Journal of the Korean Nuclear Society
Volume 27, Number 2, April 1995

Small Break LOCA Analysis for RCP Trip Strategy for YGN 3&4 Emergency Procedure Guidelines

Jong Tae Seo and Kyoo Hwan Bae
Korea Atomic Energy Research Institute
(Received September 27, 1994)

영광3, 4호기 비상운전지침용 원자로냉각재펌프 정지전략을 위한 소형냉각재상실사고 분석

서종태 · 배규환
한국원자력연구소
(1994. 9. 27 접수)

Abstract

A continued operation of RCPs during a certain small break LOCA may increase unnecessary inventory loss from the RCS causing a severe core uncovery which might lead to a fuel failure. After TMI-2 accident, the CEOG developed RCP trip strategy called "Trip-Two/Leave-Two" (T2/L2) in response to NRC requests and incorporated it in the generic EPG for CE plants. The T2/L2 RCP trip strategy consists of tripping the first two RCPs on low RCS pressure and then tripping the remaining two RCPs if a LOCA has occurred. This analysis determines the RCP trip setpoint and demonstrates the safe operational aspects of RCP trip strategy during a small break LOCA for YGN 3&4. The trip setpoint of the first two RCPs for YGN 3&4 is calculated to be 1775 psia in pressurizer pressure based on the limiting small break LOCA with 0.15 ft^2 break size in the hot leg. The analysis results show that YGN 3&4 can maintain the core coolability even if the operator fails to trip the second two RCPs or trips at worst time. Also, the YGN 3&4 RCP trip strategy demonstrates that both the 10 CFR 50.46 requirements on PCT and the ANSI standards 58.8 requirements on operator action time can be satisfied with enough margin. Therefore, it is concluded that the T2/L2 RCP trip strategy with a trip setpoint of 1775 psia for YGN 3&4 can provide improved operator guidance for the RCP operation during accidents.

요 약

소형냉각재상실사고시 원자로냉각재펌프(RCP)의 지속적인 운전은 원자로냉각재의 불필요한 누출을 초래하여 심각한 노심노출 및 이에따른 핵연료 손상을 야기시킬 수 있다. TMI 사고 후 미국 NRC의 요구에 따라 CE형 발전소 사용자 단체에서는 "T2/L2"라는 RCP 트립전략을 개발하여 CE형 발전소에 적용가능토록 일반비상운전지침서에 반영하였다. 상기 T2/L2 RCP 트립전략은 사고후 원자로냉각재계통의 압력이 감소하여 RCP 트립설정치에 도달하면 처음 두대의 RCP를 우선 정지시키고, 사고가

LOCA임이 확인되면 나머지 두대의 RCP를 정지시키는 방식을 채택하고 있다. 본 논문에서는 영광3, 4호기의 RCP 트립설정치를 분석, 선정하고 T2 / L2 전략의 안전운전양상을 입증하였다. 분석결과, 최악의 파단크기로 밝혀진 0.15 ft^2의 고온관 파단 LOCA시 영광3, 4호기 RCP 트립설정치는 가압기 압력 1775 psia로 나타났으며, 운전원이 마지막 두대의 RCP를 트립시키지 못하였을 경우 혹은 최악의 시점에서 정지시켰을 경우에도 영광3, 4호기의 노심냉각능력은 확보될 수 있음이 확인되었다. 또한 영광3, 4호기의 RCP 트립전략은 미국 NRC가 요구하는 최대 핵연료피복재온도 관점에서의 10 CFR 50.46 요구조건과 운전원 조치시간 관점에서의 ANSI 58.8 요구조건도 충분히 만족함이 판명되었다. 따라서, 1775 psia의 RCP 트립설정치를 사용한 영광3, 4호기의 T2 / L2 RCP 트립전략은 사고시 운전원에게 향상된 운전지침을 제공할 수 있을 것으로 판단된다.

1. Introduction

The post accident operational status of the Reactor Coolant Pumps (RCPs) has great impact on the accident mitigation as well as operator's recovery actions. A continued operation of RCPs during a certain small break Loss of Coolant Accident (LOCA), in particular, may increase unnecessary inventory loss from the Reactor Coolant System (RCS) causing a severe core uncovery which might lead to a fuel failure. However, the incentives for operating the RCPs during non-LOCA depressurization events are to maintain the forced circulation core decay heat removal capability through the steam generators and to maintain the availability of the main spray flow to the pressurizer for a better RCS pressure control. In addition, the RCP operation provides better plant control by minimizing void formation in the reactor vessel upper head region due to the forced coolant flow through this region. RCP operation also provides better mixing in the reactor vessel downcomer and lower plenum regions minimizing pressurized thermal shock concerns.

After the TMI-2 accident, the importance of RCP operating strategy during the plant transients caused the U.S. NRC to issue several regulatory requirements[1, 2]. In response to NRC requests, the CEOG (Combustion Engineering Owners Group) developed an RCP trip strategy called "Trip-Two/Leave-Two" (T2/L2) and incorporated it in the generic Emergency Procedure Guideline (EPG) for CE plants[3, 4]. The T2/L2 RCP trip strategy consists of tripping the first two RCPs on low RCS pressure and then tripping the remaining two RCPs if a LOCA has occurred. This RCP trip strategy as well as its trip setpoint, 1300 psia for the generic EPG, are based on the analysis performed for the 2700 MWt class generic CE plant, which is a pre-System 80 design.

However, the Yonggwang nuclear power plant units 3 and 4 (YGN 3&4) is a 2825 MWt System 80 design[5] and is quite different from the plant on which the generic T2/L2 RCP trip strategy is based. YGN 3&4 has a higher safety injection tank pressure and a larger safety injection system capacity per rated power compared to the generic plant. Also, YGN 3&4 is designed with a relatively higher RCS inventory versus rated power ratio. Therefore, in order to apply the generic T2/L2 RCP trip strategy into the plant specific EPG for YGN 3&4[6], plant specific analyses need to be performed not only for evaluating its applicability but also for the plant specific setpoints.

Therefore, the purposes of this analysis are to determine the RCP trip setpoint and to demonstrate the safe operational aspects of the RCP trip strategy during a small break LOCA for YGN 3&4. The RCP trip strategy with its plant specific trip setpoints determined by this analysis provides a basis for the YGN 3&4 EPG development. In this analysis, the realistic small break LOCA evaluation model, CEFLASH-4AS/REM and PARCH/REM[7], are used. The CEFLASH-4AS/REM code is used to calculate the thermal hydraulic responses and the PARCH/REM code is used to calculate the core hot rod fuel temperature.

2. Requirements on RCP Trip Strategy

After the accident at TMI-2, the U.S. NRC recommended through IE Bulletin 79-06C[1] that all operating RCPs be tripped immediately upon reactor trip and the initiation of High Pressure Safety Injection (HPSI) caused by a low pressurizer pressure signal. However, actual events in operating plants since that time have demonstrated that the plant control can be maintained more easily if RCP operation is continued during events which reduce primary system pressure but which do not cause a loss of primary coolant inventory sufficient enough to threaten core cooling[8]. Therefore, in response to this additional information, the NRC issued a generic letter to the utilities requesting to develop a strategy for RCP operation during transients and accidents[9]. The NRC letter contains guidance and criteria for use in this development effort in order to resolve TMI Action Plan Item II.K.3.5[2].

The NRC letter also included two criteria for justification of manual RCP trip. A small break LOCA analysis based on licensing assumptions was needed to demonstrate that the RCP trip strategy meets the Peak Cladding Temperature (PCT) requirements of 10 CFR 50.46. Lastly, a most probable best estimate small break LOCA analysis was needed to determine that the time available for operator action to manually trip the RCPs satisfies ANSI Standard 58.8 recommendations.

3. T2/L2 RCP Trip Strategy

The RCP trip strategy was developed by CEOG to provide the operator with the maximum flexibility in restoring control to the plant following the transient or accident. Following a reactor trip, the T2/L2 RCP trip strategy results in the manual trip of two RCPs in opposite loops during all depressurization events since it can be demonstrated that the plant can be maintained in a safe condition with only two RCPs operating regardless of event diagnosis. The T2/L2 RCP trip strategy calls for tripping the second two RCPs if plant conditions pertaining to a LOCA are ascertained. Thus, all RCPs are tripped for a LOCA since the consequence of a LOCA are reduced without RCPs operating.

For non-LOCA events, at least two RCPs in diametrically opposed coolant loops may remain operating. Two RCPs are sufficient to enable the operator to provide a safe and controlled cooldown to the shutdown cooling entry conditions. Also, the two operating RCPs will reduce the lag between changes in the upper head temperature and changes in the hot leg temperature minimizing the formation of steam void in this region for events such as Steam Generator Tube Rupture (SGTR) and Steam Line Break (SLB). A schematic representation of the T2/L2 RCP trip strategy is presented in Figure 1.

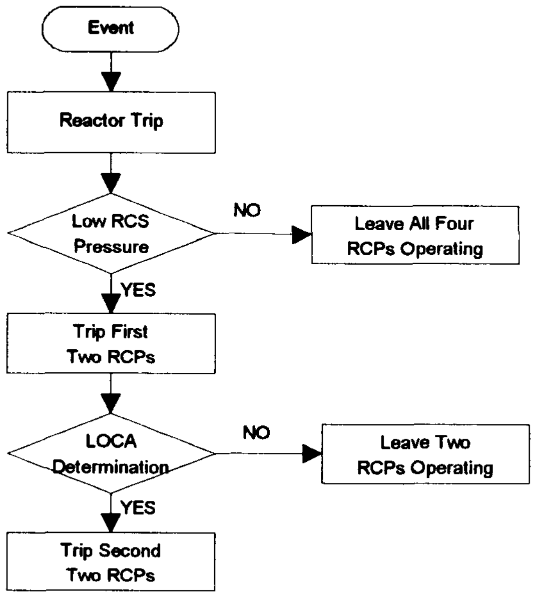

Fig. 1. Schematic Representation of the T2/L2 RCP Trip Strategy

4. Analysis for YGN 3&4 RCP Trip Strategy

The purposes of this analysis are to determine the RCP trip setpoint for YGN 3&4 based on the T2/L2 RCP trip strategy and to demonstrate the safe operational aspects of the T2/L2 RCP trip strategy during a small break LOCA. In this analysis, two computer codes with realistic evaluation model are used : CEFLASH-4AS/REM and PARCH/REM codes[7]. The RCS thermal-hydraulic responses are calculated using the CEFLASH-4AS/REM code and the fuel rod temperatures are calculated using the PARCH/REM code. The computer codes and analysis methodology with assumptions used in this analysis are detailed in the following subsections.

4.1. Computer Codes

The CEFLASH-4AS/REM code is a multi-node, multi-flowpath code with which the Nuclear Steam Supply System(NSSS) is described as a series of volume nodes connected by flowpaths. The thermal-hydraulic model is a five-equation thermal non-equilibrium model (two mass, two energy, and one momentum equation). Each loop and every major component within each loop is modeled separately. The conservation equations of mass, energy, and momentum are integrated simultaneously at each time step using an implicit integration technique.

The version F4R.1.5 of the CEFLASH-4AS/REM code was modified to analyze the small break LOCA with continued RCP operation. Since the code calculates the inner vessel subcooled water level based on the enthalpy and flowrate from the downcomer region, the inner vessel subcooled water level is erroneouly calculated resulting in a higher subcooled water level and an early core recovery when the downcomer region dries out. This phenomena occurs only for a small break LOCA with continued RCP operation because the developed RCP head drives the downcomer water into the inner vessel region while the case with tripped RCPs usually results in a subcooled water existing in a portion of lower plenum and even in the lower core regions as the cold Emergency Core Cooling System(ECCS) water injection continues. Therefore, in this analysis, the code has been modified by setting the inner vessel subcooled water level to zero when the downcomer region dries out in order to conservatively calculate the inner vessel subcooled water level and, hence, a slow core recovery. The CEFLASH-4AS/REM nodalization scheme used in this analysis is shown in Figure 2 and explained in Tables 1 and 2. As shown in this figure, the nodalization scheme uses 27 nodes and 51 flow paths to model the NSSS and related systems for the small break LOCA analysis.

The PARCH/REM model evaluates the hot rod peak cladding temperatures and local oxidation percentages using boundary conditions supplied by CEFLASH-4AS/REM. PARCH/REM heat transfer correlations cover the same forced convection and pool boiling regimes as CEFLASH-4AS/REM. The code solves the one-dimensional radial conduction equation for the fuel pellet, gap, and cladding at different axial positions along the fuel rod. The fuel rod model includes variable gap conductance and cladding swelling and rupture.

4.2. Methodology

In order to accomplish the purpose of this analysis, a series of computer code analyses and a hand calculation are performed as follows :

- Determination of worst break size,
- Determination of RCP trip setpoint,
- Demonstration of safe operational aspects of RCP trip strategy, and
- Justification of manual RCP trip.

4.2.1. Determination of Worst Break Size

As a first step for the RCP trip strategy evaluation, the worst break size is determined via a break size spectrum analysis using a Conservative Best Estimate (CBE) small break LOCA analysis method under the

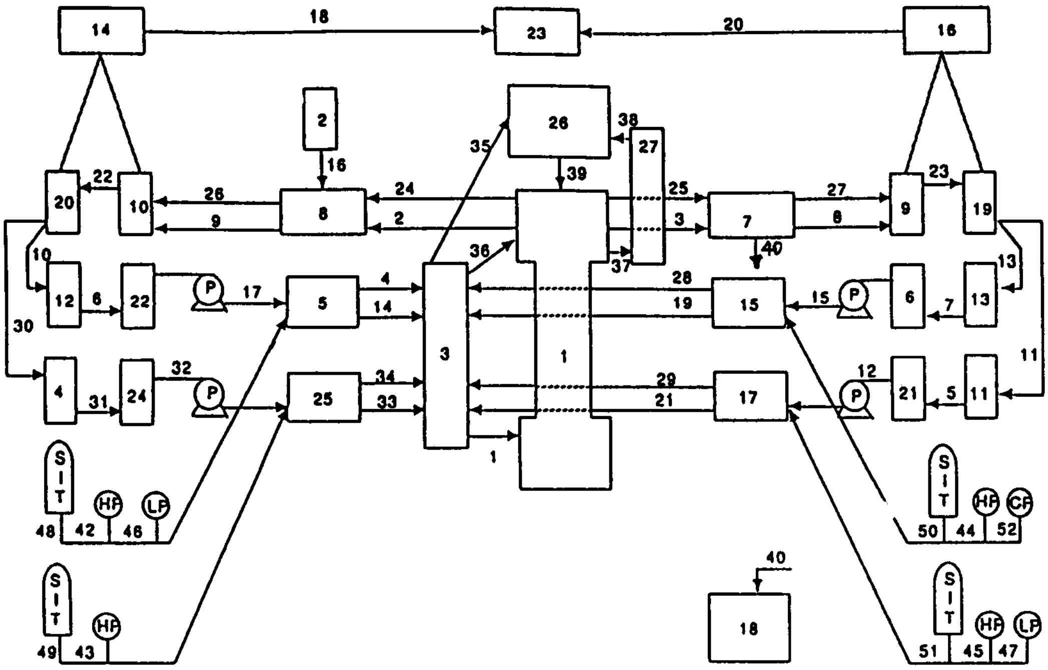

Fig. 2. CEFLASH-4AS/REM Nodalization Scheme for Hot Leg Break LOCA

Table 1. Description of CEFLASH-4AS/REM Nodes

Nodes	Description
1	Reactor core, lower and upper plenum
2	Pressurizer and surge line
3	Reactor vessel downcomer region
4, 11, 12, 13	3/5 loop seal (SG side) including 1/2 of SG outlet inactive U-tubes and outlet plenum
5, 15, 17, 25	Discharge leg including part of RCP and reactor vessel inlet nozzle
6, 21, 22, 24	2/5 loop seal (RCP side) including part of RCP
7, 8	Hot leg including reactor vessel outlet nozzle, SG inlet plenum, and inlet inactive U-tubes
9, 10	1/2 of active U-tubes
14, 16	SG secondary side
18	Containment
19, 20	1/2 of SG active U-tubes
23	Atmosphere
26	Reactor vessel upper head region
27	Reactor vessel CEA shroud region

Table 2. Description of CEFLASH-4AS/REM Flow Path

Flowpaths	Description
1	Downcomer to midpoint of active core
2, 3	Lower path connection from midpoint of active core to midpoint of hot leg
4, 28, 29, 34	Upper path connection from midpoint of discharge leg to downcomer
5, 6, 7, 31	1/2 suction leg
8, 9	Lower path connection from midpoint of hot leg to 1/4 of SG active U-tubes
10, 11, 13, 30	1/4 of SG active U-tubes to suction leg
12, 15, 17, 32	Suction leg through RCP to midpoint of discharge leg
14, 19, 21, 33	Lower path connection from midpoint of discharge leg to downcomer
16	Pressurizer surge line
18, 20	SG secondary side relief valves
22, 23	1/2 of SG active U-tubes
24, 25	Upper path connection from midpoint of active core to midpoint of hot leg
26, 27	Upper path connection from midpoint of hot leg to 1/4 of SG active U-tubes
35	Reactor vessel alignment key leakage
36	Reactor vessel outlet nozzle gap

situation of RCPs in operation. Following a small break LOCA, the RCS depressurizes rapidly until it reaches saturation condition. Shortly after reaching saturation condition, the RCS pressure stabilizes at a pressure plateau sufficiently above the secondary side pressure to remove the decay heat from the RCS. The RCS pressure will stabilizes at the plateau pressure for a period of time until the break uncovers and high volumetric steam flow is commenced.

The worst case small break LOCA of 0.5 ft^2 break in the RCP discharge leg documented in YGN 3&4 FSAR [5] has been conservatively determined based on the licensing evaluation model with the assumption of all RCPs tripped on the loss of offsite power at the time of reactor trip. Based on the results presented in Reference 8, with a continued operation of RCPs, the core mixture level degradation condition is worse if the break is in the hot leg. The hot leg breaks were more severe than cold leg breaks because RCPs forces water into the hot leg and keeps the break covered for longer period of time. Thus, more water is lost through the break and the pressure plateau period is considerably longer for the hot leg breaks. The hot leg break size which yields the smallest core inventory and deepest and longest duration of core uncovery with RCP operation and, hence, the highest peak cladding temperature will be selected as the limiting case and will be used for the rest of analyses.

The main aspects of the CBE analysis assumptions are

- homogeneous equilibrium break flow model,
- 1979 ANS decay heat curve with 1.0 multiplication factor,
- only one train of High Pressure Safety Injection (HPSI) available,
- minimum HPSI delivery curve, and
- the Main Steam Safety Valves (MSSV) for secondary heat removal.

4.2.2. Determination of RCP Trip Setpoint

During the initial transient in which the SGs are either operating normally or have not yet pressurized to the relief valve setpoint, the RCS has not yet depressurized to its plateau pressure and has much higher liquid inventory level. To prevent the severe core level degradation for the hot leg break LOCA, RCPs should be tripped in an early time into the transient before the pressure plateau is reached. Thus, the use of the RCS plateau pressure represents the lower limit for the RCP trip setpoint.

Although the RCS plateau pressure can be determined based on the computer code simulations, it is desirable to use hand calculations with conservative assumptions for maximizing the plateau pressure. Therefore, this analysis calculates the RCP trip setpoint for YGN 3&4 plants using the following energy balance equation for the RCS when the RCS is maintained at the plateau pressure :

$$Q_{CORE} + Q_{RCP} - \frac{UA\,(T_{pri} - T_{sec})}{3600} = Q_{LEAK} + Q_{ECCS} \quad (1)$$

where, Q_{CORE} = core decay heat,
Q_{RCP} = RCP heat input,
Q_{LEAK} = energy release through break flow, and
Q_{ECCS} = energy addition due to the ECCS water.

The third term in the left hand side equation represents the heat transfer through the steam generators where UA represents the overall heat transfer coefficient and T_{pri} and T_{sec} represent the primary and sencondary coolant average temperatures in the steam generator. This equation applies to an RCS condition in which the RCS have stabilized on its plateau pressure based on the SG secondary side relief capacity of the Main Steam Safety Valves (MSSVs). In this analysis, this equation has been solved using graphical method by evaluating the right and left hand side terms (Q_R and Q_L) as a function of the RCS pressure to determine the RCS pressure at which Q_R and Q_L become equal.

4.2.3. Demonstration of Safe Operational Aspects of RCP Trip Strategy

In this analysis, two small break LOCA cases with CBE method are analyzed to show the safe operational aspects of YGN 3&4 RCP trip strategy by demonstrating that the core coolability can be maintained even with the failure of tripping the second two RCPs or with the worst trip time delay for the second two RCPs. In the first case, the first two RCPs are tripped after the low RCS pressure RCP trip setpoint is reached and allowing for an additional 30 second delay for operator action. The remaining two RCPs are left operating for the rest of the transient to determine the time at which the minimum inventory on the hot side of the RCS occurred. The hot side inventory includes the liquid mass in the reactor vessel including the downcomer, the hot legs, and the riser portion of the SG. The liquid inventory in these regions represents the fluid available for core cooling during a transient. Thus, the time at which the minimum liquid inventory occurs is the worst time to trip the second two RCPs. The second case is run similar to the first case but the second two RCPs were assumed to be tripped at the time of minimum hot side liquid inventory as determined in the first case. For both cases, the hot leg break with the worst break size determined in Section 4.2.1 is simulated.

4.2.4. Justification of Manual RCP Trip

In order to justify a manual RCP trip strategy as requested by the NRC[9], the following two specific small break LOCA analyses were performed.

4.2.4.1. Compliance with 10 CFR 50.46

In accordance with the NRC guidance for the Appendix K calculation, all four RCPs are tripped 2 minutes after the low RCS pressure RCP trip setpoint is reached. An additional 30 second delay for operator action to trip the RCPs were also considered. The main assumptions used in this analysis are as follows:

- one HPSI train available,
- Henry-Fauske/Moody break flow model,
- 1971 ANS decay heat curve with 1.2 multiplication factor, and
- minimum HPSI pump flow curves.

4.2.4.2. Compliance with the Operator Action Time Requirement

This analysis is performed to justify a manual RCP trip scheme by showing that the time required for operator action complies with the standard in ANSI 58.8 which was recommended by the generic letters [9]. In this case, all four RCPs are left in operation throughout the transient to determine the time at which the operator should take manual action to prevent severe core damage. In accordance with the NRC guidance, the Most Probable Best Estimate (MPBE) assumptions are used for this case as follows:

- two HPSI train available,
- homogeneous equilibrium break flow model,
- 1979 ANS decay heat curve, and
- nominal HPSI pump flow curves.

5. Results and Discussions

5.1. Worst Break Size

Table 3 summarizes the results of main parameters obtained from the small break LOCA cases with the hot leg break sizes of 0.5, 0.3, 0.2, 0.15, 0.1, 0.08, 0.05, and 0.02 ft^2. The results of this break size spectrum analysis shown in this table indicate that a continued RCP operation could produce partial core uncovery for break sizes greater than or equal to 0.1 ft^2. For those breaks smaller than 0.1 ft^2, core uncovery does not occur due to the enough HPSI flow which matches the relatively low break flow. For the larger breaks, the vessel mixture level falls below the bottom of hot leg before the Safety Injection Tank (SIT) actuation due to a relatively larger break flow compared to the HPSI flow. As the break size decreases, the depressurization rate becomes slow and the SIT actuation is delayed causing more mass depletion and deeper core uncovery.

Among these breaks, the 0.15 ft^2 hot leg break shows the minimum hot side inventory, the deepest and longest duration of core uncovery, and the highest PCT. Therefore, the 0.15 ft^2 hot leg break LOCA case is determined to be the worst case and is used in the subsequent analyses. However, it should be noted that the PCT obtained with the PARCH/REM code is as low as 733 °F which is well within the PCT limit of 2200 °F as defined in 10 CFR 50.46.

5.2. RCP Trip Setpoint

Equation (1) has been evaluated to find the plateau pressure during the worst case hot leg break LOCA. In this calculation, the left hand side (Q_L) and the right hand side (Q_R) terms are evaluated and plotted in Figure 3 as a function of RCS pressure. The major assumptions used are as follows :

- 4.8% of 102% core power for the decay heat,
- RCP heat input of 20 MWt,
- SG overall heat transfer coefficient of 300 Btu/hr-ft^2 · °F,
- SG heat transfer area of only one SG,
- SG pressure of 1330.7 psia (the highest MSSV set pressure), and
- RCS makeup by one HPSI train with minimum flow curve and one charging pump.

The decay heat level and the SG overall heat transfer coefficient are derived from the CEFLASH-4AS/REM analysis results during the pressure plateau period with some conservatism.

As shown in Figure 3, the left hand side term decreases as the RCS pressure increases mainly due to the SG heat transfer enhancement caused by the increase in the primary temperature which is assumed to be the saturation temperature corresponding to the RCS pressure. The right hand side term also dec reases as the RCS pressure increases due to the reduced ECCS flow. However, the decrease rate of the right hand side term is much slower than that for the left hand side term and, thus, the two curves intersect at around the RCS pressure of 1505 psia which represents a conservative RCS plateau pressure for YGN 3&4 as discussed in Section 4.2.2.

Based on the above results, the nominal setpoint for tripping the first two RCPs becomes 1505 psia for YGN 3&4 plants. The actual RCS pressure which will be used for the setpoint to trip the first two RCPs should include an allowance for the instrument un-

Table 3. Summary of Size Spectrum Analysis Results

Break Size (ft^2)	Hot Side Minimum Inventory		Inner Vessel Minimum Two Phase Mixture Level		Core Uncovery Duration	SIT Injection Time	Peak Cladding Temp.
	$\times 10^3$ lbm	Time (sec)	ft	Time (sec)	(sec)	(sec)	(°F)
0.5	23,398	142	18.7	162	4.6	142	540
0.3	18,432	226	14.6	261	19.0	226	546
0.2	16,624	341	13.8	397	33.4	341	665
0.15	14,482	482	13.3	563	63.6	482	733
0.1	16,155	800	20.1	660	3.1	843	572
0.08	18,895	960	22.8	820	0.0	1254	572
0.05	24,100	1400	23.7	1234	0.0	–	572
0.02	28,753	3250	23.7	3479	0.0	–	572

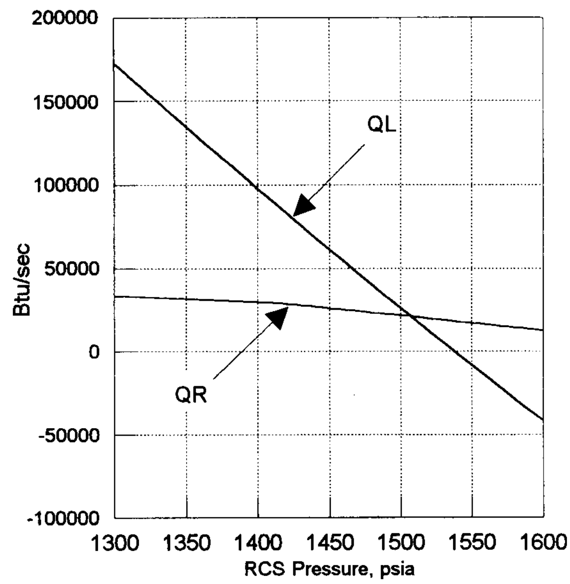

Fig. 3. Graphical Determination of the Plateau Pressure

certainty. Considering the instrument channel error under harsh environment for the pressurizer pressure gauge of 270 psi for YGN 3&4, the actual trip setpoint becomes 1775 psia in pressurizer pressure. This setpoint for tripping the first two RCPs is used for the subsequent analyses for the demonstration of safe operational aspects as well as for the justification of compliance with the NRC requirements.

5.3. Safe Operational Aspects of the RCP Trip Scheme

The two CBE small break LOCA cases analyzed for the demonstration of safe operational aspects of RCP trip scheme are summarized in Table 4 and major parameters are plotted in Figures 4 through 6. As shown in Figure 4, the hot side liquid mass for both cases decreases rapidly until it reaches its minimum value of about 22,182 lbm at 480 seconds. Beyond this time, the hot side inventory starts to increase due to the injection of the SIT water as the pressurizer pressure reaches 585 psia (see Figure 6). On the other hand, as shown in Figure 5, the two-phase mixture level in the inner reactor vessel decreases slowly but remained above the top of the reactor core until the minimum inventory is reached. However, it decreases quickly as the cold SIT water collapses the two-phase frothy mixture maintained by the continued operation of the two RCPs.

As shown in Table 4 and Figures 4 through 6, tripping the second two RCPs at 480 seconds has very little effect on the hot side liquid inventory and pressurizer pressure because the core region contains a large amount of trapped bubbles. Therefore, the termination of the second two RCPs only prevented a further mass release during the period of SIT injection showing a slightly higher mass inventory. However, it caused an early collapse of the two-phase mixture level compared to the two RCPs running case as shown in Figure 5. This early collapse in two-phase mixture level as well as subsequent cold SIT water injection resulted in an increase in the duration of core uncovery (89 versus 79 seconds) as well as a deeper level collapse (7.7 versus 6.4 ft). The peak cladding temperatures obtained for both cases increases to only 805 and 859 °F, respectively, dem-

Table 4. Summary of CBE Analysis Results

Break Size (ft^2)	Hot Side Minimum Inventory		Inner Vessel Minimum Two Phase Mixture Level		Core Uncovery Duration	SIT Injection Time	Peak Cladding Temp.
	$\times 10^3$ lbm	Time (sec)	ft	Time (sec)	(sec)	(sec)	(°F)
0.15	22,182	480	13.7	547	78.7	481	805
0.15	22,182	480	12.4	538	89.2	481	859

onstrating that the YGN 3&4 RCP trip strategy provides a safe operational aspects even if the operator fails to trip the second two RCPs or trips at worst time.

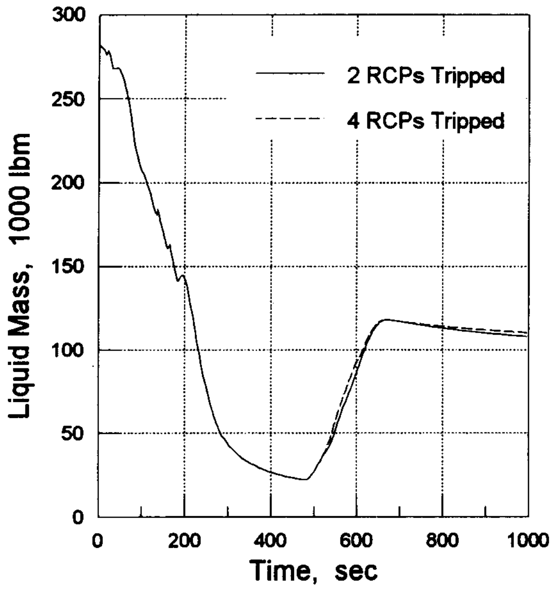

Fig. 4. Hot Side Liquid Mass for CBE SBLOCA ($0.15\ ft^2$)

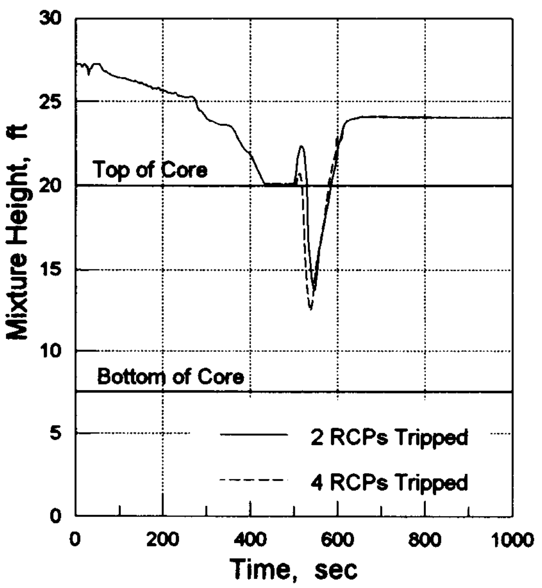

Fig. 5. Inner Reactor Vessel Mixture Level for CBE SBLOCA ($0.15\ ft^2$)

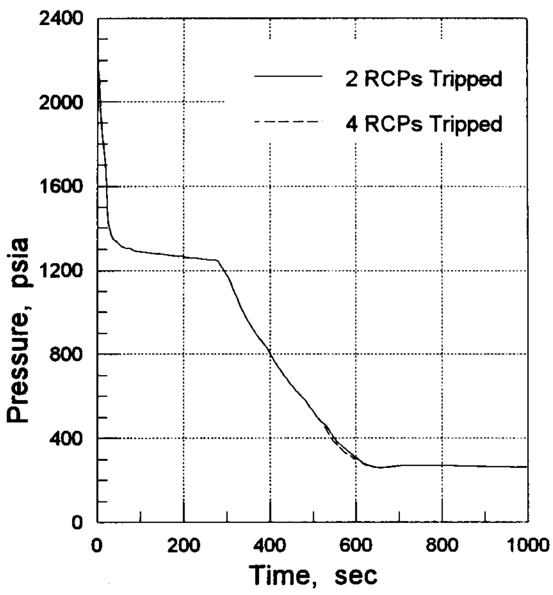

Fig. 6. Pressurizer Pressure for CBE SBLOCA ($0.15\ ft^2$)

5.4. Justification of Manual RCP Trip

The worst hot leg small break LOCA case is simulated in accordance with the NRC guidance for the Appendix K and the results are delineated in Figures 7 through 9. As shown in Figure 7, the pressurizer pressure decreased to the first two RCP trip setpoint of 1775 psia at about 15 seconds, and all four RCPs were tripped at 150 seconds later according to the 2 minute trip delay requirements and additional 30 seconds delay. After reaching the pressure plateau, the pressurizer pressure continues to decrease and reaches 585 psia at 462 seconds at which the SIT starts to inject the cold water.

As shown in Figures 8 and 9, the mixture level for the inner reactor vessel and the hot side liquid inventory continue to decrease up to the point where the SIT starts injection. During this transient, the mixture level decreased slightly below the top of core resulting in a core uncovery time of 6 seconds. The minimum liquid inventory obtained at 460 seconds is about 71,439 lbm which is more than three times larger than that obtained with continued RCP operation (see Table 4). Therefore, the steam bubbles trapped in the two-phase mixture at the time of minimum inventory was negligible and, thus, the mixture level rec-

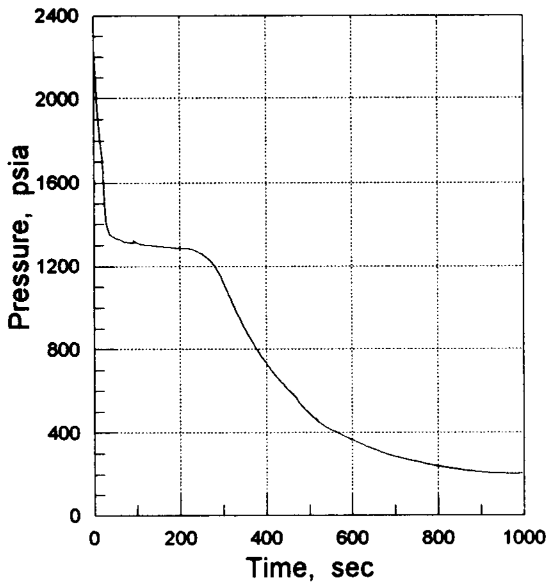

Fig. 7. Pressurizer Pressure for Licensing SBLOCA (0.15 ft²)

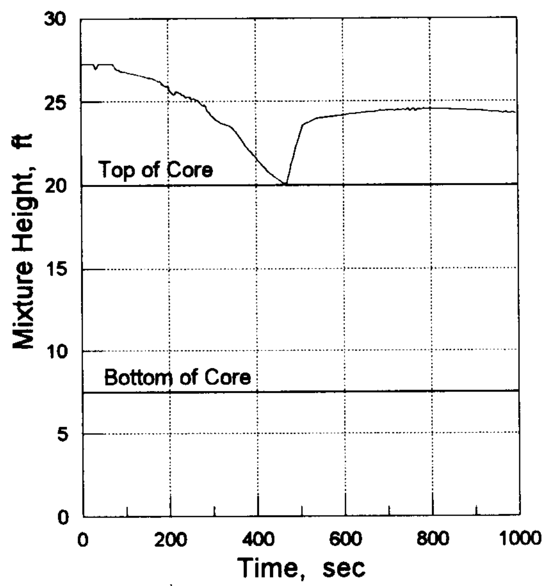

Fig. 8. Inner Reactor Vessel Mixture Level for Licensing SBLOCA (0.15 ft²)

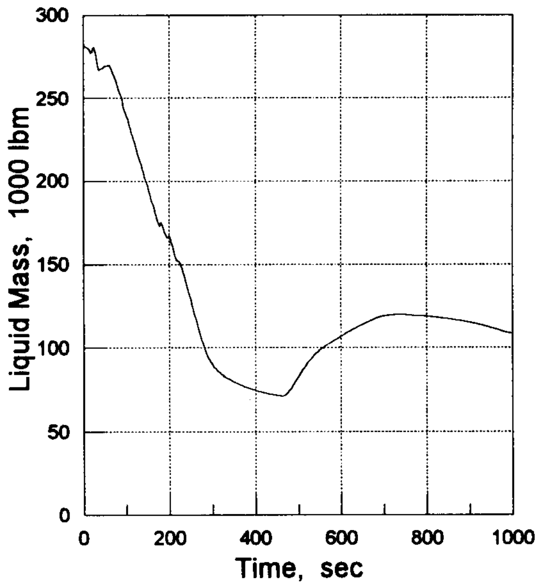

Fig. 9. Hot Side Liquid Mass for Licensing SBLOCA (0.15 ft²)

overs as soon as the SIT starts injection without showing any level collapse.

Since the core was covered throughout the transient, the peak cladding temperature would have been well below the limit of 2200 °F set forth in 10 CFR 50.46 and no PARCH/REM analysis was performed. Therefore, the manual RCP trip setpoint determined for YGN 3&4 has been justified to be acceptable in accordance with the NRC requirements.

Another case to justify the operator action time requirement as specified in ANSI standards 58.8 was run according to the NRC recommendations and the results are plotted in Figures 10 and 11. Since the ANSI standards 58.8 recommends approximately 15 minutes to be allowed for the operator to manually take recovery action, which is a turning off the RCPs for this analysis, the limiting case with most probable best estimate assumptions was run with all four RCPs on. As shown in Figure 10, the core uncovers, but only for a short period of time, caused by the collapse of two-phase frothy mixture due to the SIT injection. Also, this short period of core uncovery occurred prior to the minimum operator action time of 15 minutes. This results demonstrate that, for YGN 3&4, the RCPs do not have to be tripped prior to 15 minutes after the RCP trip setpoints were reached. If the RCPs had been tripped at the time of minimum inventory (see Figure 11), the core would uncover slightly deeper as shown in Figure 5. Under such conditions, calculated clad temperature at the fuel hot spot would be well under licensing limits. In conclusion, the results of the most probable best estimate analysis demonstrate the minimum time required for the operator to trip the RCPs is infinite. Therefore, the manual RCP trip strategy satisfies the ANSI standard time response criteria.

6. Conclusions

This analysis determined the RCP trip setpoint for YGN 3&4 plants and demonstrated the safe operational aspects of the T2/L2 RCP trip strategy during a small break LOCA in order to provide the bases for the YGN 3&4 EPGs. The analysis results showed that the 0.15 ft^2 hot leg break LOCA case is the limiting case under the condition with all RCPs in operation. Also, the trip setpoint for the first two RCPs is determined to be 1775 psia in the pressurizer pressure based on the plateau pressure for the 0.15 ft^2 small break LOCA.

A confirmatory analysis performed to demonstrate the safe operational aspects of the YGN 3&4 RCP trip strategy showed that the YGN 3&4 plants can maintain the core coolability in case the operator fails to trip the second two RCPs or trips at worst time. Finally, the analysis performed to justify the manual RCP trip strategy based on the U.S. NRC's recommendation resulted that both the 10 CFR 50.46 requirement on the peak cladding temperature and the ANSI standards 58.8 requirements on the operator action time can be satisfied with great margin. Therefore, it is concluded that the T2/L2 RCP trip strategy with the 1775 psia trip setpoint for YGN 3&4 plants can provide improved operator guidance for the RCP operation during accidents.

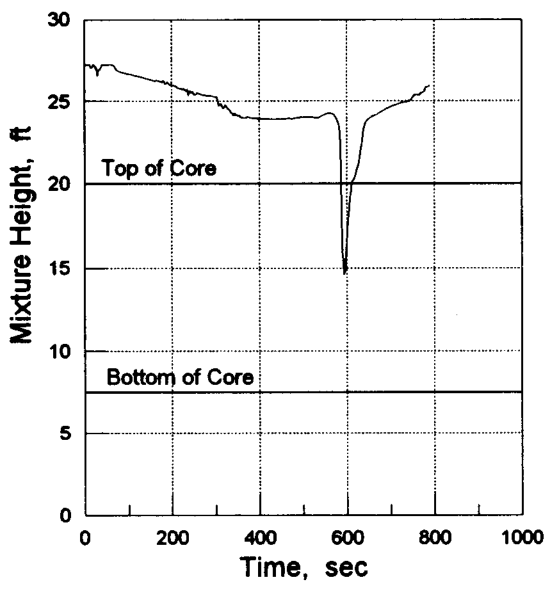

Fig. 10. Inner Reactor Vessel Mixture Level for MPBE SBLOCA (0.15 ft^2)

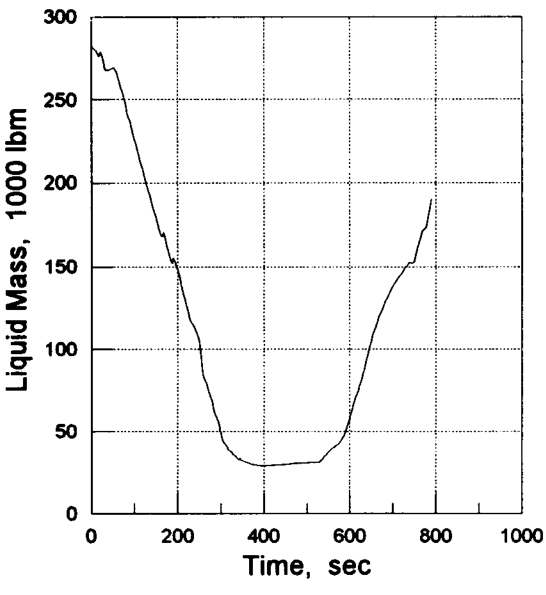

Fig. 11. Hot Side Liquid Mass for MPBE SBLOCA (0.15 ft^2)

References

1. Nuclear Incident at Three Mile Island, U.S. NRC IE Bulletin 79-06C (1979).
2. NUREG-0933, Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," (1983).
3. Justification of Trip two/Leave two Reactor Coolant Pump Trip Strategy during Transients, CEN-268, Combustion Engineering, Inc. (1984).
4. Combustion Engineering Emergency Procedure Guidelines, CEN-152, Rev. 03, Combustion Engineering, Inc. (1987).
5. Final Safety Analysis Report for YGN 3&4. Korea Electric Power Corporation (1993).
6. Emergency Procedure Guidelines for YGN 3&4, Rev. 0, Korea Atomic Energy Research Institute (1994).
7. Realistic Small Break LOCA Evaluation Model, CEN-373-P, Combustion Engineering, Inc. (1988).
8. Response to NRC IE Bulletin 79-06C, Items 2 and 3 for CE NSSS, CEN-115-P, Combustion Engineering, Inc. (1979).
9. Resolution of TMI Action Item II.K.3.5, Automatic Trip of RCPs, U.S. NRC Generic Letter 83-10 (1983).