Vessel Inventories (TLOFW w/o Recovery)

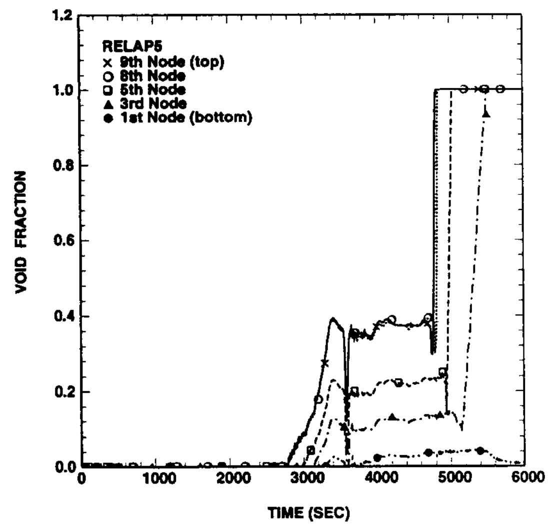

Fig. 10. Core Void Fraction (TLOFW w/o Recovery)

4.3. TLOFW with Feed & Bleed

Presented in this subsection are the results of the case where F & B operation is utilized to attempt to cool the core and make up the RCS inventory. The assumptions used in the simulation of this transient are : 1) Operator opens one train of SDS bleed path and aligns one train of HPSI pump for injection at the time of PSVs' lift. 2) The SDS is modeled by an orifice located on the top of the pressurizer whose analytical bleed area corresponds to 26 cm^2.

This case is identical to TLOFW without recovery case until PSVs lift. Table 2 provides chronology of major event scenarios predicted by CEFLASH and RELAP5. Soon after the bleed path is opened the RCS pressure decreases rapidly as shown in Figure 11, and hence HPSI injection flow is initiated at 1511/1385 seconds. A significant amount of energy is removed through the SDS bleed path when the discharge flow is a single phase steam. However, as the flow leaving the SDS path becomes two-phase, the energy removal slows down. Hence the RCS pressure decrease also slows down. And the RCS pressure briefly begins to repressurize when the discharge flow becomes single phase liquid. At that point the pressurizer becomes solid (refer to Fig. 11, 12 and 13). The pressurizer behavior during repressurization is generally in good agreement between CEFLASH and RELAP5. CEFLASH shows rather smooth pressurizer pressure transient. The pressurizer inventory keeps increasing after 4200/5400 seconds as can be seen in Fig. 12.

The RCS becomes saturated quickly after the bleed valves open. The steam generated in the core migrates from the core to the pressurizer, which increases the amount of steam bubbles in the pressurizer and consequently increases break flow quality. As the quality of bleed flow increases, the energy removal rate increases. This results in the decrease in the pressurizer pressure. Therefore, the HPSI injection flow is reinitiated (see Fig. 14). As the RCS depressurizes, HPSI flow with low temperature at 50°C also increases, which contributes to further reduction of RCS pressure as shown in Fig. 11. The combination of opening the SDS bleed path, which results in loss of RCS inventory, and the HPSI injection of cold fluid, which lowers the RCS average temperature and therefore leads to contraction of RCS fluid, eventually causes voiding in the RCS. Void formation is evident in the core as early as 2000 seconds into the transient. However the void fraction at the top of the core is maintained below 40% due to increased HPSI injection flow (see Fig. 15). The reactor vessel inventory reaches minimum at 3600 seconds in RELAP5 predictions and continuously increases as shown in Fig. 16. The CEFLASH prediction shows that core mixture level is always maintained above the top of the core as shown in Fig. 17. This result indicate that the selected SDS bleed capacity meets the first design criterion discussed in section 3.1.

As shown in Fig. 16 the reactor vessel inventory behavior is quite different between two predictions after RCS starts to void. This difference in RV inventories is judged to be from the differences in inventory distribution among various RCS components.

Therefore, further study on the nodalization scheme of RV upper head is recommended. However, this difference does not affect the conclusion of this analysis, since the event scenarios before the time of minimum RV inventory are almost identical for both cases and both predictions show that core is covered two-phase mixture.

The peak cladding temperature is calculated to evaluate the impact of core voiding. As shown in Fig. 18 the cladding temperature is well below acceptance criteria, which assures core to coolant heat transfer is well maintained.

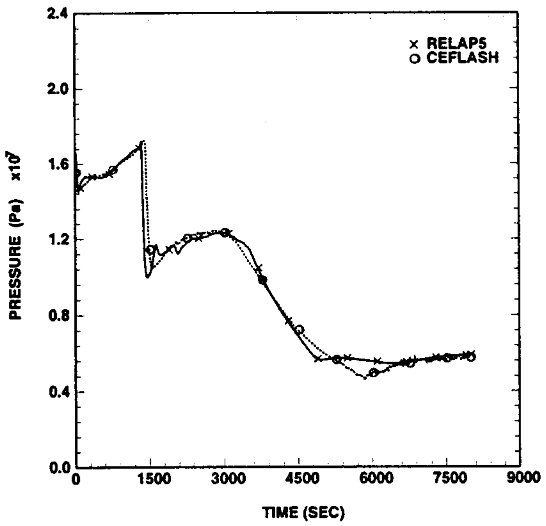

Fig. 11. Pressurizer Pressure (F & B)

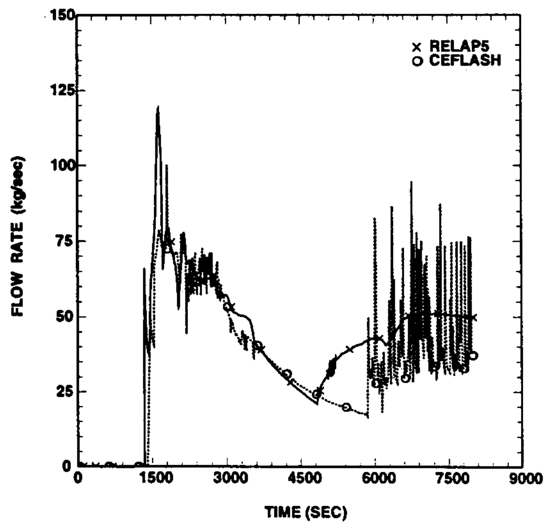

Fig. 13. Bleed Path Discharge Flow Rate (F & B)

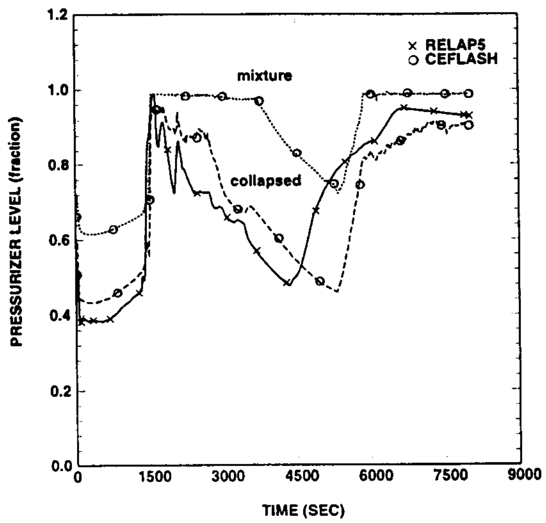

Fig. 12. Normalized Pressurizer Level (F & B)

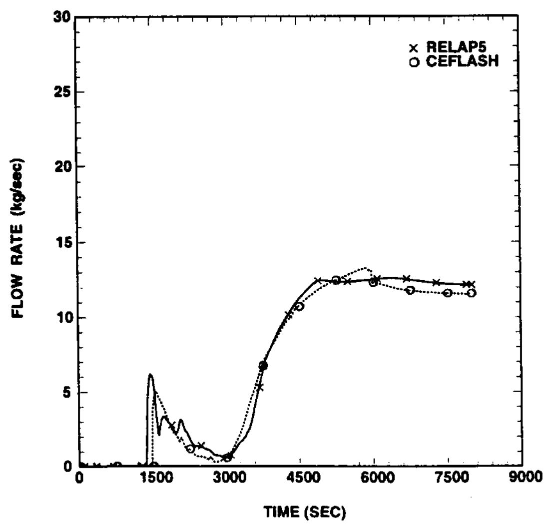

Fig. 14. HPSI Flow (F & B)

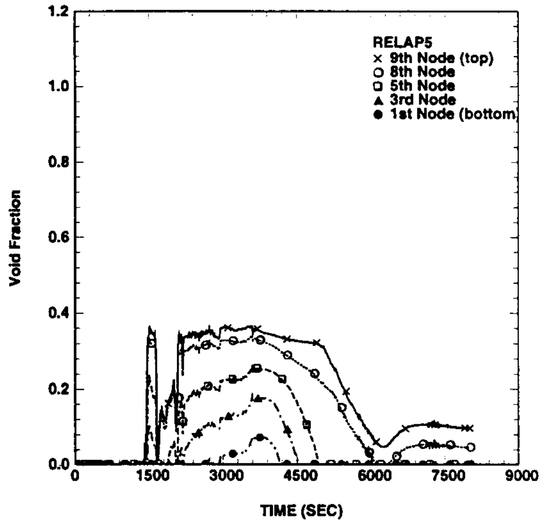

Fig. 15. Core Void Fraction (F & B)

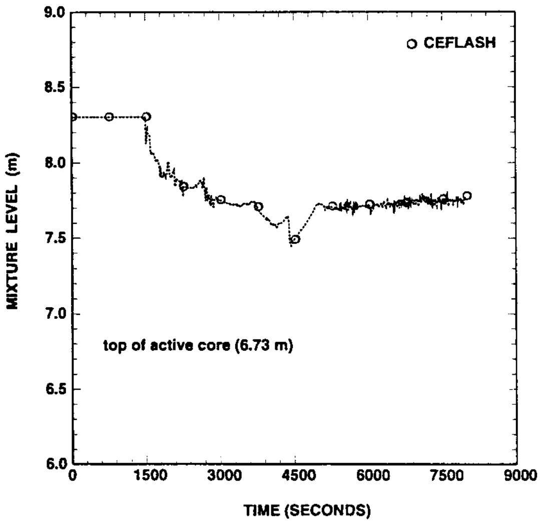

Fig. 17. Core Mixture Level (F & B)

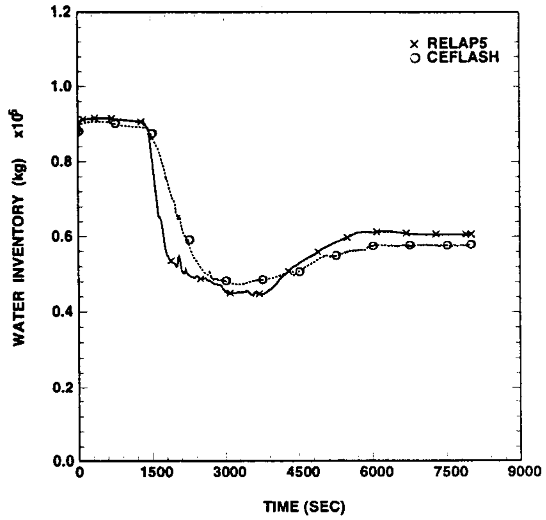

Fig. 16. Reactor Vessel Inventory (F & B)

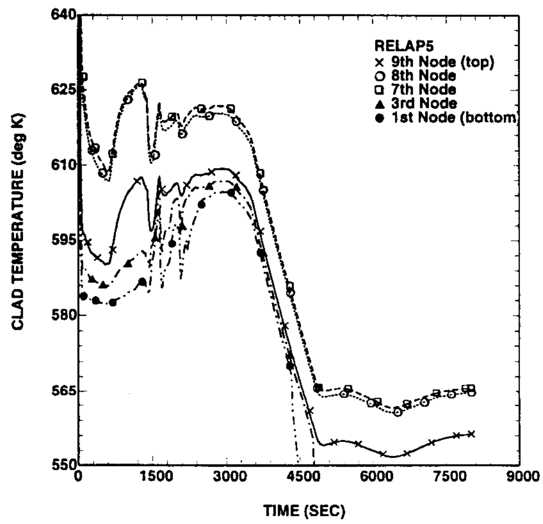

Fig. 18. Fuel Cladding Temperature (F & B)

5. Summary and Conclusions

The SDS bleed capacity is determined by numerical simulation of TLOFW event without operator recovery and TLOFW event with F & B by CEFLASH computer code. The analytical bleed capacities to prevent core uncovery are investigated by varying the analytical bleed area and number of operating HPSI pumps.

To verify the results of CEFLASH simulation a comparative analysis has also been performed by more sophisticated computer code. The predictions by the CEFLASH simulation of the transient two phase system behavior is found to be in good agreement with those by the RELAP5 simulation, except the RCS water inventory distribution which shows small difference after the hot leg voiding.

In conclusion, the results of analyses for TLOFW event with F & B by CEFLASH and RELAP5 have demonstrated that decay heat removal and core inventory make-up can be successfully accomplished by F & B operation for Ulchin 3&4 Nuclear Power Plants.

Nomenclature

ABB-CE Asea Brown Boveri-Combustion Engineering
HPSI High Pressure Safety Injection
INEL Idaho National Engineering Laboratory
LOCA Loss of Coolant Accident
NSSS Nuclear Steam Supply System
PSV Pressurizer Safety Valve
PWR Pressurized Water Reactor
RCS Reactor Coolant System
RCP Reactor Coolant Pump
RDT Reactor Drain Tank
RV Reactor Vessel
USNRC United States Nuclear Regulatory Commission

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