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The Effect of an Aggressive Cool-Down Following A Refueling Outage Accident in which a Pressurizer Safety Valve is Stuck Open

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Abstract

A PSV (pressurizer safety valve) popping test carried out in the early phases of a refueling outage may trigger a test-induced LOCA (loss of coolant accident) if a PSV fails to fully close and is stuck in a partially open position. According to a KSNP (Korea standard nuclear power plant) low power and shutdown PSA (probabilistic safety assessment), the failure of a high pressure safety injection (HPSI) accompanied by the failure of a PSV to fully close was identified as a dominant accident sequence with a significant impact on low power and shutdown risks (LPSR). In this study, we aim to investigate and verify a new means for mitigating this type of accident using a thermal-hydraulic analysis. In particular, we explore the applicability of an aggressive cool-down combined with operator actions. The results of the various sensitivity studies performed here will help reduce LPSR and improve Refueling outage safety.

Key Words : PSV, low power and shutdown, PSA, aggressive cool-down

1. Introduction

A Korea standard nuclear power plant (KSNP) is a 1000 MWe two-looped pressurized water reactor (PWR) designed on the basis of System 80 of combustion engineering (CE) [1]. Three pressurizer safety valves (PSV), located on the top head of the pressurizer in a KSNP, provide protection from excessive pressure for the reactor coolant system (RCS). They are totally enclosed, backpressure compensated, and spring-loaded, thereby meeting ASME (American Society of Mechanical Engineers) code requirements [2]. The integrity of these valves is verified by the so-called PSV popping test, which is carried out during the early stages of a refueling outage when the RCS pressure is still close to that of normal operation. If a PSV is not fully closed during the PSV popping test, the PSV discharges the steam of the pressurizer through the relief line that pipes into the reactor drain tank (RDT). If the steam discharge exceeds the capacity of the RDT, the tank contents are relieved to the containment vessel via a rupture disk. This accident scenario results in a test-induced loss of coolant accident (LOCA), which contributes significantly to low power and shutdown (LPSD) risks (see YGN 5&6 LPSD PSA [3]). As a dominant accident sequence leading to core damage, high pressure safety injection (HPSI) system failure following a PSV open-valve accident was identified in previous probabilistic safety assessments (PSA), but subsequent operator action and useful means for mitigating the accident sequence were not considered. The objectives of the present study are to find and verify a new means for mitigating this type of accident and, in particular, to assess the applicability of an aggressive cool-down for an accident sequence involving HPSI failure subsequent to a PSV' s failure to close in the early stages of a refueling outage.

Aggressive cool-down is an operator-initiated primary system depressurization method that uses the steam generator (S/G) as a primary side heat removal. An operator feeds auxiliary water to the S/G and relieves the steam using an atmospheric dump valve (ADV). A maximum RCS cool-down limit is usually given to prevent RCS pressurized thermal shock (PTS) [24]. As in a combustion engineering (CE) type of nuclear power plant (NPP), 55℃/hr is used as the maximum cool-down rate in the KSNPs. The emergency operation plan (EOP) of the KSNP requires that an operator perform aggressive cool-down at the maximum cool-down rate. The effect of aggressive cool-down has been widely researched experimentally and analytically over the past dozen years [Kawanishi et al., 1991; Liu et al., 1998; Liu et al., 2000; Asaka et al., 1998; Han et al., 2003]. Based on these studies, aggressive cool-down is being used as an accident management procedure to prevent core damage of KSNP in the small break LOCA following HPSI failure [11]. Aggressive cool-down accelerates primary side depressurization, and thereby shortens the time needed to reach the point of the subsequent safety injection. It also reduces the break flow and promotes swelling of the two-phase mixture level in the core [20].

The MARS code [7, 9] developed by KAERI was used as the best estimate thermal-hydraulic system code. The RELAP5/MOD3 [5] and COBRA-TF [6] codes were consolidated into the MARS code in the form of one-dimensional (1D) and three-dimensional (3D) thermal-hydraulic modules, respectively, by implicitly integrating the hydrodynamic solution scheme and by unifying various models and input/output features. The one-dimensional thermal-hydraulic module of the MARS code was used for the analysis. The applicability of the MARS code was verified by a comparison with the RELAP5/MOD3.3 verification and validation report [8, 9].

The calculations showed that an accident in which the PSV is stuck open in a KSNP has the characteristics of a small LOCA in the sense that the primary system pressure decreases slowly, but it also resembles a medium-sized LOCA in that the break flow is sufficient to uncover the core in the early stages of the accident. When a HPSI fails, the primary system pressure decreases slowly enough so that neither the safety injection tank (SIT) nor the low pressure safety injection (LPSI) is actuated. For these accident sequences, we simulated an aggressive cool-down by a S/G (steam generator). Also, a sensitivity study for several important parameters was performed to investigate their effects on the accident progression. In chapter 2 the overall simulation conditions for the accident are described, and the operator action model for an aggressive cooldown is presented in chapter 3. The simulation results and sensitivity study are depicted in chapter 4 and chapter 5, respectively.

2. Simulation Conditions

The RCS and the secondary side initial and boundary conditions were assumed to be similar to normal plant operation conditions, except in terms of reactor power as the decay heat level used is 1 hour after shutdown. Table 1 shows the important initial and boundary conditions used in this analysis. For the best estimation of the results for each accident sequence, the model and assumptions used in this analysis were chosen to reduce conservatism as much as possible.

Identification of Mitigating Systems

To analyze each accident sequence, it is necessary to find an available mitigation system in the accident progression. The following mitigation systems or functions were considered in this analysis:

- High Pressure Safety Injection (HPSI)
- Rapid Cool-down by Steam Generator (S/G)
- Safety Injection Tank (SIT)
- Low Pressure Safety Injection (LPSI)

It was assumed in this analysis that reactor core residual heat removal by recirculation is always accomplished, provided that the RCS water inventory at the shutdown cooling system (SCS) start time is sufficient to prevent core boiling. The HPSI and LPSI were modeled as automatically operated when their actuation set-pressure is reached. A signal delay time of 21.34 seconds is considered at the start of the HPSI system. Also, the SIT is modeled as operating when the SIT actuation pressure is reached. Aggressive cool-down by a S/G cannot be initiated automatically in the KSNPs. Therefore, we assume that an operator manually initiates the operation several minutes after the accident occurs. Detailed control methods and models are given in the next chapter.

Decay Heat Model

For fission products, ANS79 data with an input fraction of 1.0 is usually used for best-estimate calculations [5]. For realistic calculations, ANS79 with an input fraction of 1.0 was used in the analysis.

Critical Flow Model

A modified Henry-Fauske critical flow model was used to model the discharge flow rate at the PSV. This model is a default in both RELAP5/MOD3.3 and MARS2.1. In the simulation of Edward' s Pipe Problem and

Table 1. Initial & Boundary Conditions for the Analysis of PSV Stuck Open Accident

Parameter	Values	Remarks
Reactor Power (MWth)	39.326/2815(1.397%)	
RCS Pressure (MPa)	15.5	
Hot-Leg Temperature(K)	599.6	
Cold-Leg Temperature(K)	568.6	
S/G Pressure (MPa)	7.27	
S/G Level (m)	11.87	
HPSI Setpoint (MPa)	12.15 (1762psia)	Signal delay
SIT Actuation Setpoint (MPa)	4.11	
LPSI Actuation Setpoint (MPa)	1.58	Signal delay
MSSV Setpoint (MPa)	9.045	

Marveken Test, the Henry-Fauske model installed in RELAP5/MOD3.3 over-predicted the discharge flow rate while RELAP5/MOD3.2 under-predicted the discharge flow rate [10]. Also, preliminary calculations for an accident in which the PSV is stuck open confirmed that the overall discharge flow rate using the Henry-Fauske model was larger than that of RELAP5/MOD3.2 [26]. Since the timing of the core uncover is closely related to the discharge flow rate, it is expected that the core damage will occur faster than that of RELAP5/MOD3.2.

Determination of Discharge Flow Area

The discharge flow area of the PSV is an important factor in determining the timing of core damage. The PSV in KSNPs has the design value of a vapor discharge flow rate from 208,651 kg/hr to 285,762 kg/hr at RCS pressure, 175.133 MPa [14]. To consider the uncertainty of the evaluation, the maximum discharge rate was chosen as the discharge flow area. From the single phase vapor Henry-Fauske critical flow model [5], the mass flux can be computed as follows.

$$G_c^2 = \left[\frac{\gamma P_0}{v_o}\right]\left(\frac{2}{\gamma+1}\right)^{\frac{\gamma+1}{\gamma-1}} \quad (1)$$

where P_0, v_o, and γ represent stagnation pressure, stagnation vapor specific volume, and polytropic constant, respectively.

For a given mass flow rate, W of 285,762 kg/hr, the discharge area can be computed as

$$A = W/(C_D{*}\,G_c) \quad (2)$$

We used a discharge coefficient, C_D, as 1. From Eq (2), we obtained a discharge area of about 2.26 inches in diameter. The calculated discharge flow area is in good agreement with the data-based mechanical drawings of the PSV [2, 15].

Decay Heat Level

Based on data from refueling outages of various plants, the representative decay heat level was assumed to be 1 hour after shutdown. Since the decay heat level decreases sharply in the early stages after shutdown, it is expected that the use of an earlier decay heat level would accelerate the core damage time while the operator' s available time is reduced. To examine the effects of decay heat level on accident progression, a sensitivity analysis was also performed for this parameter in chapter 4.5.

Other Assumptions

As a core damage indicator, we used the hottest fuel rod temperature above 2200 °F, which is widely used as the core damage criteria in PSA and safety analyses. We modeled the heat structure representing the fuel rods to have a top-skewed cosine shape with a linear heat generation rate (LHGR). It is expected that, because the core is uncovered from the top, this assumption would help to estimate the core damage time more conservatively.

Nodalization

The RCS and secondary side were nodalized with 221 hydrodynamic volumes, 222 junctions, and heat structures, including the fuel assemblies, heat exchanger, and all the related structures. The hydrodynamic volumes included the reactor vessel, hot leg, cold leg, two S/G, a pressurizer, and ECCS. Figure 1 shows the overall nodalization of the RCS and secondary system. The pressurizer is nodalized with 5 volumes, on which the PSV is connected to the containment. The reactor core is nodalized with 12 volumes in which the heat structures representing the fuel assemblies are categorized into two groups, one for the average

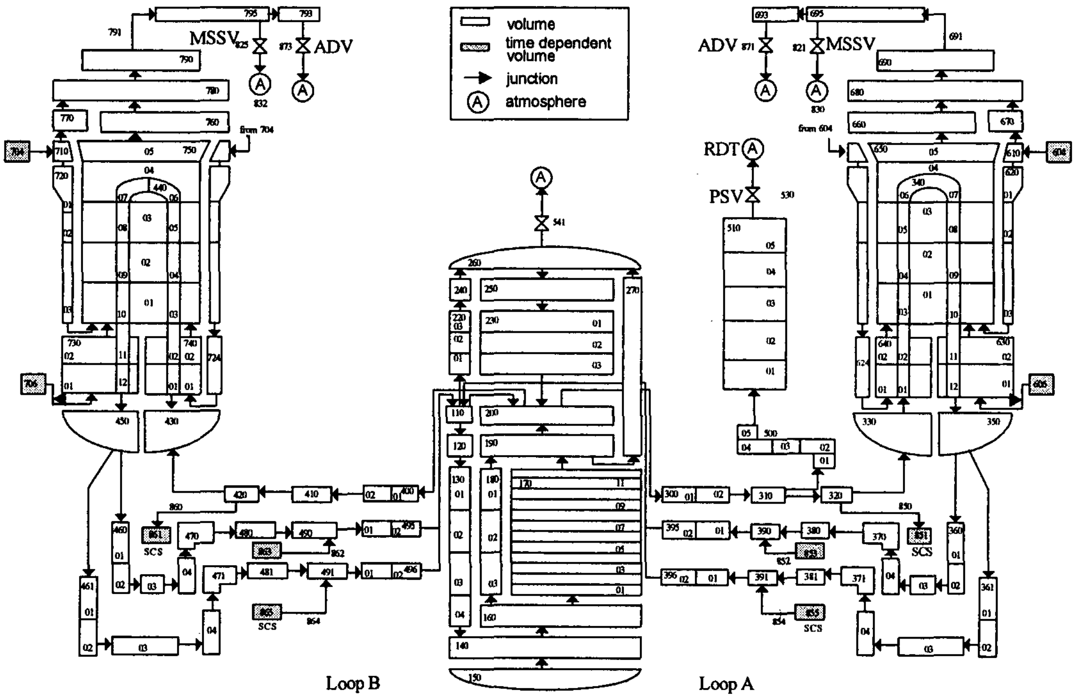


Fig. 1. RCS Nodalization

and one for the hot assembly. The HPSI, the LPSI, and the feed water are modeled by "time dependent volumes" and "time dependent junctions." The SIT is modeled by "accumulator component." The PSV is represented by "trip valves." The S/G consists of U-tubes, inlet/outlet nozzles of the primary side coolant, downcomer, evaporator, etc. The heat generated from the reactor is exchanged at the heat structures between the U-tubes and evaporator. The MSSV and ADV are connected to the steam line between the S/G and the turbine. The MSSV and ADV are modeled by a "servo valve," whose areas are controlled by control variables. The MSSV is controlled by secondary side pressure and the ADV is controlled by the aggressive cool-down model given in the next section.

3. Aggressive Cool-down Model

In the small break LOCA, when the HPSI system fails to operate, RCS depressurization is required to facilitate a subsequent low pressure safety injection system such as SIT or LPSI [11]. The RCS depressurization can be attained by either using a primary system safety depressurization system (SDS) or a steam generator (S/G). RCS depressurization by S/G is performed by transferring the primary system' s heat to a secondary system. Since the SDS is not used for a LOCA, RCS depressurization using S/G was modeled in this analysis.

The steam generated in the S/G can be removed by opening either the turbine bypass valve (TBV) or the atmospheric dump valve (ADV).

These two valves have sufficient capacity to meet the required RCS coòl-down rate [2]. Since the ADV operations are prescribed as an emergency operation procedure (EOP), we modeled the ADV as a secondary side steam discharge mechanism.

According to the plant operation manual [4], the operator manipulates the valves using the temperature difference between the averaged RCS temperature and the targeted reference temperatures:

$$\Delta T = T_{RCS,avg} - T_{ref} \tag{3}$$

The averaged RCS temperature means the average of the cold and hot leg temperature:

$$T_{RCS,avg} = \frac{1}{2}\left(\frac{1}{2}(T_{h1} + T_{c1}) + \frac{1}{2}(T_{h2} + T_{c2})\right) \tag{4}$$

When the temperature difference between the RCS and the target temperature is larger than 4℃, the valve is designed to fully open [4]. We assumed that the valve area change rate for temperature is linear between 0 and 4 ℃:

$$\frac{dA}{dT} = \frac{1}{4}\ normalized\quad area/^{o}C \tag{5}$$

The valve area change for the temperature difference, ΔT, can be obtained by the following relation:

$$\Delta A = \frac{dA}{dT}\Delta T = \frac{1}{4}\Delta T \tag{6}$$

Using Eq. (6), the new time step normalized valve area can be obtained as:

$$A^{n+1} = A^{n} + \frac{dA}{dT}\Delta T^{n} = A^{n} + \frac{1}{4}\Delta T^{n} \tag{7}$$

The plant system description manual [2] states that the Open/Close speed of the ADV per second is more than 0.05 normalized area. 0.05 normalized area per second was used as the maximum valve Open/Close speed in this study. According to the allowed maximum valve Open/Close speed limit, the maximum valve area change rate is limited in the code calculations as:

$$-0.05 \le \frac{\Delta A}{\Delta t} \le 0.05 \tag{8}$$

The RCS maximum cool-down rate is usually limited by the pressurized thermal shock (PTS) of the RCS piping and vessel [24, 25]. In a KSNP, the maximum cool-down rate is limited to within 55 ℃/hr (100 ℉/hr) [11]. When the operator opens the ADV, the RCS temperature change does not occur immediately and it is difficult for the operator to know the resulting RCS temperature change for a specific ADV area. It is assumed that the operator manipulates the valve and that no other action is taken for the valve for some time, which causes the RCS temperature to drop step-wisely [23]. To take the uncertainties of operator action into account, we conservatively assumed that the valve operator manipulates the valve once every 2 minutes. For a given cool-down rate of 55 ℃/hr and an ADV manipulation interval of 2 minutes, the cool-down process will progress as shown in Figure 2.

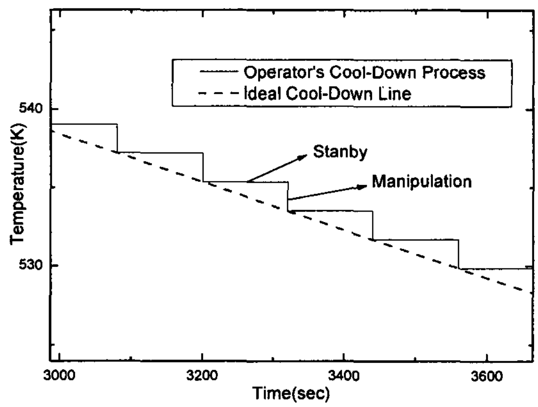


Fig. 2. Illustration of Operator Valve Control Intervals

4. Results and Discussions

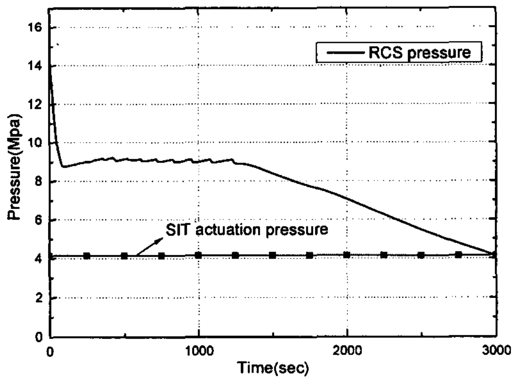
4.1. HPSI Success Sequences

The sequence of the HPSI success was simulated first. To investigate the minimum requested success criterion for the accident mitigating system, it is assumed that only one train of the HPSI system is available. As shown in Figure 3(a), after the PSV fails to close, the pressure of the RCS rapidly decreases to reach the saturation pressure. Since the HPSI signal, which is generated from the pressurizer low pressure, is set up at 12.75 MPa, the HPSI starts within 1 minute.

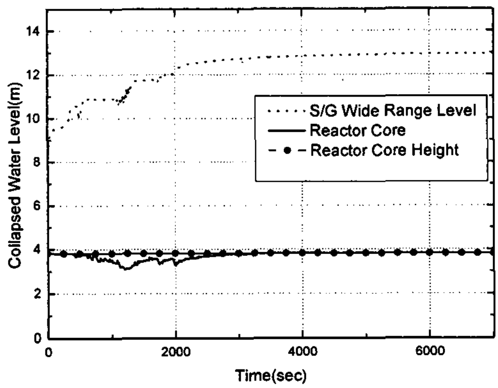
In the early stages of the accident, the break flow from the PSV, as shown in Figure 3(d), is larger than the HPSI flow, which results in a slight decrease of RCS water level, as shown in Figure 3(c). After about 2000 seconds, the HPSI flow balances the break flow, and then the RCS level recovers its full height. From these results, we concluded that there is no core damage in this accident sequence, provided that a HPSI of one train operates successfully.

4.2. HPSI Failure with no Mitigating Actions

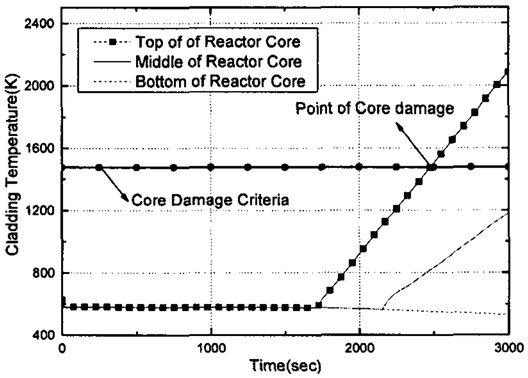
In this case, we assumed that the HPSI failed to



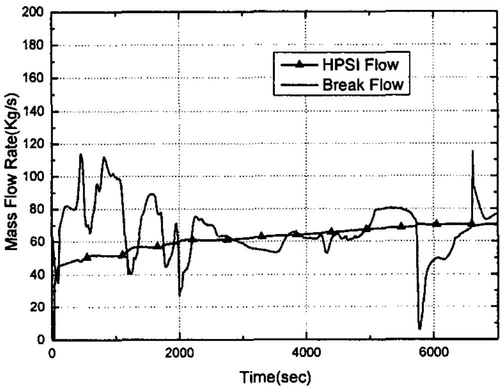
(a) Pressure



(c) Collapsed Water Level



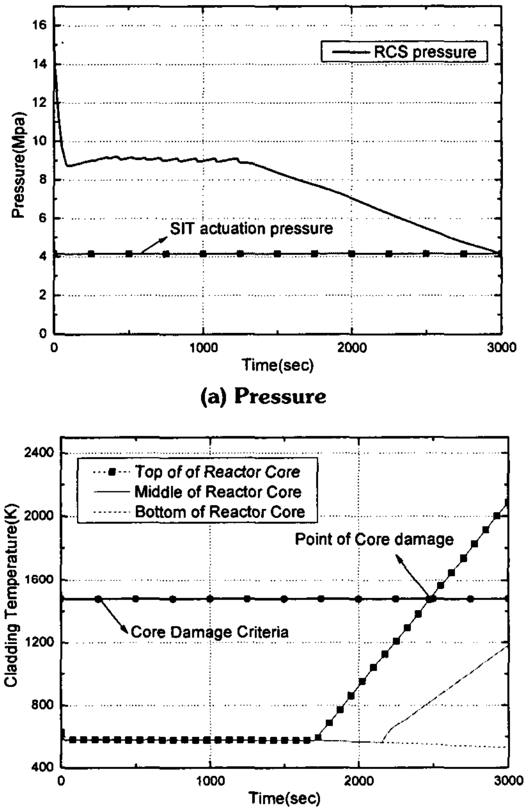
(b) Cladding Temperature



(d) Flow Rate

Fig. 3. RCS Thermal-hydraulic Behavior for the Sequence of a HPSI Success

operate and that no subsequent mitigating action was taken by an operator. As in the previous simulation, after the PSV fails to close, the RCS pressure rapidly decreases to saturation pressure, as shown in Figure 4(a). Since there is no mitigating action following the HPSI failure, the RCS maintains a constant pressure due to flashing of depressurized water for a considerable time, which prevents a subsequent safety injection such as a SIT from being actuated. As the RCS inventory is depleted, the reactor core starts to heat up, as shown in Figure 4(b).



(a) Pressure

(b) Cladding Temperature

Fig. 4. RCS Thermal-hydraulic Behavior for the Sequence of a HPSI Failure Without Mitigating Actions

4.3. S/G Steam Dump

As shown in the previous cases, if no mitigating actions follow a HPSI failure, the HPSI failure directly causes core damage. When a HPSI failure occurs, to operate a subsequent safety injection such as SIT or LPSI, appropriate RCS depressurization means are required to mitigate the accident.

A S/G removes the heat generated from the reactor during normal operation. If a secondary system is available in the accident condition, the depressurization of RCS can be accomplished via this system. To investigate the feasibility and the heat removal capability of a S/G, a steam dump through the secondary side valve was simulated. It is assumed in the analysis that one main steam safety valve of each of the two steam generators was fully open and that auxiliary feed water was continuously supplied to the steam generator. Figure 5 shows the results of this simulation. By opening the MSSV, the pressure of the secondary side is rapidly decreased following primary system' s pressure to decrease. It is shown from Figure 5(a) that the SIT injection actuation pressure is reached at about 3 minutes after the steam dump, and the LPSI injection pressure is reached 1 minute later. From these results, it was confirmed that the RCS cooldown by S/G can be an effective method for the early actuation of a safety injection such as SIT or LPSI.

4.4. RCS Cool-down by 55 ℃/hr Cool-down Line

The steam dump operation is the most effective method to depressurize the RCS, as the SIT and LPSI systems can be actuated within a few minutes of the accident. However, such an

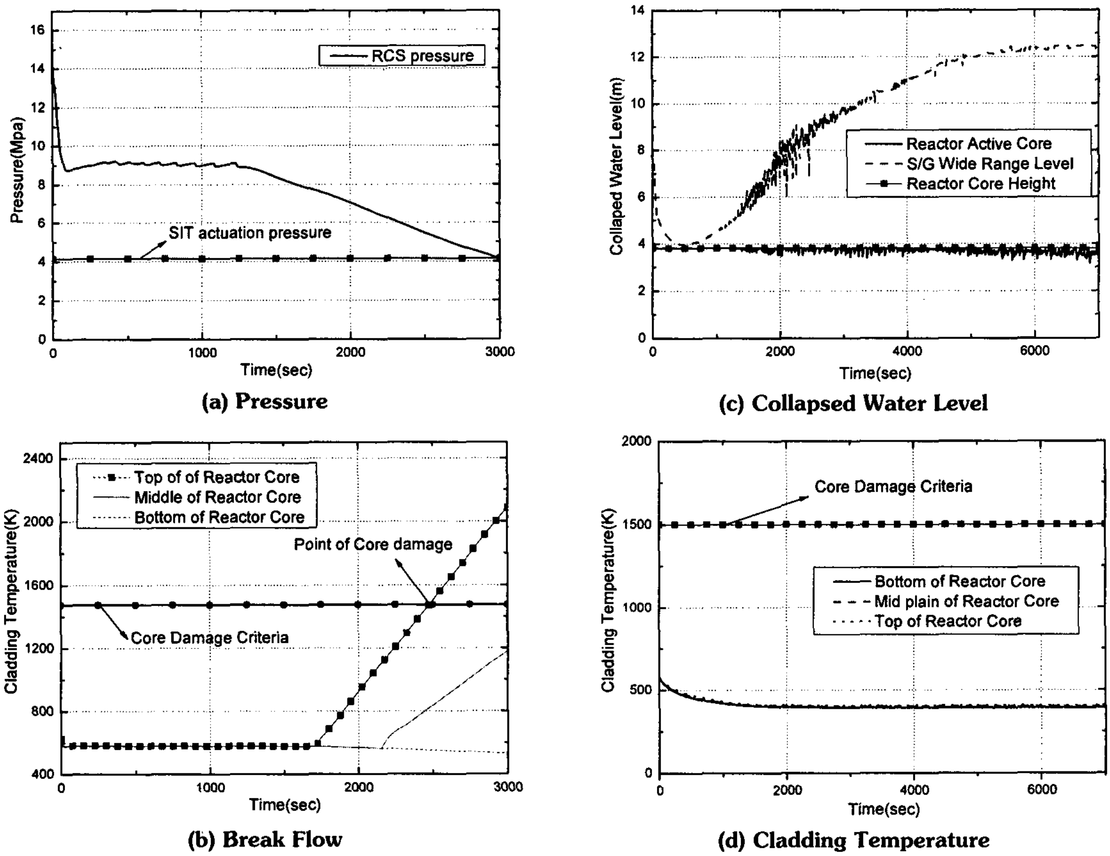


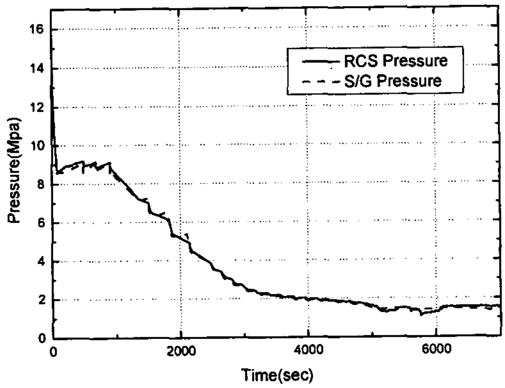
Fig. 5.Thermal-hydraulc Behavior for the S/G Steam Dump

abrupt temperature change can impose a pressurized thermal shock on the RCP piping and vessel materials [25]. In this section, we analyze the feasibility of accident mitigation when a limited cool-down rate of 55℃ /hr is maintained.

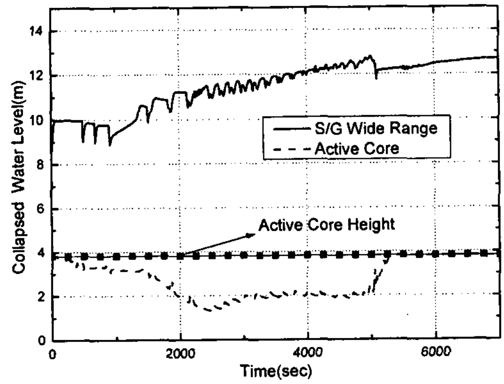
4.4.1. RCS Cool-down with SIT & LPSI

In this section, the scenario of an aggressive cool-down of RCS with SIT and LPSI is simulated. Since this accident occurs during the PSV test period, the operator can perceive the accident immediately. We assumed that an operator may control the ADVs at 15 minutes after the PSV fails to close, which is similar to an accident scenario during full power operation [11]. As shown in Figure 6(a), after the ADV control starts, the pressure of the RCS rapidly decreases. The core heat-up starts at about 1800 seconds (Figure 6(b)), at which point SIT actuation pressure is not yet reached. When the RCS pressure reaches 4.1 MPa, at about 2500 seconds, SIT starts to inject water to RCS, thereby stopping the rapid increase of cladding temperature. Since the SIT injection flow is not sufficient to fill up the depleted RCS water inventory, as shown in Figure 6(c), the cladding temperature slightly increases between

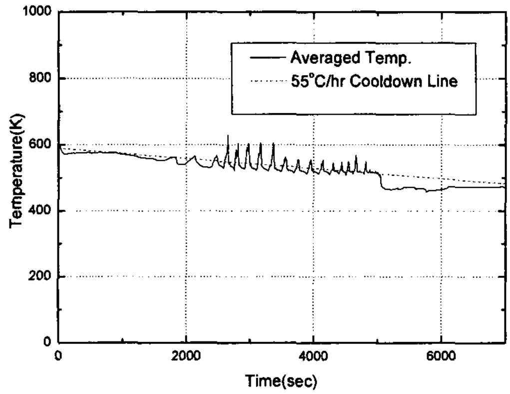
4000 seconds and 5000 seconds, as in Figure 6(b). When the LPSI actuation pressure is reached by a continuous cool-down operation, the core temperature rapidly decreases at about 5000 seconds, as in Figure 6(b). Figure 6(d) shows the average temperature of the cold and hot leg compared to the 55 ℃/hr cool-down line, which shows that the RCS temperature is controlled appropriately according to the cool-down line. Figure 6(e) also shows the break flow at the PSV, SI flow rate, and LPSI flow rate. From these results, we conclude that core damage can be prevented under the condition of a HPSI failure if an aggressive cool-down with a 55 ℃/hr cool-down rate is successfully operated and the SIT and LPSI are subsequently carried out.



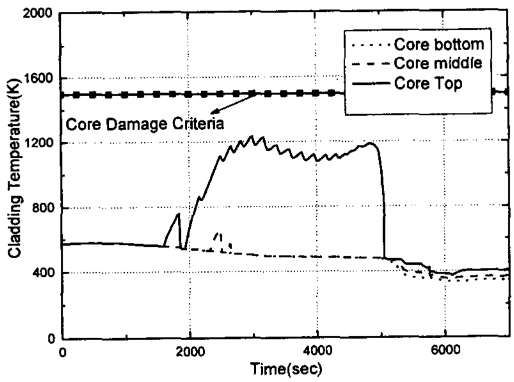
(a) Pressure



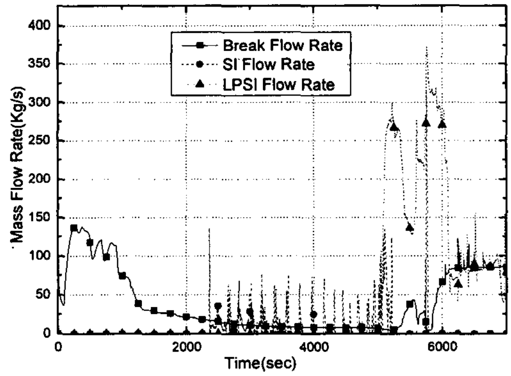
(b) Collapsed Water Level



(d) RCS Avg. Temperature



(c) Cladding Temperature

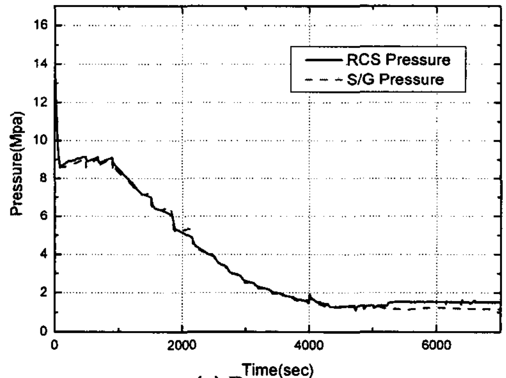


(e) Mass Flow Rate

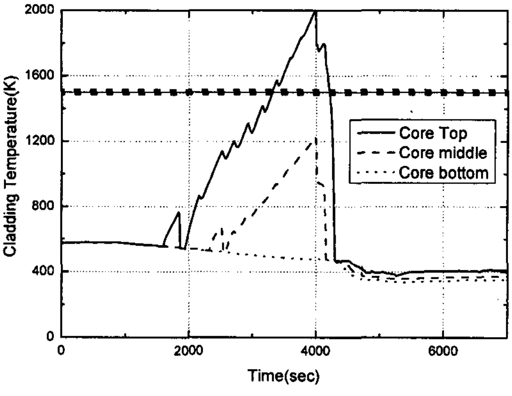
Fig. 6. Thermal-hydraulic Behaviors for RCS Cooldown with SIT and LPSI

4.4.2. RCS Cool-down Without SIT

In order to obtain minimum safety functions for the mitigation of an accident, a scenario was simulated in which a SIT injection failure occurred in addition to a HPSI failure. The control method of the ADV was the same as in the previous case. Since the RCS pressure cannot be lowered to the LPSI set-point using a given cool-down rate of 55 ℃/hr, core damage occurs at about 3000 seconds. Figure 7 shows the behavior of the pressure and core temperature.



(a) Pressure



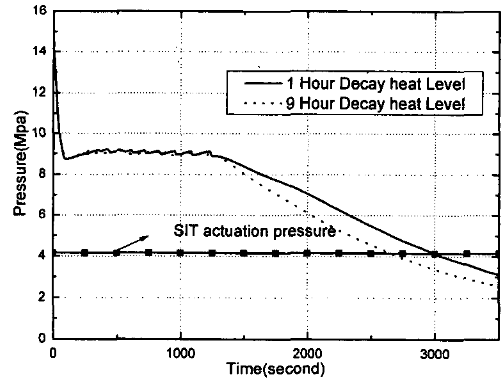
(b) Cladding Temperature

Fig. 7. Thermal-Hydraulic Behaviors for RCS Cooldown LPSI

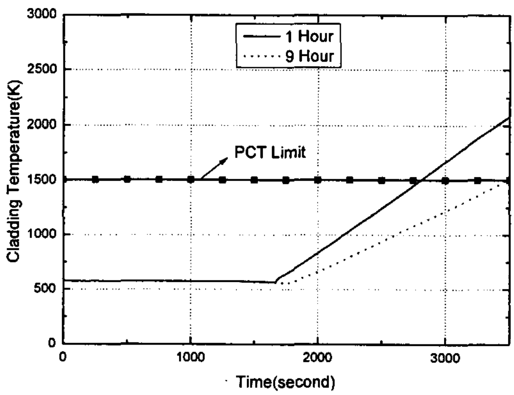
4.5. Sensitivity Analysis

4.5.1. The Effects of Decay Heat Level

Since the decay heat generation decreases sharply with time, it may be desirable to test the PSV after sufficient time has elapsed following reactor shutdown. The decay heat level used in the present analysis was 1 hour after reactor shutdown. Presently, it is under consideration for the test time of a PSV to be prolonged to 9 hours after shutdown in order to reduce the risk of a PSV failing to close. To compare the effects on the test time delay, the scenario in section 4.2



(a) Pressure



(b) Cladding Temperature

Fig. 8. The Effects of the Decay Heat Level

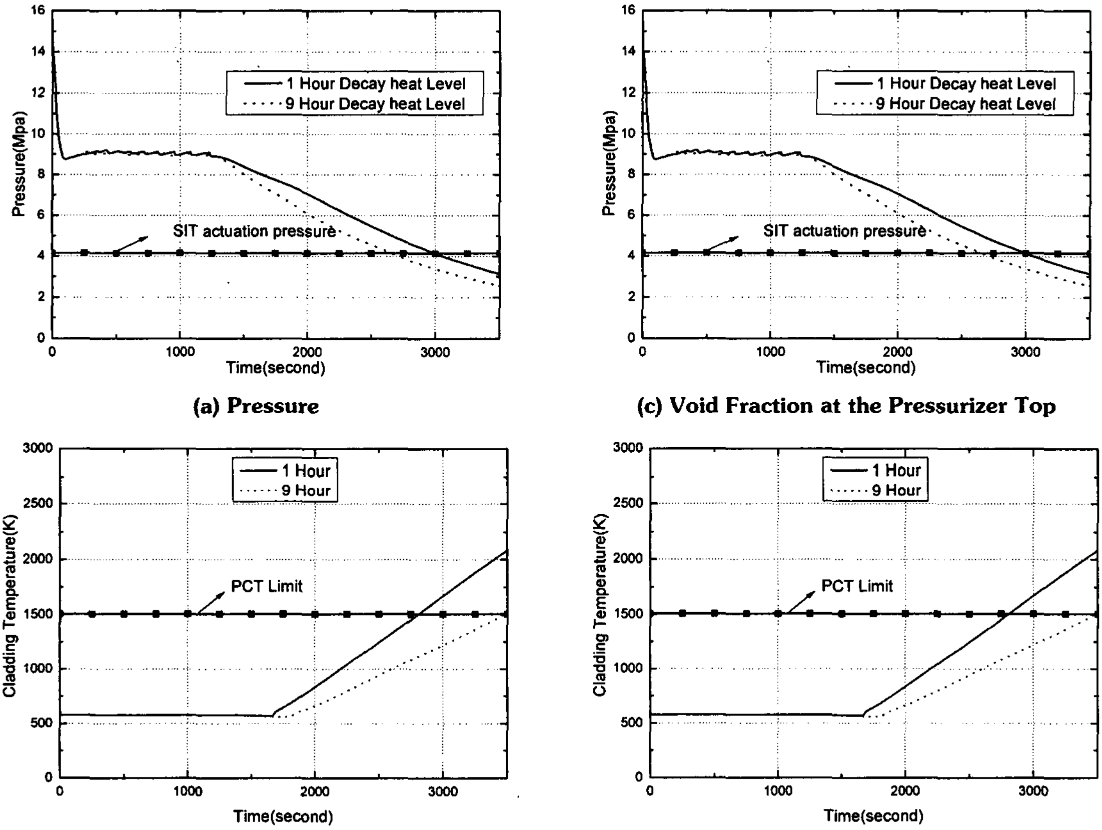
with a 9-hour decay heat level was simulated. Especially, we concentrated on the feasibility of the SIT actuation without aggressive cool-down operation.

As shown in Figure 8(a), the RCS pressure of a 9-hour decay heat level decreases faster than that of a 1-hour decay heat level. The heat-up rate of the cladding significantly decreases in the 9-hour decay heat level, as shown in Figure 8(b). However, the RCS pressure is still high compared with that of the cool-down operation in section 4.4.1. Consequently, to mitigate an accident in which a PSV fails to close following HPSI failure, a subsequent RCS cool-down operation is also needed for the 9-hour decay heat level.

4.5.2. The Effects of Cool-Down Initiation Time

Since the PSV may fail to close in the LP/SD period during valve testing, the operator can immediately perceive the accident and can prepare the accident-mitigating procedure within a few minutes. In section 4.4, we assumed the operator action time was 15 minutes after the accident. For comparison purposes, in this sensitivity analysis we assumed that the operator could start the mitigating action at 5 minutes after the accident.

As shown in Figure 9, the thermal-hydraulic behavior is similar to that of section 4.4.1.



(a) Pressure

(c) Void Fraction at the Pressurizer Top

(b) Cladding Temperature

(b) Break Flow

Fig. 9. The Effects of the Aggressive Cool-down Initiation Time

Unexpectedly, the pressure of the 5-minute response case is higher than that of the 15-minute response case. The depressurization delay is due to the late transition to single phase vapor critical flow at the PSV in the 5-minute response case. Since the break flow of the 15-minute response case is larger than that of the 5-minute response case in the early stages of an accident (Figure 9(d)), the PSV of the 15-minute response case is disclosed in a single phase vapor faster than that of the 5-minute response case (Figure 9(c)). If the break area is exposed to steam, the flow pattern is changed from a two-phase critical flow to a single phase vapor critical flow, in which the volumetric flow rate increases. Thus, RCS depressurization is accelerated [12, 13].

Although the time of SIT actuation is delayed at the 5-minute response case, the peak cladding temperature of the 5-minute response case is lower than that of the 15-minute response case, as shown in Figure 9(b). This is due to the decrease of break flow in the 5-minute response case, which thus maintains a larger RCS water inventory. From this analysis, it is expected that starting a cool-down operation early may be an effective way to lower the peak cladding temperature.

5. Concluding Remarks

An accident in which the PSV fails to close has a significant impact on the potential for LPSD problems. To find a way to mitigate the severity of the accidents in the scenarios, we have simulated the failure of a PSV to fully close during the LPSD period. We found that this type of accident in a KSNP has the characteristics of both a small and medium-break LOCA; that is, the RCS pressure decreases slowly but the break flow is large enough to uncover the reactor core in the early stages of an accident. For accident sequences including HPSI failure, the RCS depressurization operation is needed due to the slow RCS pressure change. We simulated an aggressive cool-down by a S/G for this accident sequence to investigate the feasibility of subsequent safety injections such as SIT and LPSI. From these simulations, if the operator starts the cool-down operation within an appropriate time frame, it was found that the SIT and LPSI system could be actuated and core damage could be prevented. We expect that, if these results are reflected in the LPSD abnormal response guidelines, LPSD safety will be significantly improved. Also, since these results are applicable to the power operation period, they will help to enhance the quality of full power PSA.

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Nomenclature

G : Mass flux
W : mass flow rate
A : area
P : pressure
γ : polytropic constant
v : specific volume of steam
C_D : Discharge Coefficient
Δt : time step size
T : temperature

superscript
n : 'n' th Time step
subscript
c : choked
$h1$: hot leg 1
$h2$: hot leg 2
$c1$: cold leg 1
$c2$: cold leg 2

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