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Estimation of Operator Action Time Windows by RELAP5/MOD3.3

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ABSTRACT

This report presents the results of analyses performed for the updated human reliability analysis. The analysis estimates time windows available to perform operator action to satisfy the success criteria to prevent core damage. The best-estimate RELAP5/MOD3.3 computer code was used. In the past, the conventional probabilistic safety assessment used a conservative approach to address this factor. However, the current standard for probabilistic safety assessment recommends the use of best-estimate codes. The RELAP5/MOD3.3 best-estimate code calculations were performed for three selected cases in which human actions supplement safety system actuations: (1) small or medium loss-of-coolant accident requiring a manual start of the auxiliary feedwater system, (2) loss of normal feedwater requiring a manual start of the auxiliary feedwater system, and (3) a loss-of-coolant accident requiring manual actuation of the safety injection signal. The analysis used a qualified RELAP5 input model representing a Westinghouse-type, two-loop pressurized water reactor for the calculations. The results of the deterministic safety analysis were examined to identify the latest time that an operator could act and still satisfy the safety criteria. The results show that the time available to perform operator action (i.e., the time window) is greater than the actual time needed to perform the action. The difference is considered additional available time for action. The results of human reliability analysis show that uncertainty analysis of realistic deterministic safety analysis is needed only for significant risk contributors in situations where having additional time available for action makes the difference between considering or not considering recovery operator action.

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ABBREVIATIONS

AFW	auxiliary feedwater
cm	centimeter
ECCS	emergency core cooling system
HEP	human error probability
HPSI	high-pressure safety injection
HRA	human reliability analysis
in.	inch
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LPSI	low-pressure safety injection
m	meter
MAAP	Modular Accident Analysis Program
MFW	main feedwater
min	minute
MPa	megapascal
MWt	megawatt thermal
NPP	nuclear power plant
PORV	power-operated relief valve
PSA	probabilistic safety assessment
PWR	pressurized-water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RY	reactor-year
s	second
SG	steam generator
SI	safety injection
T_a	additional available time for action
T_p	actual time needed to perform the action
T_w	time window of the action

1. INTRODUCTION

To estimate the time windows of operator actions to satisfy the success criteria, i.e. core cooling criteria to prevent core damage in level 1 probabilistic safety assessment (PSA), the conventional PSA has used the results of a severe accident code such as the MAAP (Modular Accident Analysis Program). However, information obtained with such codes is often too conservative to permit a realistic PSA for a risk-informed application. Instead, the PSA standard (Ref. 1) recommends the use of a best-estimate code to improve the quality of a PSA. Therefore, the aim of this study was to estimate the operator action time windows, which satisfies the criteria for core cooling, needed for updated human reliability analysis (HRA) by using the RELAP5/MOD3.3 Patch 03 best-estimate computer code (Ref. 2). The specified time windows are important for HRA to determine the likelihood of operator actions. The human error probability of a specified action is lower if operators have more time to act. In the control room of a nuclear power plant, a team of operators works under the supervision of a shift supervisor. If operators have more time to act, their colleagues or the shift supervisor may have time to observe and correct a possible error. Consideration of correction the error (recovery action) causes lower human error probability and may result in human error having a different impact on the overall PSA results. The actual times needed to perform the action were assessed on the basis of simulator scenarios, while the time windows were identified by deterministic safety analysis. In this study, RELAP5/MOD3.3 best-estimate code calculations were performed for three selected initiating events: (1) establishing auxiliary feedwater (AFW) in case of a small or medium loss-of-coolant accident (LOCA), (2) establishing AFW in case of transient (loss of feedwater (LOFW) being the most limiting transient), and (3) manually actuating the safety injection (SI) signal in a LOCA. In these events, human actions supplement the safety system actuations. The qualified RELAP5 input model representing a Westinghouse-type, two-loop pressurized water reactor (PWR) was used for the calculations (Ref. 3).

Section 2 briefly describes the Krško Nuclear Power Plant (NPP). Section 3 describes the RELAP5 input model, and Section 4 presents the scenarios. Section 5 shows the RELAP5/MOD3.3 calculations, which are the basis for determining the time windows of operator actions. Use of these time windows in HRA reveals how a change in human error probability can impact the core damage frequency. The run statistics for calculations are given separately in Section 6, while conclusions appear in Section 7.

2. PLANT DESCRIPTION

Krško NPP is a Westinghouse two-loop PWR plant with a large dry containment. The plant has been in commercial operation since 1983. After its modernization in 2000, the plant fuel cycle was gradually prolonged from 12 (Cycle 17) to 18 months (Cycle 21).

The power rating of the Krško NPP nuclear steam supply system is 2,000 megawatt thermal (MWt) (1,882 MWt before the plant modernization and power uprate) composed of 1,994 MWt (1,876 MWt before the plant modernization and power uprate) core power output plus 6 MWt of reactor coolant pump (RCP) heat input. The nuclear steam supply system consists of a PWR, a reactor coolant system (RCS), and associated auxiliary fluid systems. The RCS is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing an RCP and a steam generator (SG). An electrically heated pressurizer is connected to one of the loops.

The reactor core is composed of 121 fuel assemblies. Square spacer grid assemblies and the upper and lower end fitting assemblies support the fuel rods in fuel assemblies. Each fuel assembly is composed of 16x16 rods. Of these, fuel rods use only 235 places; of the 21 remaining places, 20 are evenly and symmetrically distributed throughout the cross-section of the assembly and are provided with thimble tubes, which may be reserved for control rods, and one control instrumentation tube for an in-core thimble.

The RCPs, one per coolant loop, are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type.

The SGs, one per loop, are vertical U-tube units of the Siemens-Framatome steam generator type SG 72 W/D4-2, installed during the plant modernization in 2000. They replaced highly degraded Westinghouse D-4 steam generators, each having preheating section.

Engineered safety features are provided to prevent accident propagation or to limit the consequences of postulated accidents, which might otherwise lead to damage of the system and release of fission products. This plant has the following engineered safety features:

- containment spray system
- hydrogen control system
- emergency core cooling system (ECCS)
- component cooling water system
- essential service water system
- AFW system

In 2006, the main turbine was replaced to gain additional power from the new SGs.

3. INPUT MODEL DESCRIPTION

To perform this analysis, Krško NPP has provided the base RELAP5 input model (the so-called "master input deck"), which has been used for several analyses, including reference calculations for the Krško full-scope simulator verification (Refs. 3, 4, 5). Figure 1 presents the scheme of the Krško NPP nodalization for the RELAP5/MOD3.3 code. The analyses were performed for uprated power conditions (2,000 MWt) with new SG and Cycle 21 settings, corresponding to the plant state after outage and refueling in September 2004.

The model consists of 469 control volumes, 497 junctions, and 378 heat structures with 2,107 radial mesh points. In addition, 574 control variables and 405 logical conditions (trips) represent the instrumentation, regulation isolation, SI and AFW triggering logic, steamline isolation, and other functions.

3.1 Hydrodynamic Component Description

The numbering scheme relates certain RELAP5 hydrodynamic component numbers to certain plant systems and components. In the following, XX indicates numbers between 00 and 99:

- Hydrodynamic components 0XX represent parts of the primary side without the reactor vessel and both loops.
- Hydrodynamic components 1XX represent the reactor vessel.
- Hydrodynamic components 2XX represent Loop 1.
- Hydrodynamic components 3XX represent Loop 2.
- Hydrodynamic components 4XX represent the secondary side (SG1 side).
- Hydrodynamic components 5XX represent the secondary side (SG2 side).
- Hydrodynamic components 6XX represent the turbine, steam dump, and AFW piping from pumps up to the header.
- Hydrodynamic components 7XX represent ECCS1.
- Hydrodynamic components 8XX represent ECCS2.
- Hydrodynamic components 9XX represent the main feedwater (MFW) and AFW pumps, refueling water storage tank, condensate storage tank, containment, atmosphere to which discharges steam generator relief and safety valves, and cold leg break model.

Modeling of the primary side without the reactor vessel and both loops includes the pressurizer vessel, pressurizer surge line, pressurizer spray lines and valves, two pressurizer power-operated relief valves (PORVs) and two pressurizer safety valves, chemical and volume control system charging and letdown flow, and RCP seal flow.

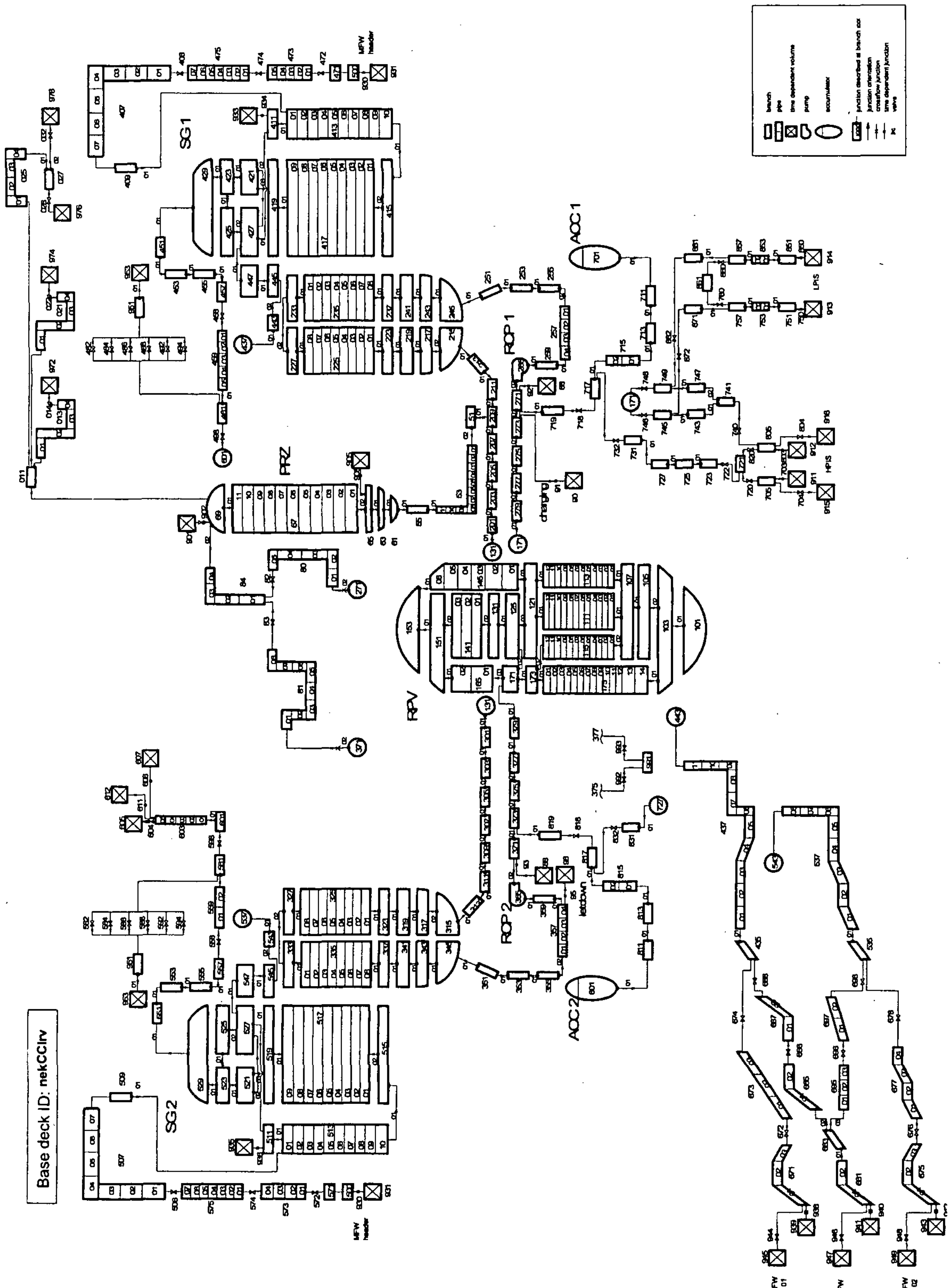

Figure 1 Krško NPP nodalization scheme

The reactor vessel consists of the lower downcomer, lower head, lower plenum, core inlet, reactor core, core baffle bypass, core outlet, upper plenum, upper head, upper downcomer, and guide tubes.

The primary loop is represented by the hot leg, primary side of the SG, intermediate leg with cold leg loop seal, and cold leg, separately for Loop 1 (2XX) and Loop 2 (3XX). Loops are symmetrical except for the pressurizer surge line and the chemical and volume control system connections layout. The primary side of the SG consists of the inlet and outlet plenum, tubesheet, and the U-tube bundle represented by a single pipe.

The secondary side consists of the SG secondary side (riser, separator and separator pool, downcomer, steam dome), main steamline, main steam isolation valves, SG relief and safety valves, MFW piping, and AFW piping from the header to the SG. The AFW injects above the SG riser. The main steamline No. 1 (4XX) has the same volume as the main steamline No. 2 (5XX), but the geometry data differ depending on the pipeline.

Components numbered 6XX represent the AFW piping from AFW pumps to the AFW header.

ECCS piping includes high-pressure safety injection (HPSI) pumps, accumulators, and low-pressure safety injection (LPSI) pumps. The hydrodynamic components representing HPSI and LPSI pumps are time-dependent junctions, while for accumulators the 'accum' hydrodynamic component was used. The ECCS connects to both cold legs and directly to the reactor vessel.

Components 7XX represent Train 1 of the ECCS plus common lines for reactor vessel injection, and Components 8XX represent ECCS Train 2.

Among the components numbered 9XX, the MFW and AFW pumps are modeled as time-dependent junctions that pump water from time-dependent volumes, representing those of the condensate storage tank. For AFW pumps, recirculation flow is also modeled. The refueling water storage tank is modeled with time-dependent volume, similarly to the modeling of containment and atmosphere. The break in the cold leg is modeled with two valves, which allows the possibility of modeling a double-ended guillotine break.

3.2 Control and Protection Logic

To accurately represent the Krško NPP behavior, the model includes many control variables and general tables. They represent protection, monitoring, and simplified control systems used only during steady-state initialization, as well as main plant control systems:

- rod control system
- pressurizer pressure control system
- pressurizer level control system
- steam generator level control system
- steam dump

The rod control system has been modeled for point kinetics. The present model can be used for transient analysis with two options:

(1) with constant or predefined core power transient as a function of time (including decay power calculation)

(2) with the rod control system in auto or manual mode

The following plant protection systems are defined using trip logic:

- reactor trip
- SI signal
- turbine trip
- steamline isolation
- MFW isolation
- AFW start

4. SAFETY ANALYSIS METHODOLOGY

The RELAP5 input model, described in Section 3, was applied to the selected scenarios, which were needed to update the HRA. The latest available RELAP5/MOD3.3 Patch 03 was used for the calculations. For the selected scenarios, the analysis determined the time windows for operator action. This section first describes the success criteria for determination of the time windows. Then, the scenario is described for each of the three selected cases in which human actions supplement safety system actuations. The selected cases are (1) a small or medium LOCA requiring manual AFW start, (2) LOFW requiring a manual AFW start, and (3) a LOCA requiring manual actuation of the SI signal.

4.1 Description of Success Criteria

Safety analyses include variations of the timing of human action to determine the latest time that operators can perform the needed action so that the main plant parameters do not exceed their limits. The analysis used the core cooling success criteria as defined in Reference 6. The success criteria used in level 1 PSA are the quantified definition of core damage. Given a certain sequence and plant response to that sequence, we can assess weather the sequence involves core damage by comparing with criteria. It is assumed that if the hottest core fuel/clad node temperature in the reactor core exceeds 923 kelvin (K) for more than 30 minutes, or if the temperature of the core exceeds 1,348 K, core damage may occur, which may lead to an accident state. For overpressurization, the criterion is that primary pressure should not exceed 18.95 megapascals (MPa). Based on these criteria, the analysis determined the time window for operator action.

4.2 Scenario Descriptions

This section describes the three scenarios needed for an updated HRA. In these scenarios, the human actions supplement the safety system actuations. In the first scenario, the human action is to establish AFW in the case of a small or medium LOCA when the HPSI system fails. In the second scenario, the human action is to establish AFW in the case of an LOFW transient. In the third scenario, the human action is to actuate the SI signal in the case of the most limiting accident, excluding a large-break LOCA (i.e., for a small or medium LOCA).

The operator actions considered in the analyses are delayed AFW pump manual start, RCP trip according to emergency operating procedure (one HPSI pump running and subcooling below 14 K), and HPSI pump termination according to emergency operating procedure criteria (pressurizer pressure above 13.83 MPa, pressurizer level above 10 percent, and subcooling greater than 19 K).

In the case of a small or medium LOCA in a nuclear power plant when the HPSI system fails, one means to cool the reactor is through secondary-side depressurization, provided that the AFW system is operating. Normally, the AFW system starts automatically when MFW system is lost. If the AFW pumps do not start automatically, operators should intervene. The success criterion requires operation of one of three AFW pumps to maintain the flow in order to depressurize the primary system below the accumulator injection setpoint at 4.9 MPa. The analysis assumed that passive accumulators, as well as LPSI, are available. The parameter indicating depressurization is primary pressure, and the parameter indicating core cooling is average rod cladding temperature. As larger breaks, after some time, can cause depressurization (through the break) in any case where the pressure falls below the accumulator injection setpoint pressure, AFW system is not needed for depressurization. Therefore, the analysis was performed for a spectrum of break sizes from 1.27 centimeters (cm) (0.5 inch (in.)) to 15.24 cm (6 in.) to determine for which break sizes operation of one AFW pump is needed to depressurize the primary system below the accumulator injection setpoint. For the most critical break in terms of depressurization, the analysis determined the time available to start the AFW pump based on the parametric study of varying delays of the AFW pump start. The break was located in the cold leg between the RCP and the reactor vessel.

The most limiting transient requiring operation of the AFW system is LOFW. The success criterion is that the capacity of one train of AFW is adequate to remove decay heat, to prevent overpressurization of the primary system, and to prevent the uncovering of the core from resulting in core heatup. The analysis varied the time when the operator succeeds in starting the AFW pump. When the AFW pump starts to inject into the secondary side, cooling of the secondary side causes the pressurizer pressure to drop below the pressurizer PORV closure setpoint and then below the maximum pressure capacity of the HPSI pump. The HPSI injection efficiently prevents further uncovering of the core.

The third scenario considered was a LOCA without automatic SI signal actuation. This means that none of the safety systems, including HPSI, LPSI, and AFW, was assumed to be available. The analysis evaluated the whole spectrum of LOCAs, from a break size of 1.91 cm (0.75 in.) to 15.24 cm (6 in.).

5. RESULTS

5.1 Loss-of-Coolant Accident Calculations with Manual Actuation of Auxiliary Feedwater

5.1.1 Loss-of-Coolant Spectrum Calculations for Scenarios with High-Pressure Safety Injection Not Available

Figures 2 through 9 show the results for a spectrum of break sizes. As Figure 2 shows, breaks of 5.08 cm (2 in.) and larger depressurize (through the break), after some time, when the pressure falls below the accumulator injection setpoint pressure of 4.93 MPa. In this case, AFW system is not needed for depressurization, as evidenced by the SG1 wide-range level shown in Figure 8 and the mass released through the SG PORVs shown in Figure 9. After the initial decrease in level and the opening of the SG PORVs, the SG1 pressure shown in Figure 7 drops below the opening setpoints of the SG PORVs. Therefore, SG1 is not further emptied. The trends for SG2 pressure and wide-range level are similar to those for SG1 and are therefore not shown. On the other hand, breaks of 2.54 cm (1 in.) equivalent in diameter and smaller require depressurization. Because core heatup (Figure 4) occurs earlier for the 2.54-cm (1-in.) break than for the 1.91-cm (0.75–in.) and 1.27-cm (0.5–in.) break, the 2.54-cm (1-in.) break was identified as the most critical regarding the time available to start AFW pump. Figure 3, which shows RCS mass inventory, and Figure 8, which shows the SG1 wide-range level, confirm this finding. In the case of the 1.91-cm (0.75-in.) and 1.27-cm (0.5-in.) break, the RCS even repressurizes. However, the operator has more time before the RCS inventory is depleted, the SGs are dried, and the core is uncovered and heated up. Figure 8 shows that for a break of 2.54 cm (1 in.) (and smaller), the SGs begin to dry out and their inventory is lost through the SG PORVs (Figure 9). To establish cooling by the secondary side, AFW system is needed to fill the SG.

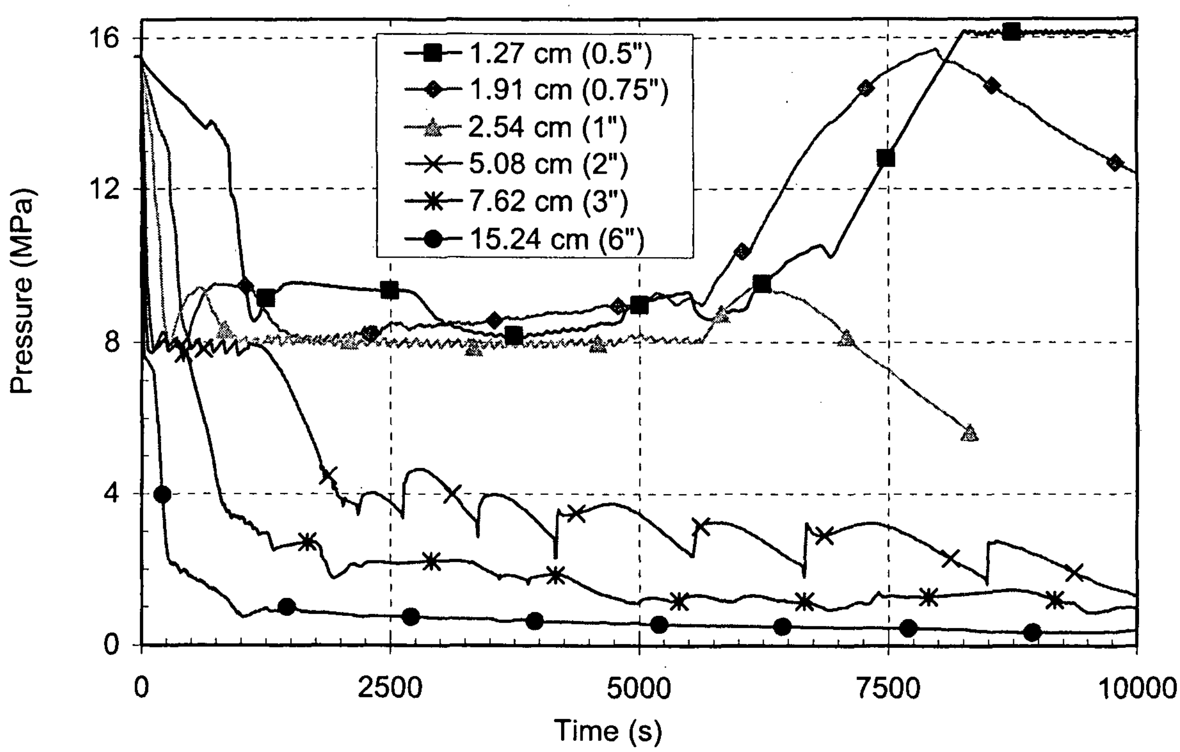

Figure 2 RCS pressure for a spectrum of LOCA break sizes

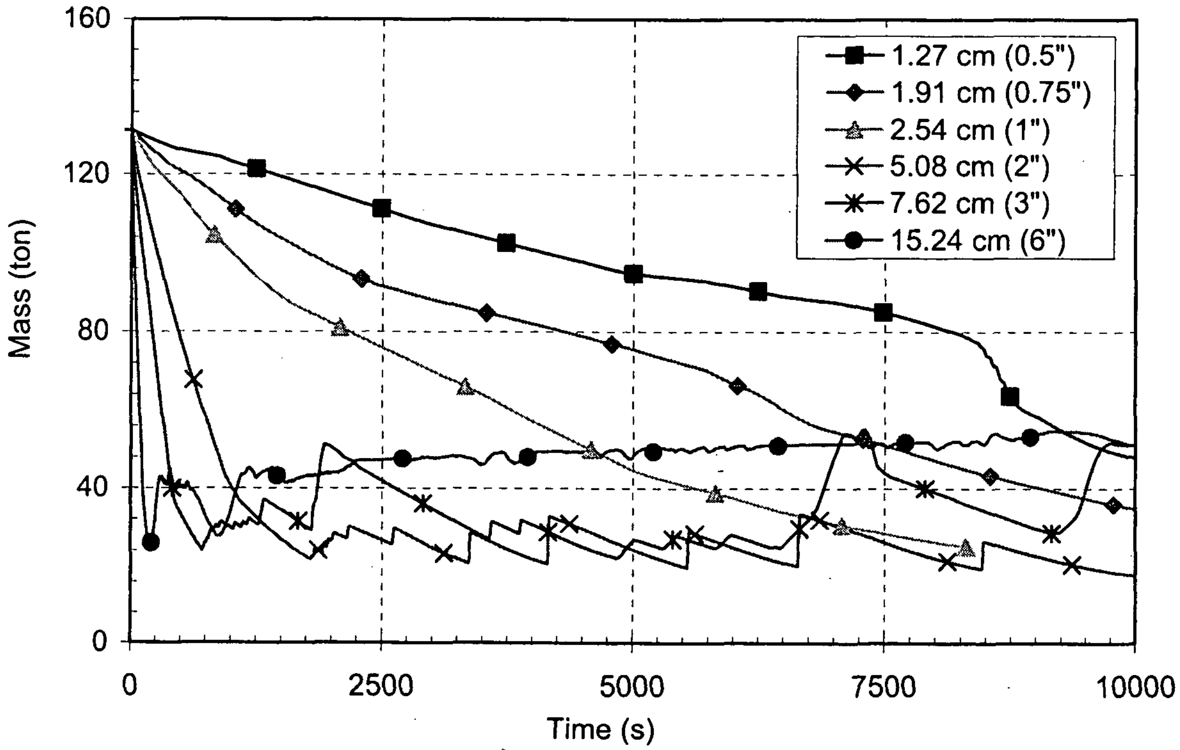

Figure 3 RCS mass inventory for a spectrum of LOCA break sizes

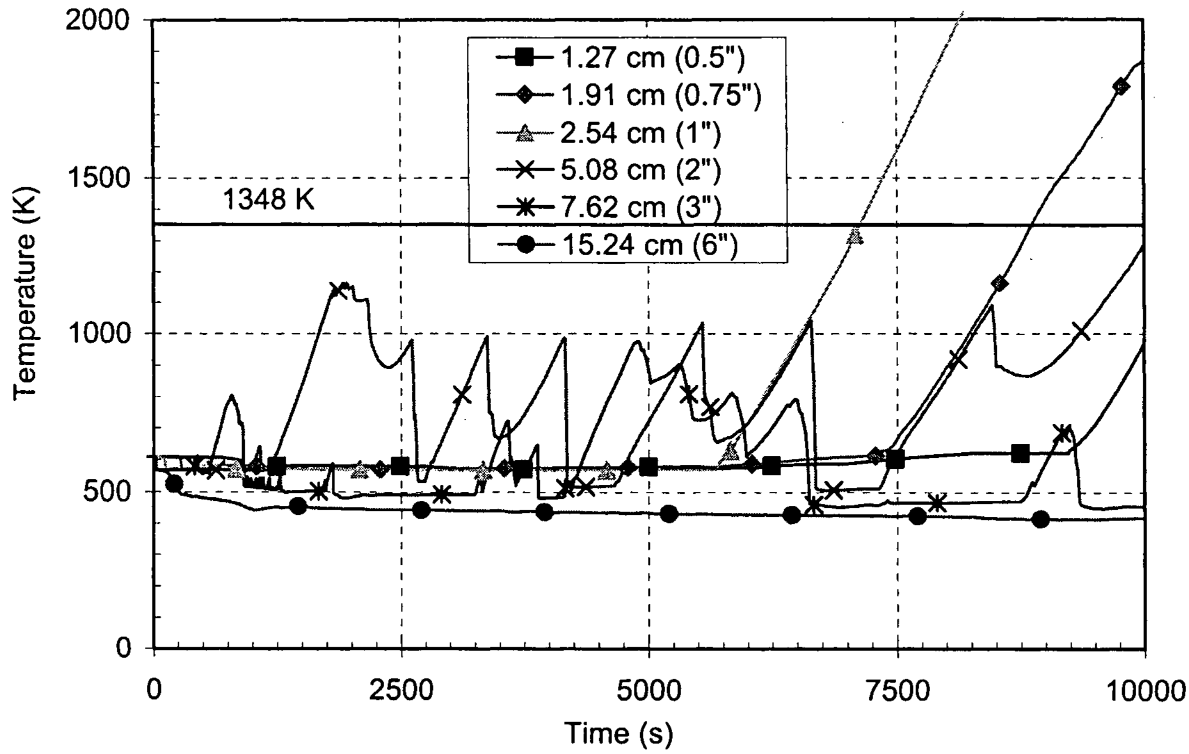

Figure 4 Core cladding temperature for a spectrum of LOCA break sizes

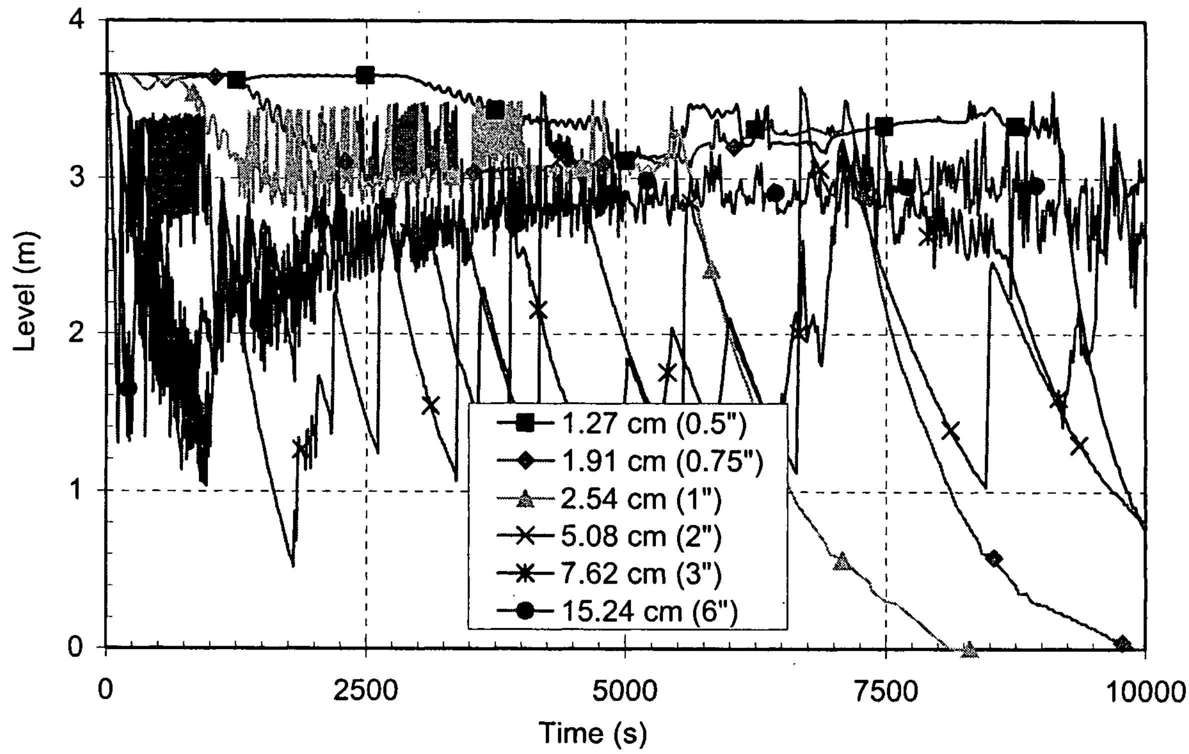

Figure 5 Core collapsed liquid level for a spectrum of LOCA break sizes

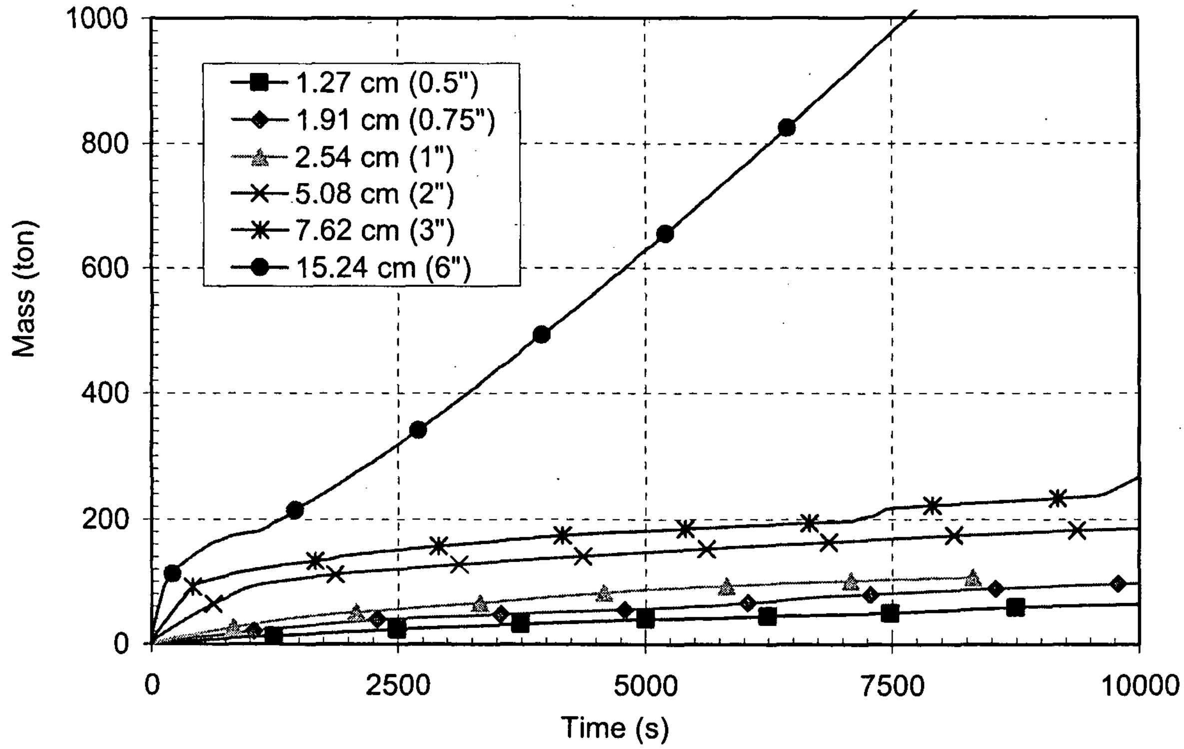

Figure 6 Mass discharged through break for a spectrum of LOCA break sizes

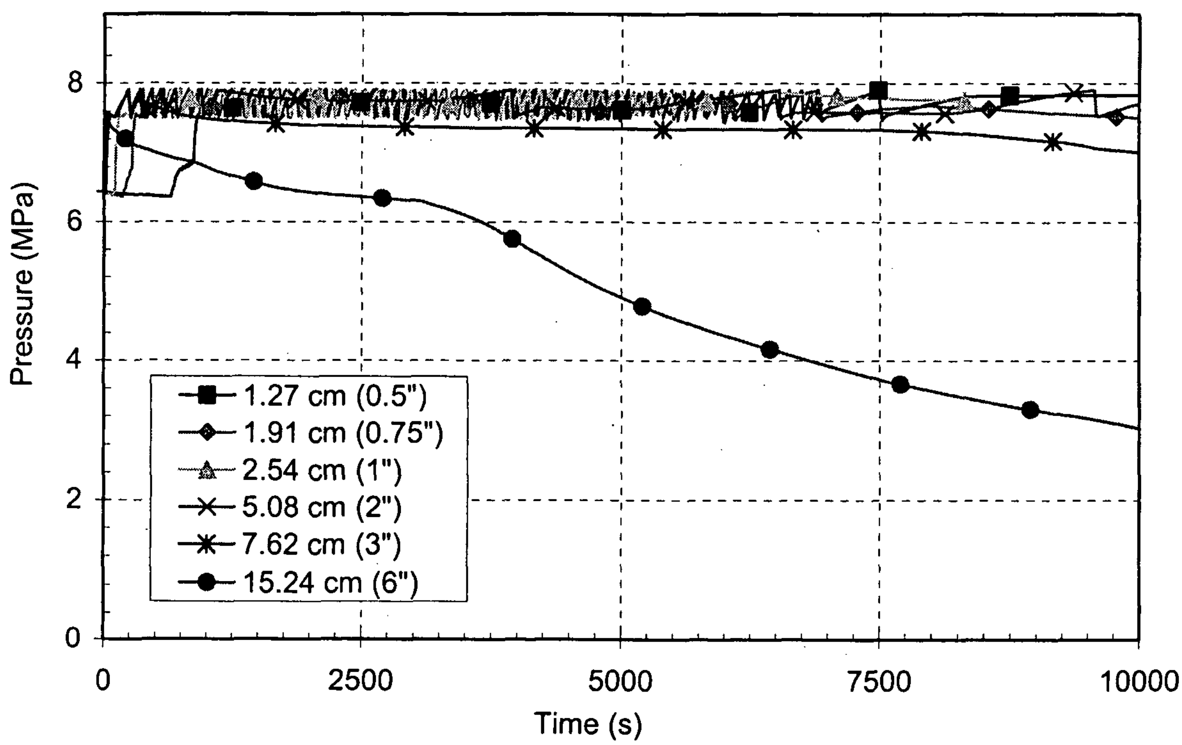

Figure 7 SG1 pressure for a spectrum of LOCA break sizes

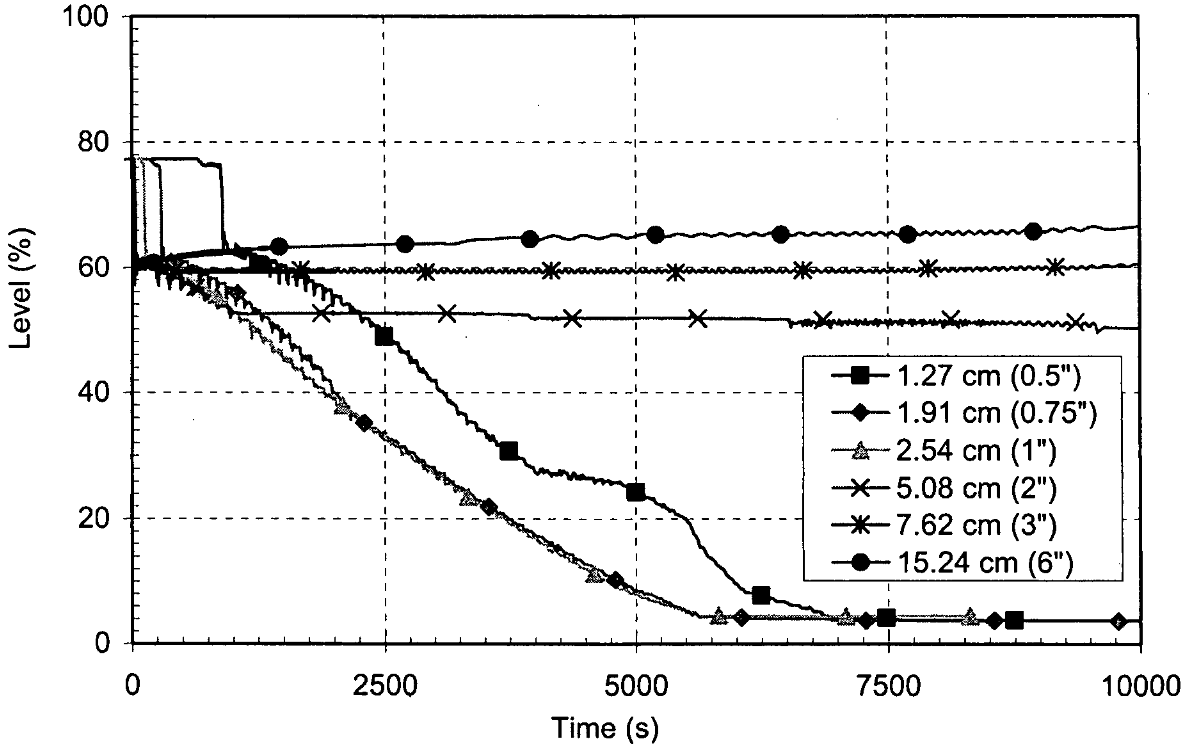

Figure 8 SG1 wide-range level for a spectrum of LOCA break sizes

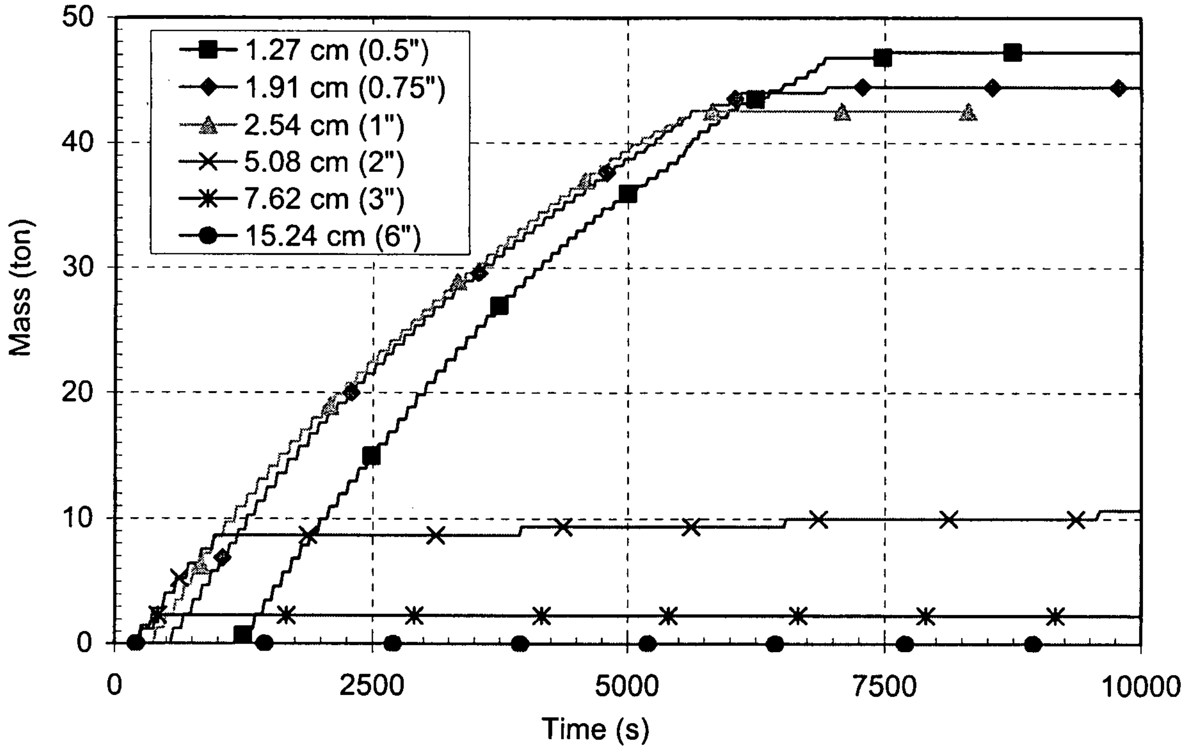

Figure 9 Mass discharged through SG1 PORV for a spectrum of LOCA break sizes

5.1.2 Calculations for a 2.54-cm Break Size Loss-of-Coolant Accident with Different Auxiliary Feedwater Delays

To determine the time window available to the operators to start AFW pump, five different scenarios were analyzed for a 2.54-cm (1-in.) break using different delays for the AFW pump start, as shown in Figures 10 through 17. Figure 10 shows that once AFW pump starts, the RCS cannot be depressurized. The RCS mass continuously decreases (Figure 11), and therefore the core begins to heat up (Figure 12) as it is uncovered (Figure 13). The secondary pressure (Figure 14) is such that the SG PORV is cycling, as can be seen from the stepwise line for mass released through the SG1 PORV (Figure 17). Until the AFW pump is started, the SG1 wide-range level decreases (Figure 15). However, the capacity of the AFW system (Figure 16) is sufficient to recover the SG level. After the steam generator level is recovered, the AFW pump injected intermittently to recover the mass lost through SG PORV cycling, but the cooling with SG PORV cycling is not sufficient to depressurize the primary system and prevent core heatup. To speed up the cooling by the secondary side, more steam should be released through the SG PORV. This can be achieved by manually fully opening the SG PORV, which is explained in the next section.

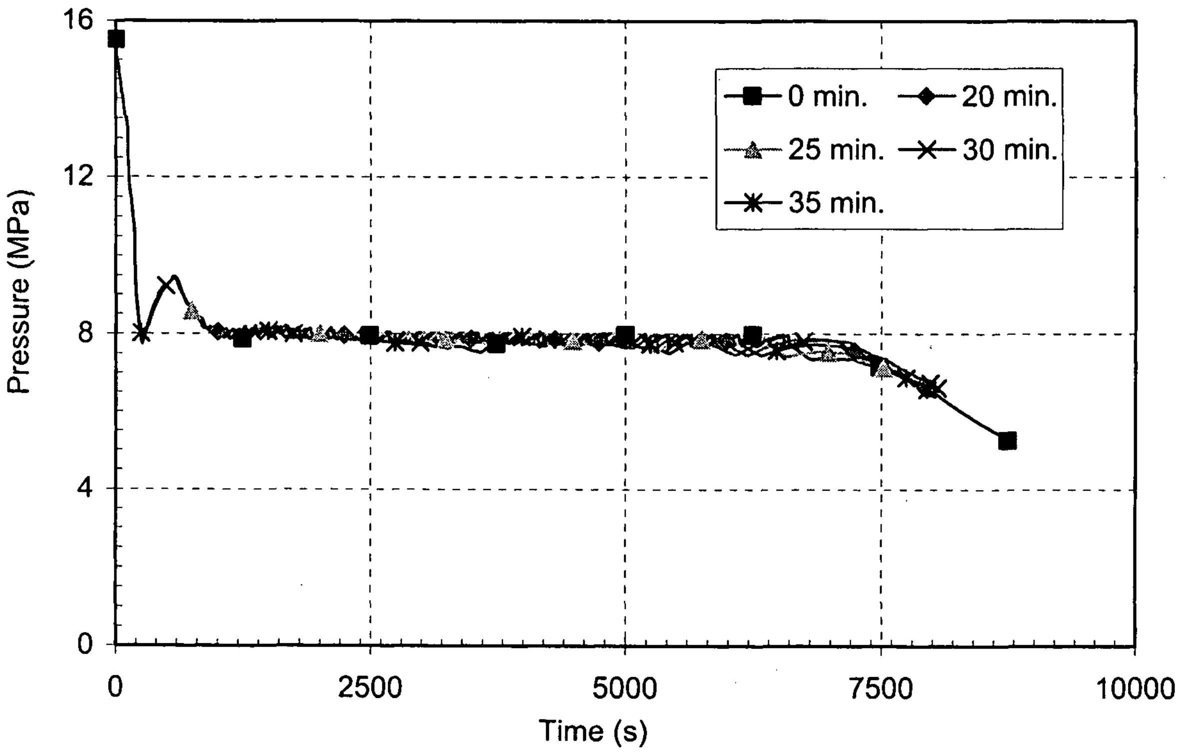

Figure 10 RCS pressure for 2.54-cm break size LOCA with AFW start delays

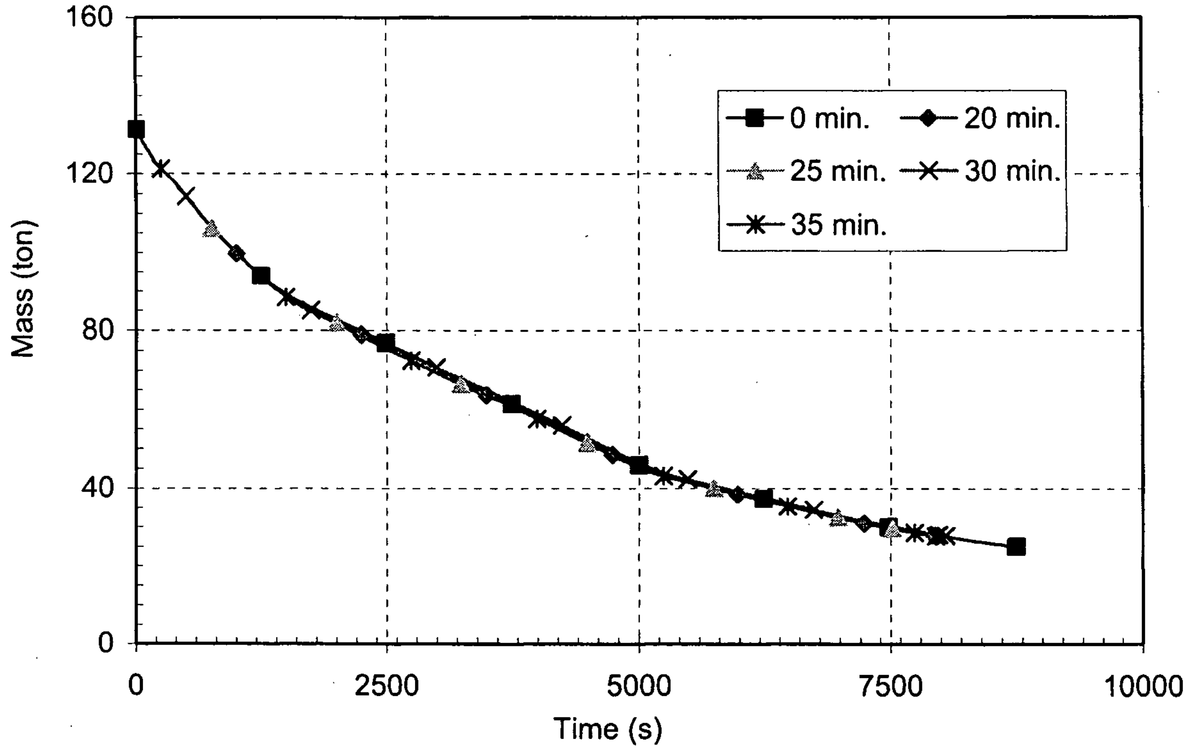

Figure 11 RCS mass inventory for break size 2.54-cm LOCA with AFW start delays

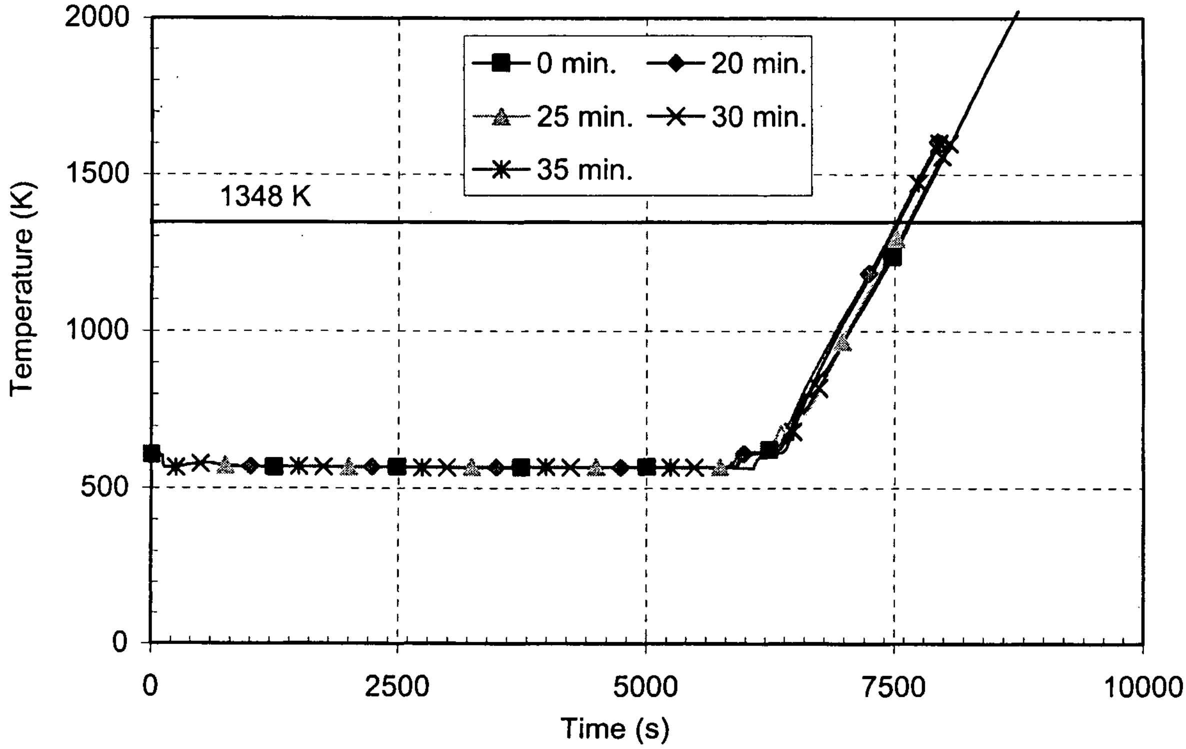

Figure 12 Core cladding temperature for 2.54-cm break size LOCA with AFW start delays

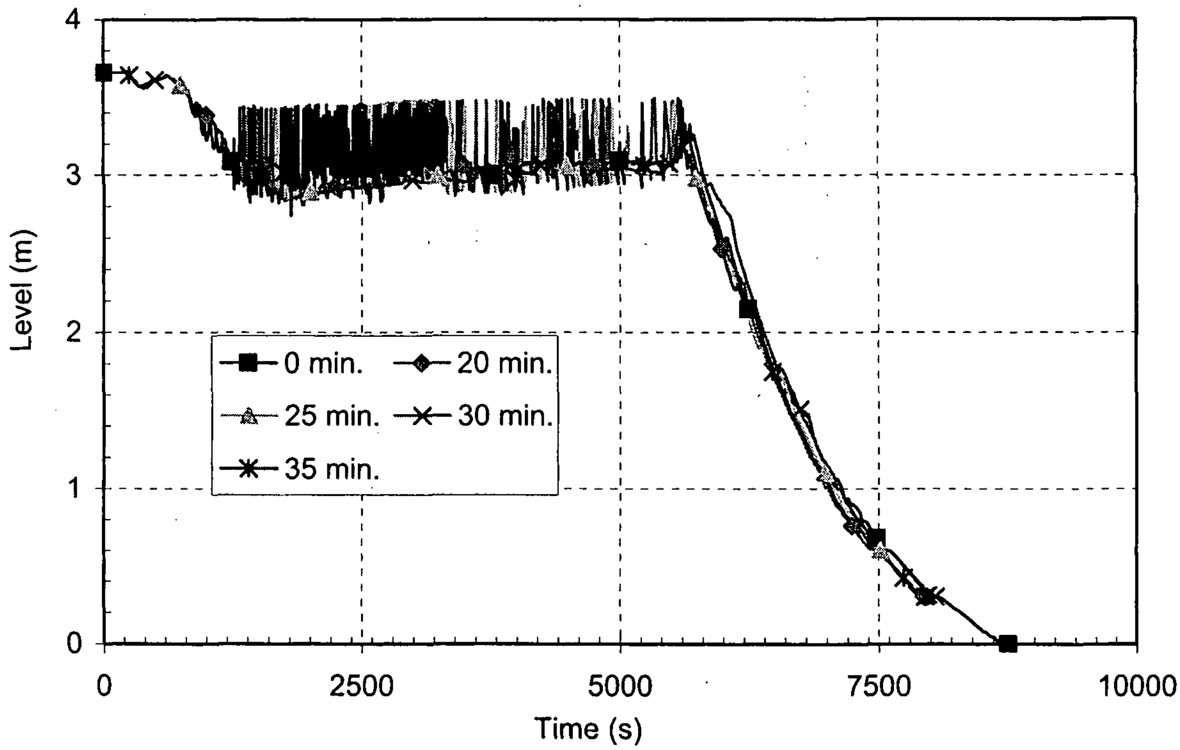

Figure 13 Core collapsed liquid level for 2.54-cm break size LOCA with AFW start delays

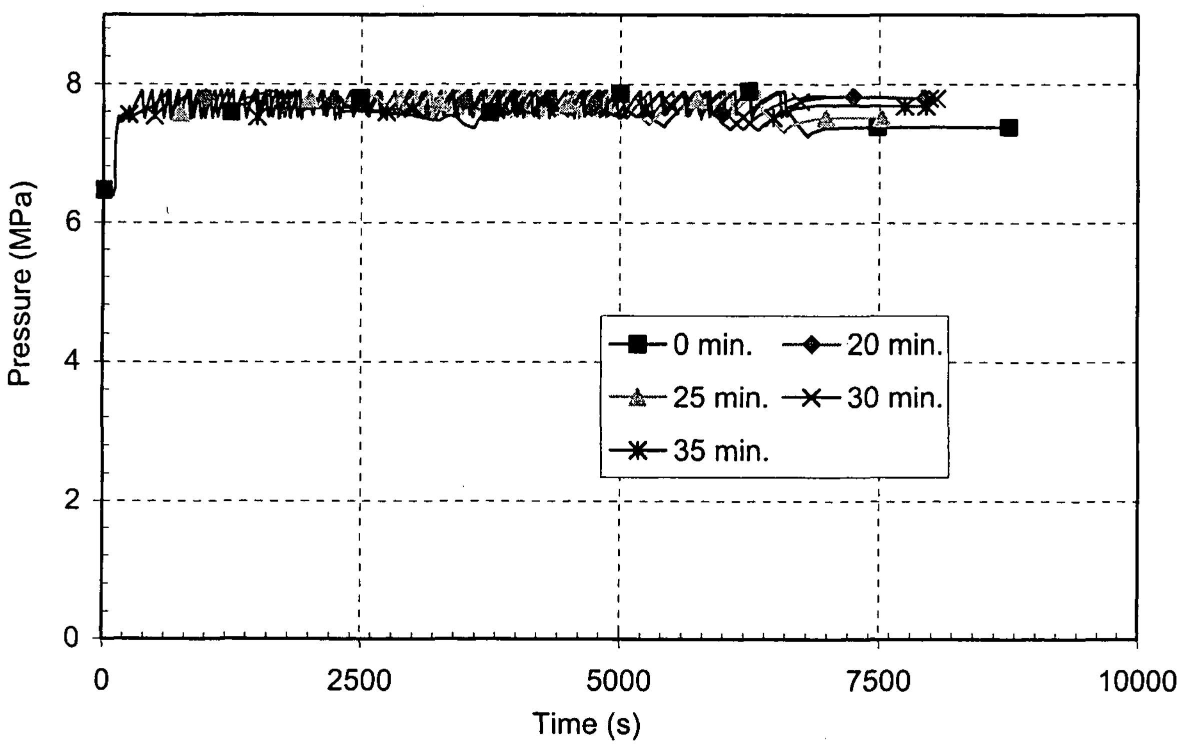

Figure 14 SG1 pressure for 2.54-cm break size LOCA with AFW start delays

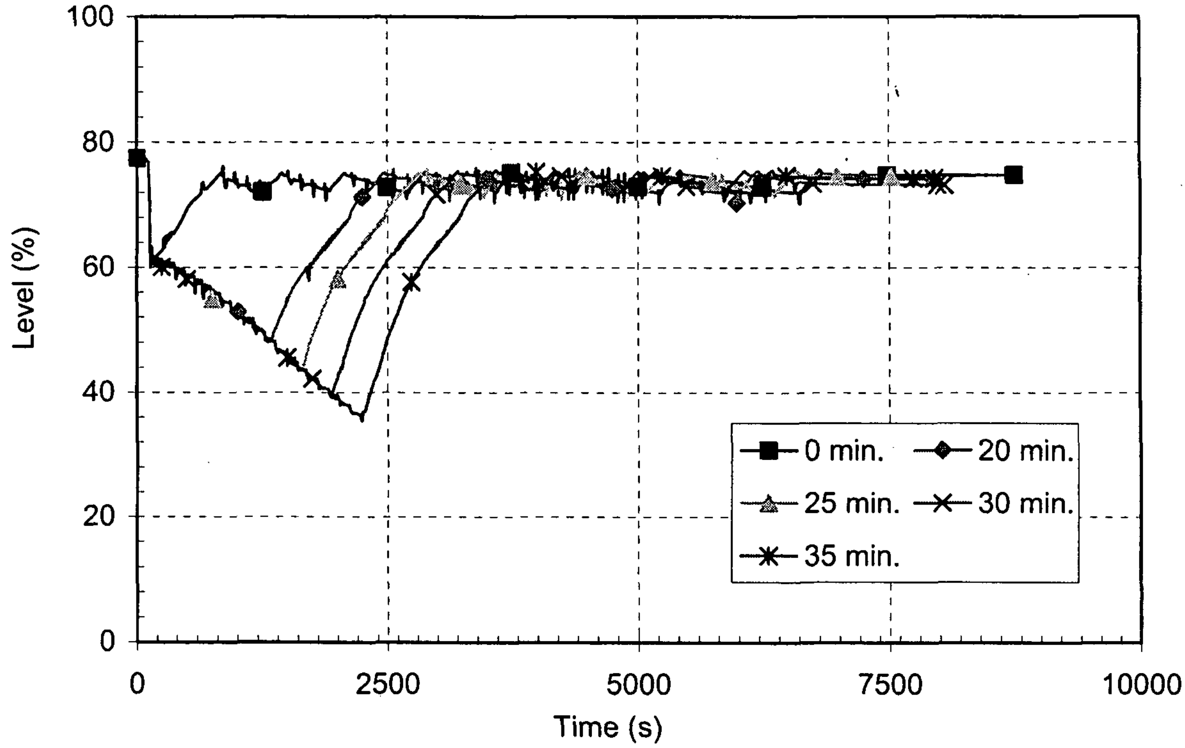

Figure 15 SG1 wide-range level for 2.54-cm break size LOCA with AFW start delays

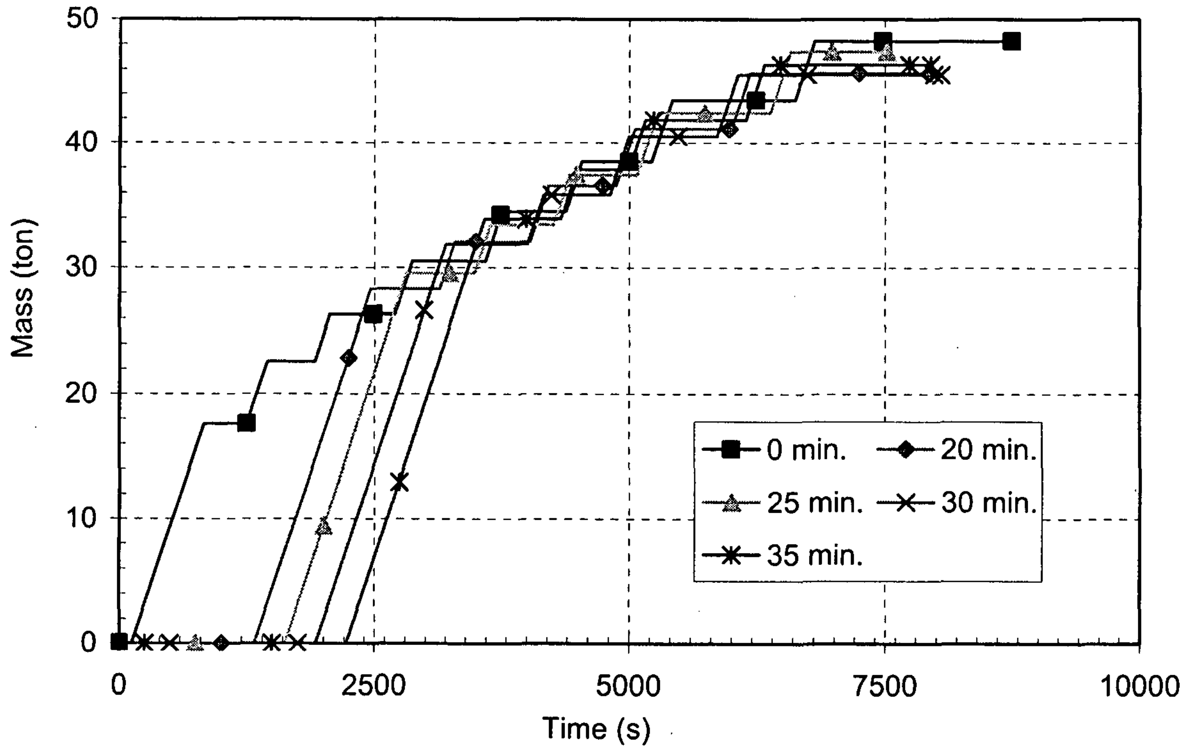

Figure 16 Integrated AFW1 flow for 2.54-cm break size LOCA with AFW start delays

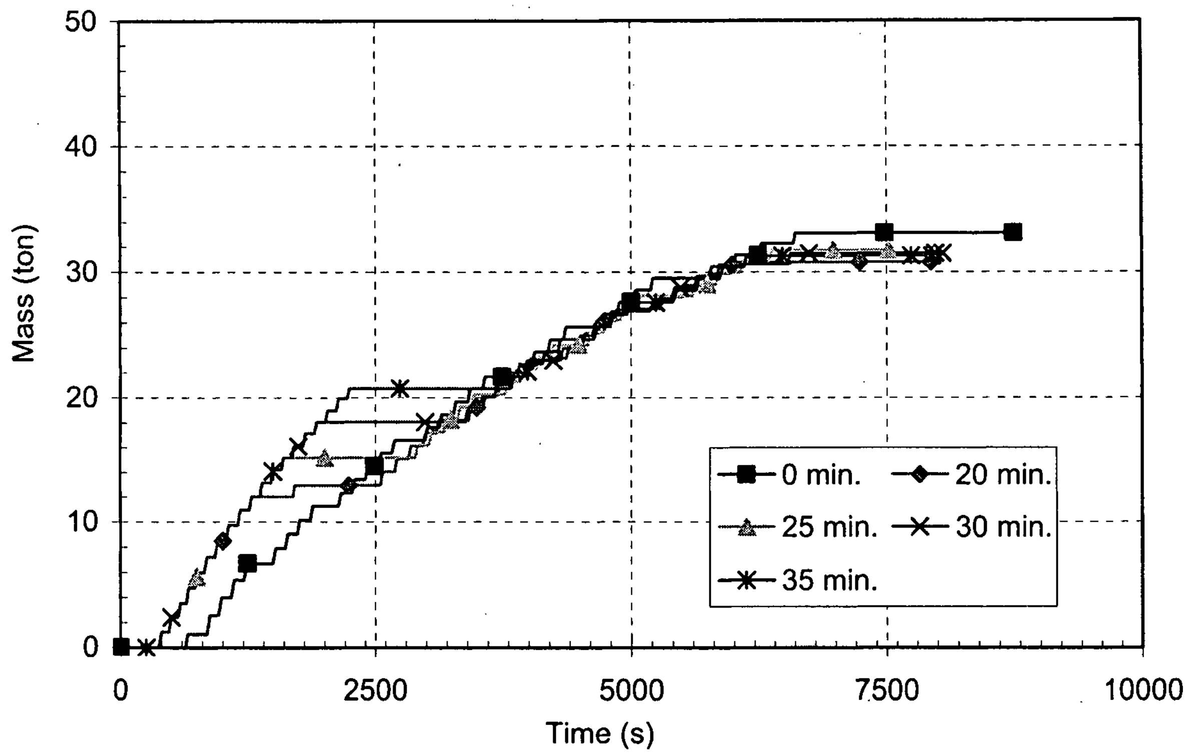

Figure 17 Mass discharged through SG1 PORV for 2.54-cm break size LOCA with AFW start delays

5.1.3 Calculations for a 2.54-cm Break Size Loss-of-Coolant Accident with Two Operator Actions

To determine the time window available to the operators to start AFW and open the SG PORV, different scenarios were analyzed for a 2.54-cm (1 in.) break, as shown in Table 1. Namely, the capacity of the AFW system is such that it fills the SG when the SG PORV is operated automatically and AFW injection is terminated. In such cases, the cooling (RCS depressurization) would be faster with the SG PORV fully opened manually to enable bleeding by the SG PORV and feeding by the AFW system. Figures 18 through 25 show that RCS depressurization with the SG PORV fully open is efficient in preventing core heatup, provided that the AFW pump maintains sufficient SG inventory. As shown in Figures 18 to 25 for Case A, immediate depressurization of the RCS with one SG PORV, without the AFW pump operating, results in emptying of the SG in 40 minutes. This means that the time delay for starting the AFW system upon manual opening of the SG PORV may be less then 40 minutes. The SG PORVs operate automatically in all cases. In Cases B through E, the SG level drops approximately linearly and cooling is sufficient as long as the SG is not emptied. This means that SG PORV must be opened before the SG is not completely emptied. In Cases A and F, the depressurization occurs in the already empty SG, and the core heatup is therefore unavoidable.

Figure 18 shows that the RCS is depressurized below accumulator injection in approximately 10 minutes after the SG PORV is manually opened. When accumulators start to inject, the RCS mass inventory recovers as shown in Figure 19; therefore the core is not uncovered as shown in

Figure 21, and core heatup is prevented (Figure 20). Figure 22 shows the SG1 pressure, which drops immediately because of the manual opening of the SG1 PORV. At the time of depressurization, the AFW pump No.1 starts to inject and the SG1 wide-range level starts to increase, as shown in Figure 23. Filling the empty SG with the SG PORV open takes almost 1 hour. Figure 24 shows the mass injected by the AFW system into SG1, while the other SG has no injection and stops emptying when the SG1 PORV is manually opened. Until that time, the trend is similar to the SG1 wide-range level.

Table 1 Operator Actions Delay

Case	Operator Action	
	SG PORV full opening delay (min)	AFW start delay (min)
A	0	Not available
B	30	30
C	50	50
D	80	80
E	100	100
F	120	120

In Case E, the heatup is very small, while in Case F, the temperature criterion is exceeded. The above results indicate that the operators have 100 minutes available to perform RCS depressurization. For the selected plant, the preferred path for RCS depressurization is SG steam dump valves and then SG PORVs. It is necessary to prevent loss of SG inventory by establishing AFW flow. The analysis shows that for RCS depressurization, manual operation of SG PORVs is needed in addition to automatic PORV operation. This also follows the severe accident management guidelines for the selected plant.

Operator experience with plant simulators shows that the actual time needed to perform the action is 1 to 10 minutes. Thus, the additional time available to perform the action is 90 to 99 minutes (i.e., the success criteria time minus the actual time to perform the event), which gives enough time for possible recovery action.

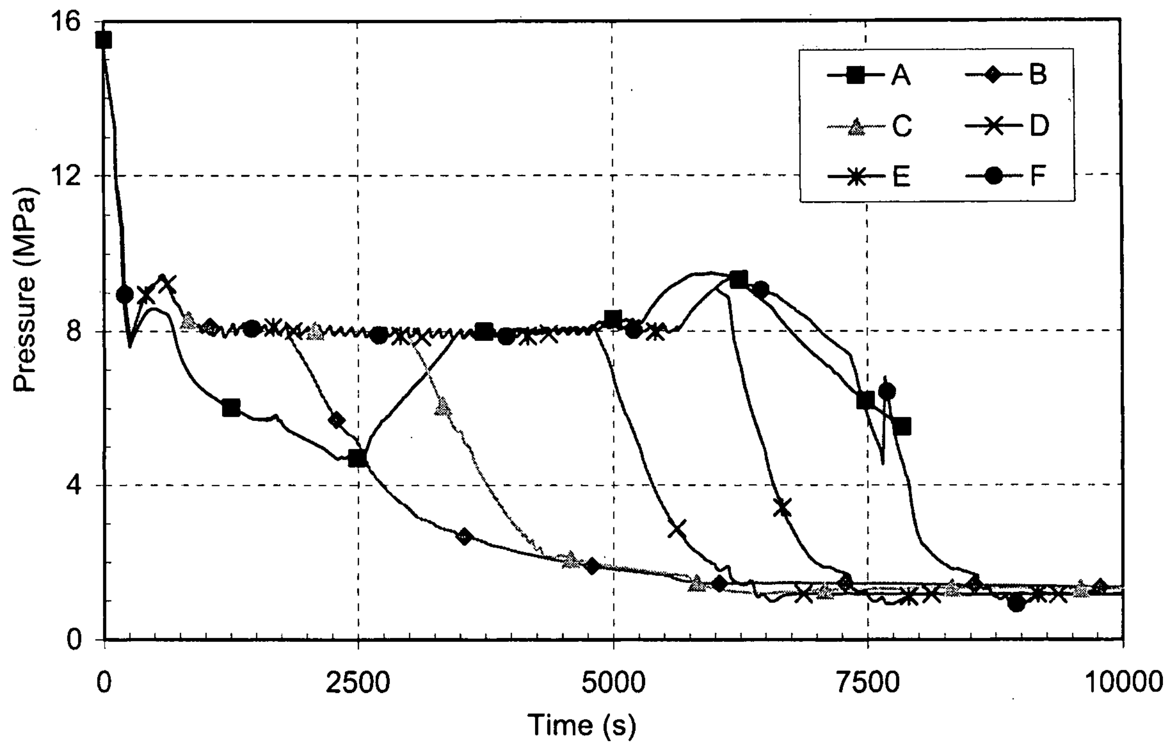

Figure 18 RCS pressure for 2.54-cm break size LOCA with manual opening of SG1 PORV

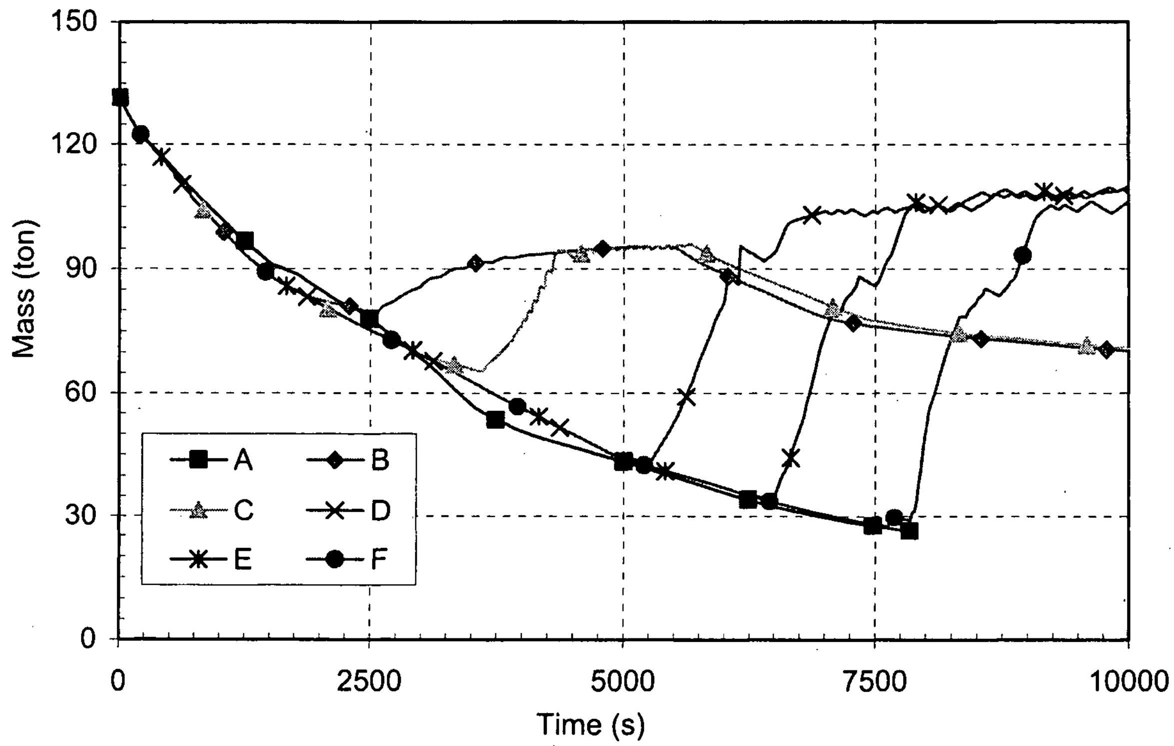

Figure 19 RCS mass inventory for 2.54-cm break size LOCA with manual opening of SG1 PORV

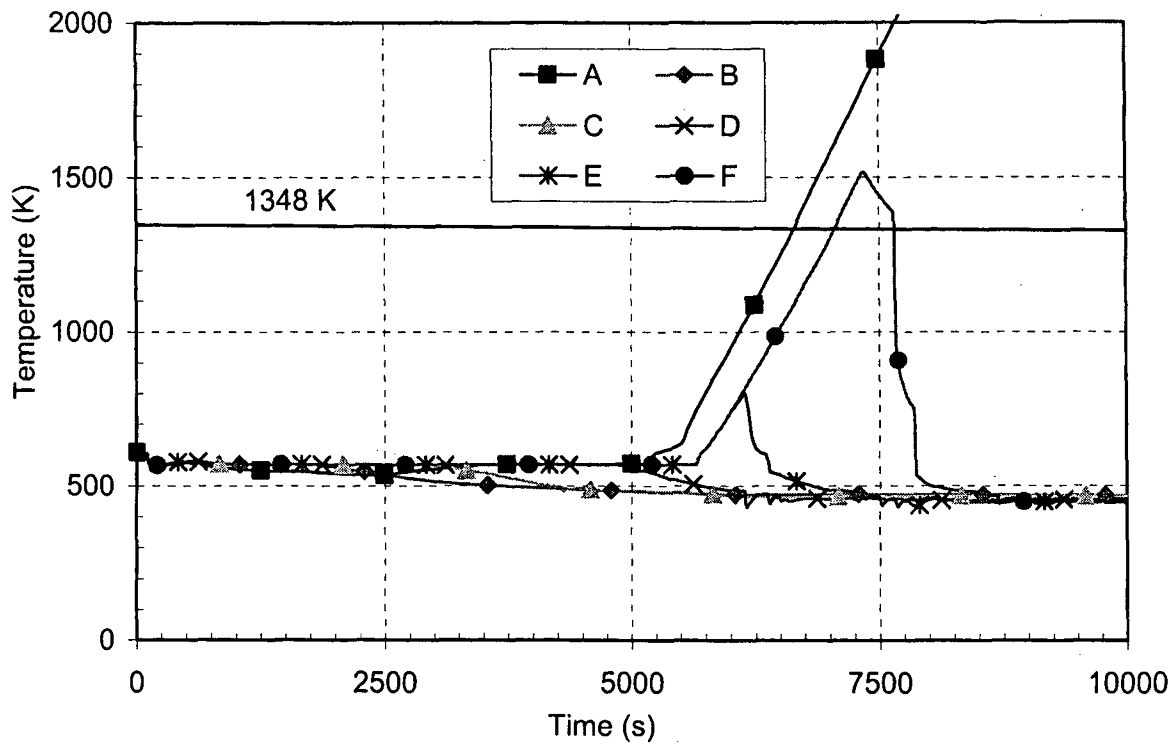

Figure 20 Core cladding temperature for 2.54-cm break size LOCA with manual opening of SG1 PORV

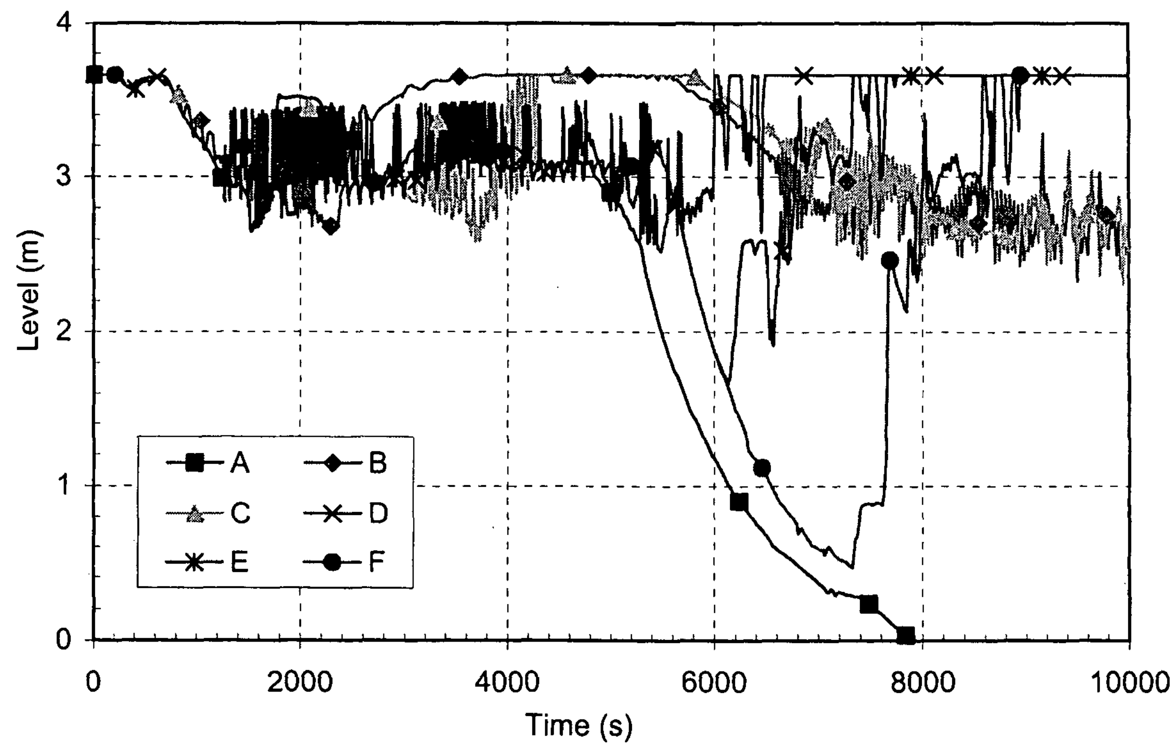

Figure 21 Core collapsed liquid level for 2.54-cm break size LOCA with manual opening of SG1 PORV

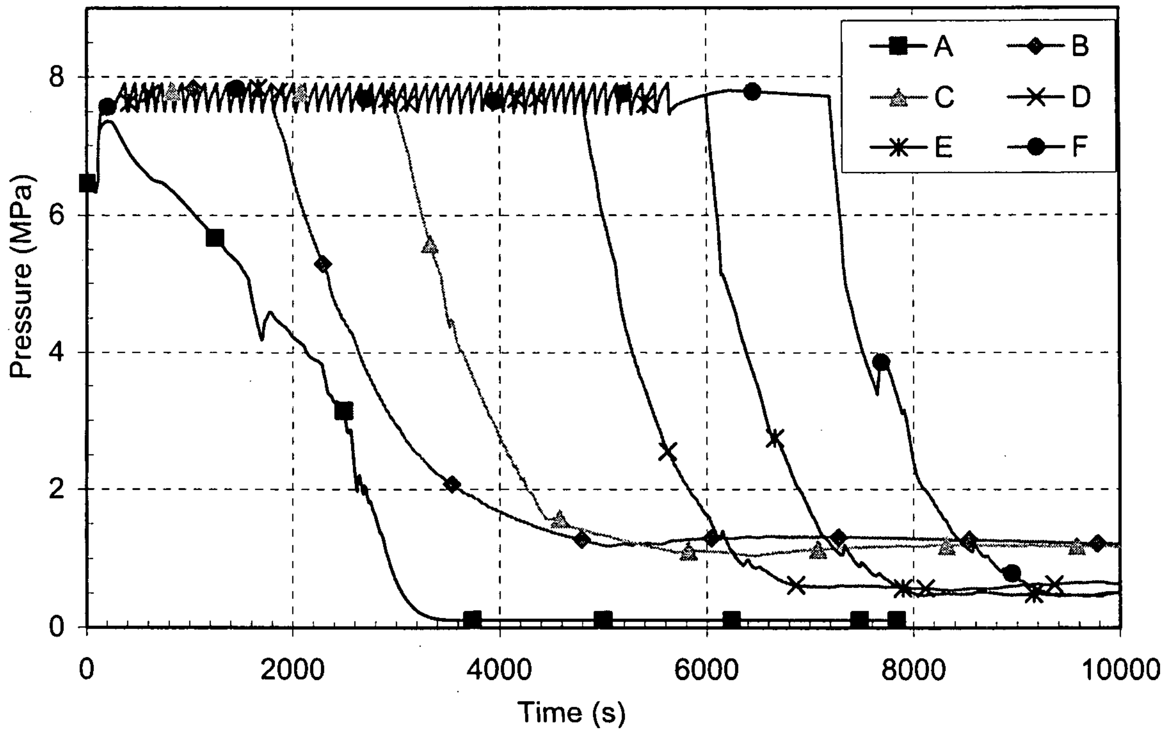

Figure 22 SG1 pressure for 2.54-cm break size LOCA with manual opening of SG1 PORV

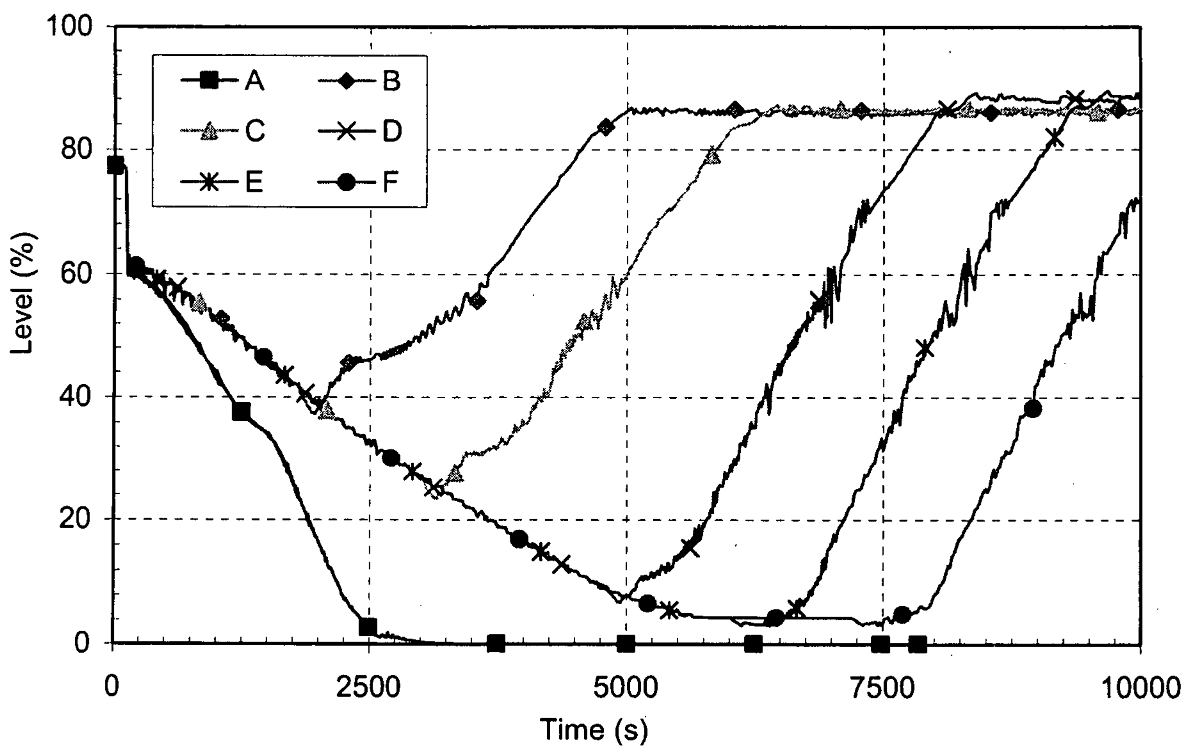

Figure 23 SG1 wide-range level for 2.54-cm break size LOCA with manual opening of SG1 PORV

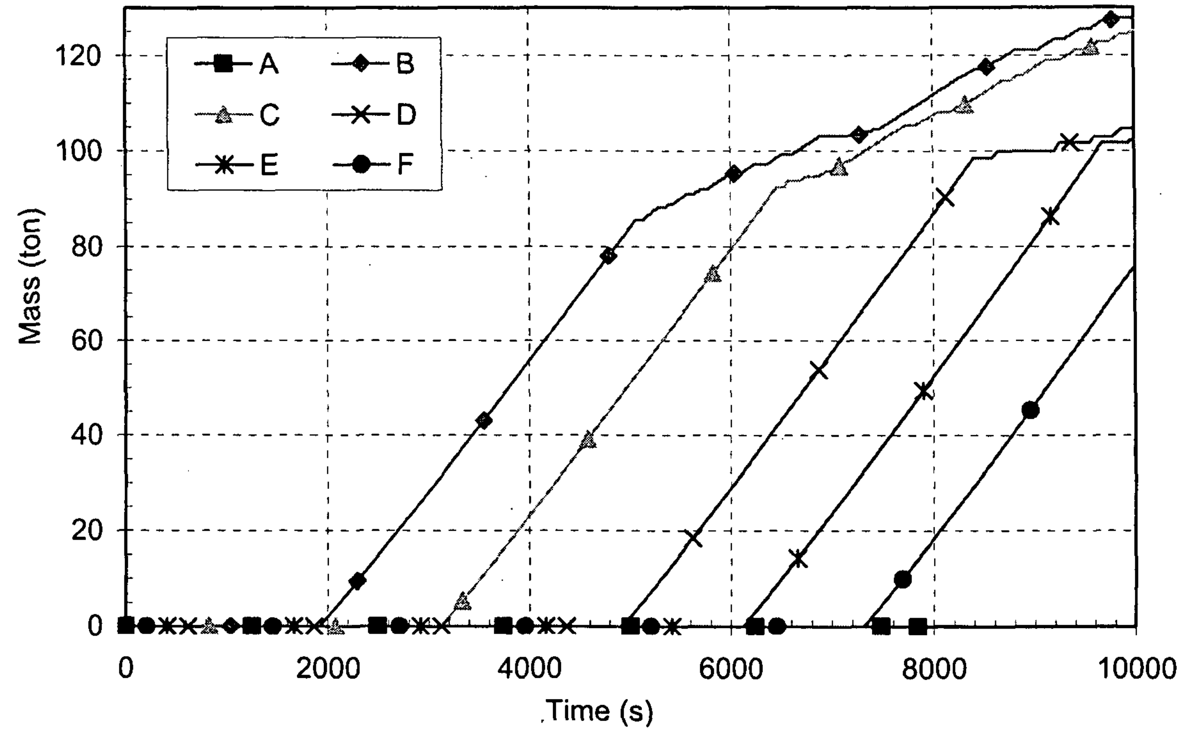

Figure 24 Integrated AFW1 flow for 2.54-cm break size LOCA with manual opening of SG1 PORV

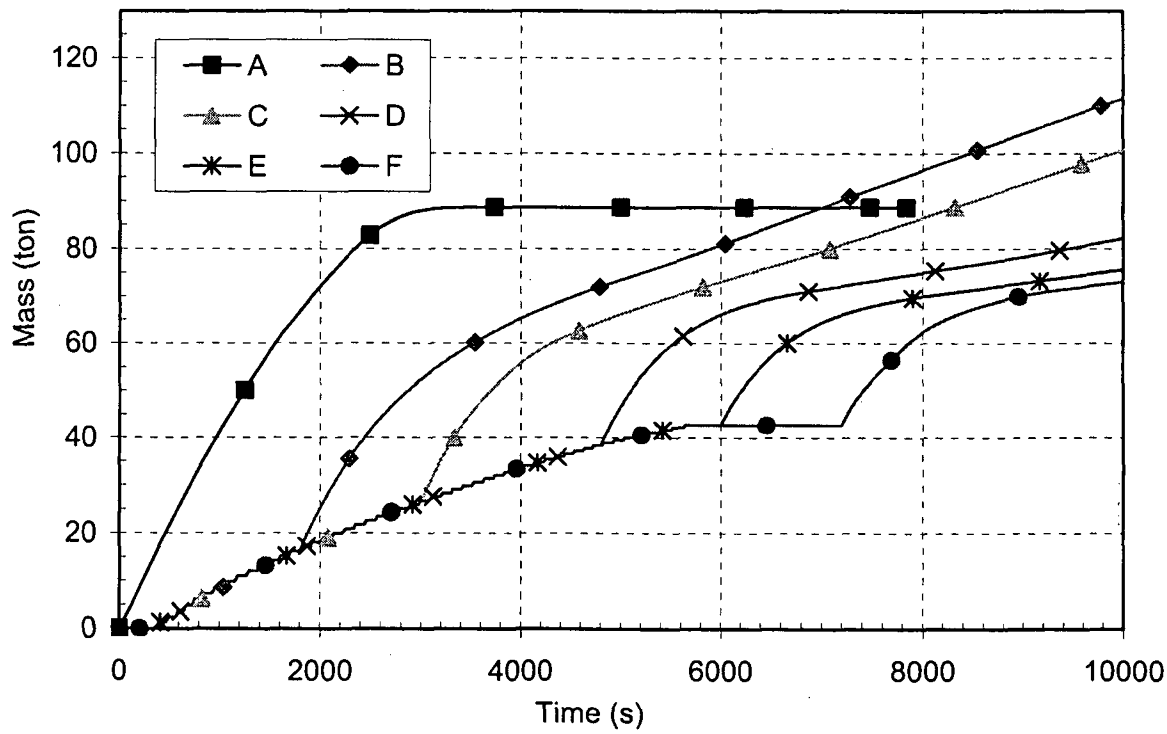

Figure 25 Mass discharged through SG1 PORV for 2.54-cm break size LOCA with manual opening of SG1 PORV

5.2 Loss of Feedwater Calculations with Manual Actuation of Auxiliary Feedwater

Table 2 and Figures 26 through 33 show the main results of LOFW calculations with manual actuation of the AFW system. Table 2 shows the sequence of main events. The focus of the calculations was to define the maximum time window for manually starting the AFW pump. The transient begins with the loss of MFW at time 0. Because of the loss of heat sink, the RCS average temperature starts to increase at 20 seconds and actuates the steam dump at 30 seconds. At 53 seconds, the reactor trips on the low-low SG level, which causes turbine trip. The RCS temperature drops to no-load value. When the AFW manual start is delayed 20 minutes, the SI signal is generated because of the low steamline pressure. The reason for low steamline pressure is that the SGs are almost empty after 10 minutes, which reduces the removal of stored and decay heat. Therefore, at 594 seconds, the RCS average temperature begins to increase, thus modulating open the steam dump valves. Because of the increased steam dump flow, the secondary-side pressure starts to decrease, which results in the generation of an SI signal because of low steamline pressure. The SI signal causes normal charging and letdown isolation and main steamline isolation. Because of main steamline isolation, the steam dump is lost. The HPSI pump starts to run upon generation of the SI signal. However, because of the high primary pressure, the HPSI pumps do not inject before the AFW pump starts, which very quickly enables cooling by the SG PORVs. The HPSI pumps are very efficient in recovering the RCS mass and pressure; therefore, they are stopped when the SI termination criteria are met. The RCPs are tripped when subcooling is lost, and with the HPSI pumps running, the criterion for tripping RCPs is fulfilled.

Table 2 Sequence of Main Events

Event	Time (s)					
Analyzed cases (AFW delay)	20 min	30 min	40 min	50 min	60 min	70 min
MFW closure	0.1	0.1	0.1	0.1	0.1	0.1
Reactor trip signal generation	52.9	52.9	52.9	52.9	52.9	52.9
Turbine trip	52.9	52.9	52.9	52.9	52.9	52.9
Steam dump discharge	30–617	30–617	30–617	30–617	30–617	30–617
SI signal generation	616.9	616.9	616.9	616.9	616.9	616.9
Letdown isolation	617.0	617.0	617.0	617.0	617.0	617.0
Steamline 1 and 2 isolation	617.0	617.0	617.0	617.0	617.0	617.0
RCP1 and 2 trip	NA	1587.2	1587.2	1587.2	1587.2	1587.2
AFW1 start (by assumption)	1205	1805	2405	3005	3605	4205
SG PORV first discharge	1275	1855	2460	3060	3660	4260
HPSI pump injection start	NA	2020	2560	3010	3630	4300
HPSI termination	NA	2450	4585	5594	6512	7372

Figures 26 through 33 show the important plant and safety variables that are factors in determining the time window. Parametric analyses were performed to get information how influences the delayed manual start of the AFW No. 1 pump on satisfying acceptance criteria described in Section 4.1.

Figure 26 shows that the RCS is not overpressurized. When one AFW pump starts to inject into the secondary side, cooling of the secondary side causes the pressurizer pressure to drop below the pressurizer PORV closure setpoint and then below the maximum pressure capacity of the HPSI pump. Figure 27 shows the RCS mass inventory. Depletion occurs because of the pressurizer PORV discharge, but HPSI pump injection efficiently recovers the RCS mass. When the RCS mass is depleted to approximately one-third, the core starts to heat up, as shown in Figure 28. The parametric analysis shows that the core heats up significantly when the AFW pump start is delayed more than 50 minutes. Figure 29 shows that the HPSI injected mass into the RCS, which is approximately balanced with the mass discharged through the pressurizer PORVs shown in Figure 30. The operator terminates SI when the criteria are met. Figures 31 through 33 show the secondary-side parameters for SG1, into which AFW is injected. Figure 31 shows the SG1 pressure. At turbine trip, the pressure initially increases and then starts to slowly drop during steam dump operation. On SI signal generation at 617 seconds, the pressure again increases to the SG1 PORV setpoint and then oscillates because of SG1 PORV cycling until the flow of AFW is started. Figure 32 shows the SG1 wide-range level. The level starts to increase when the AFW flow is established. Figure 33 shows the mass released in the SG1 PORV cycling.

The maximum available time to start the AFW pump according to the success criteria is 60 minutes. When action is taken faster, benefits are evident. Based on simulator experience (Ref. 7), the operator needs from 1 to 10 minutes to start the AFW system.

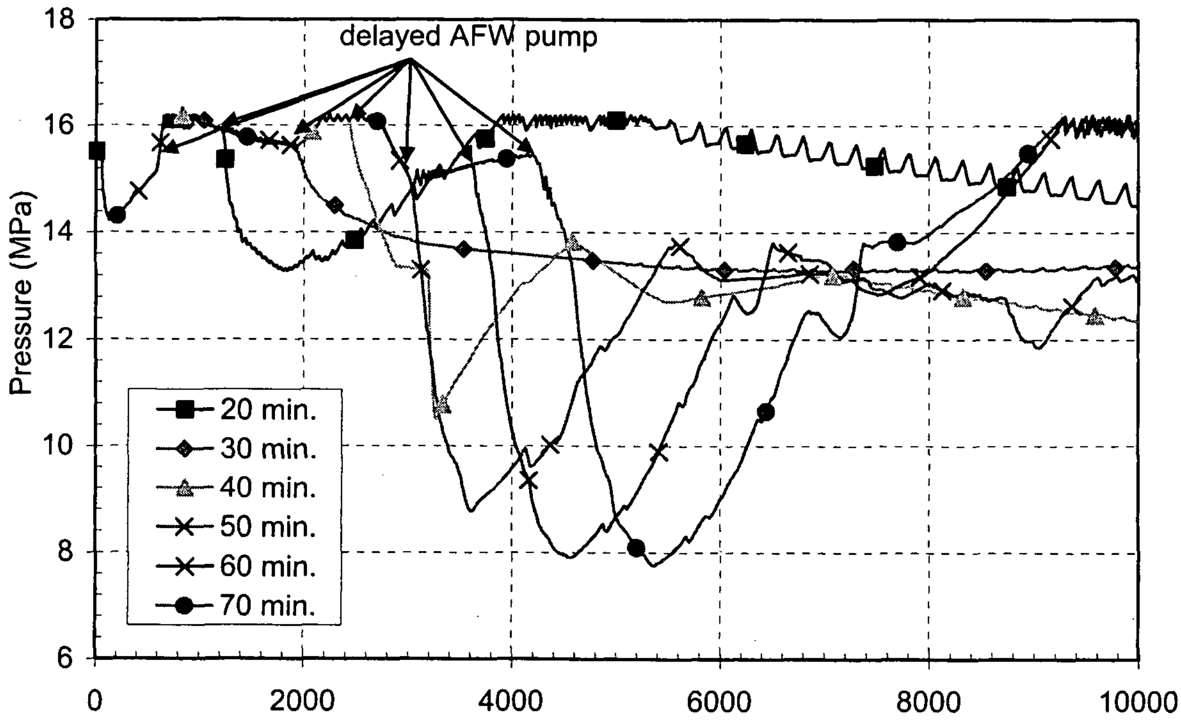

Figure 26 Pressurizer pressure for LOFW with manual actuation of AFW

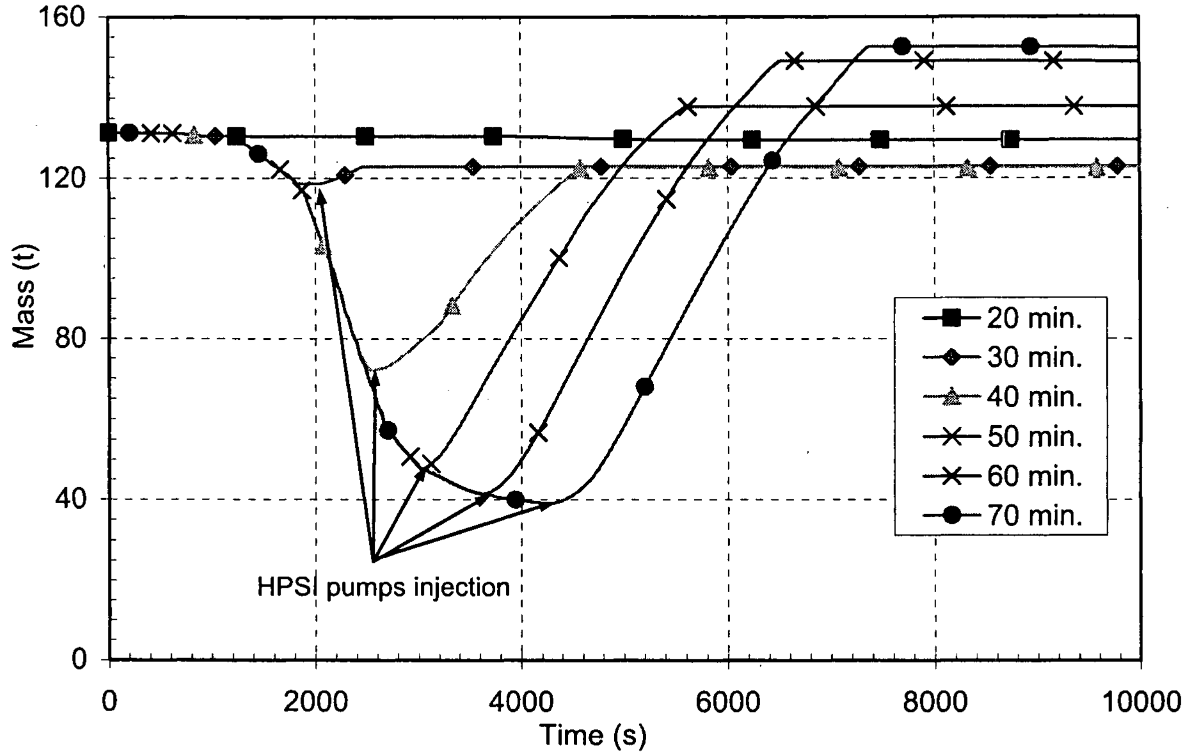

Figure 27 RCS mass inventory for LOFW with manual actuation of AFW

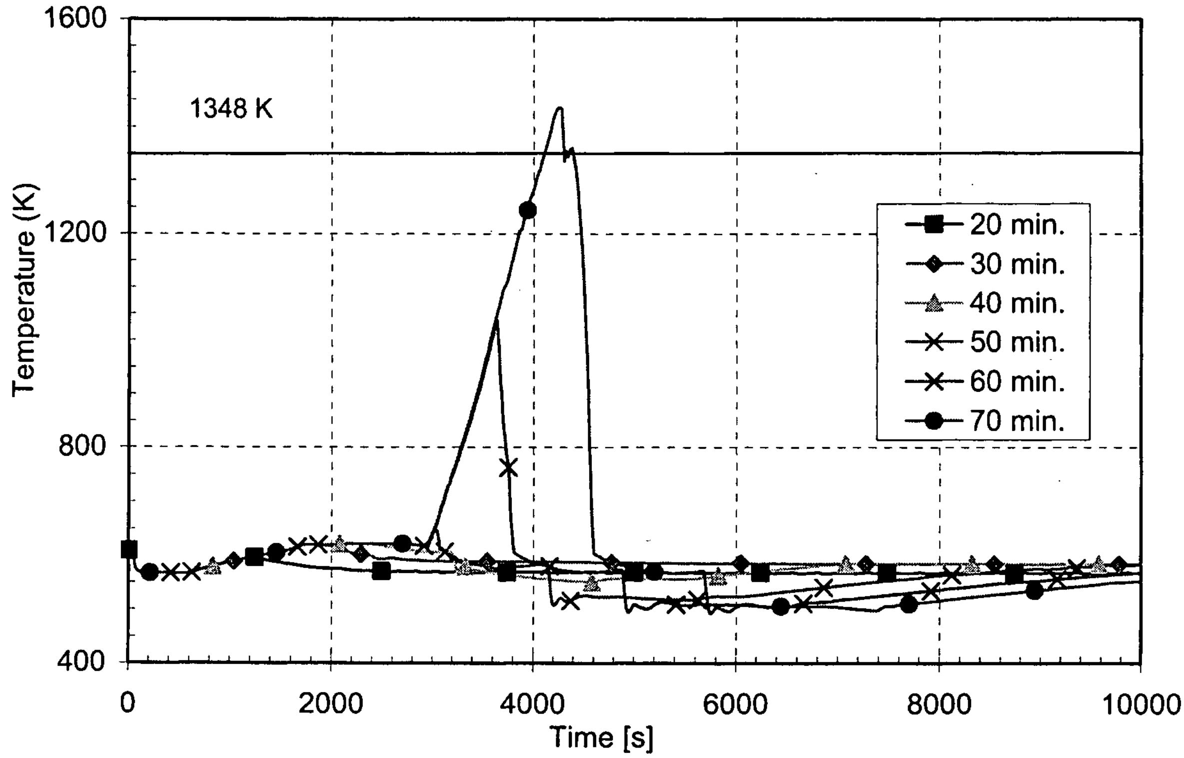

Figure 28 Cladding temperature at 11/12 height of the core for LOFW with manual actuation of AFW

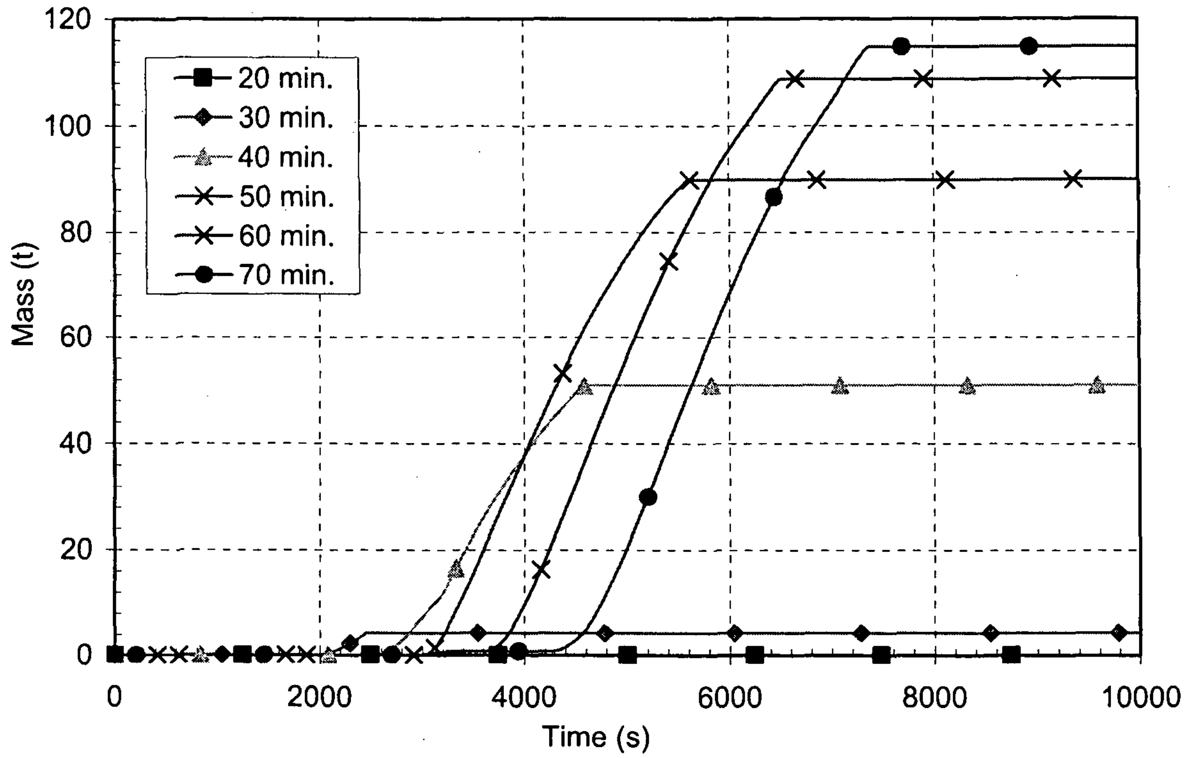

Figure 29 Integrated HPSI flow for LOFW with manual actuation of AFW

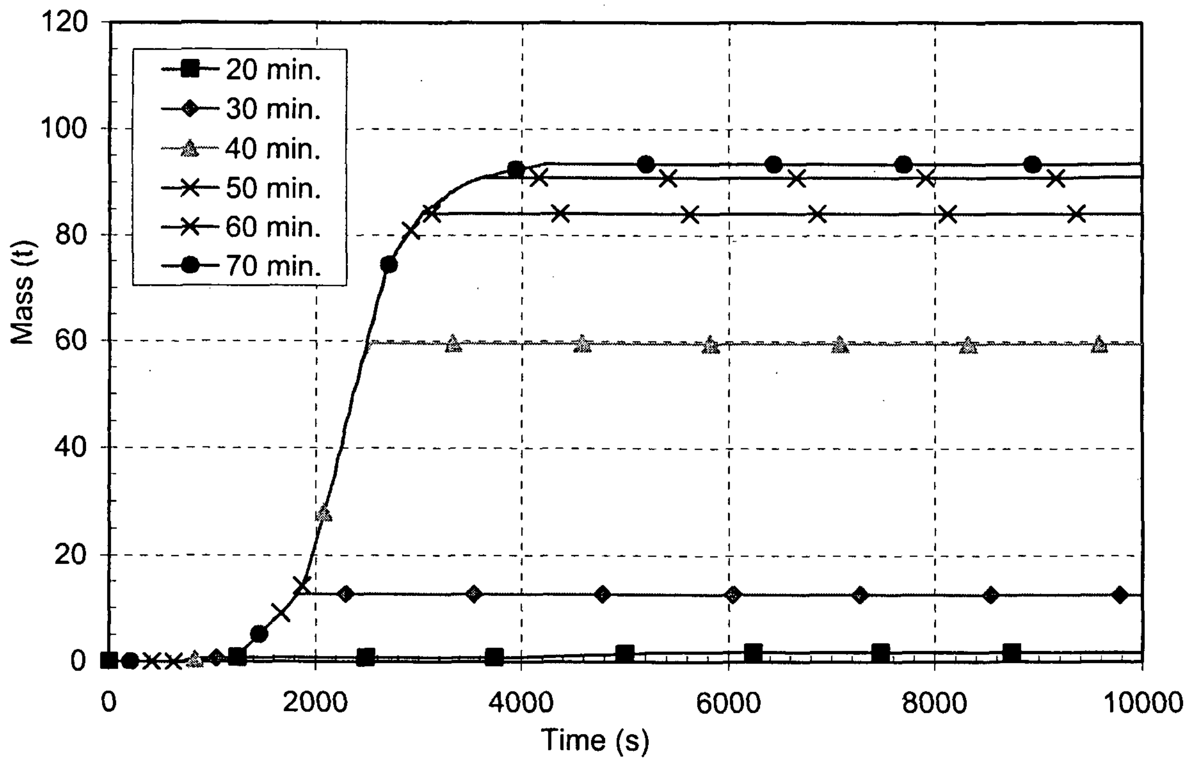

Figure 30 Integrated pressurizer PORVs flow for LOFW with manual actuation of AFW

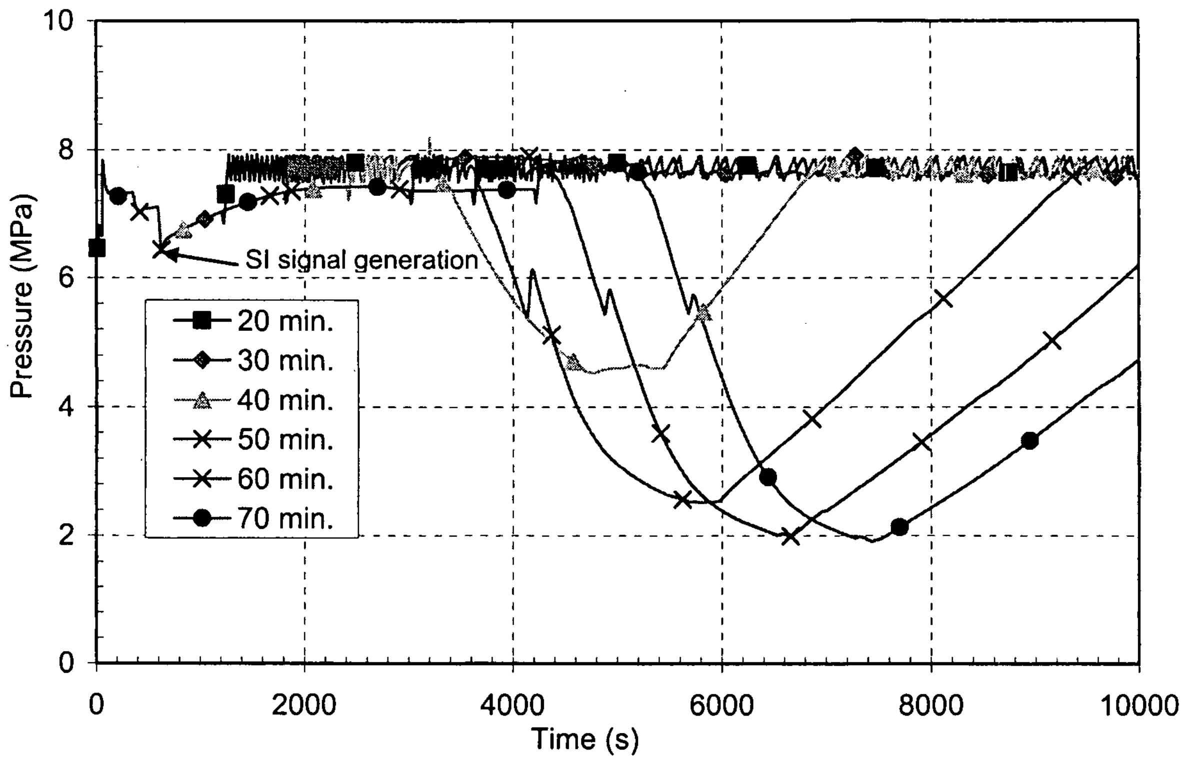

Figure 31 SG1 pressure for LOFW with manual actuation of AFW

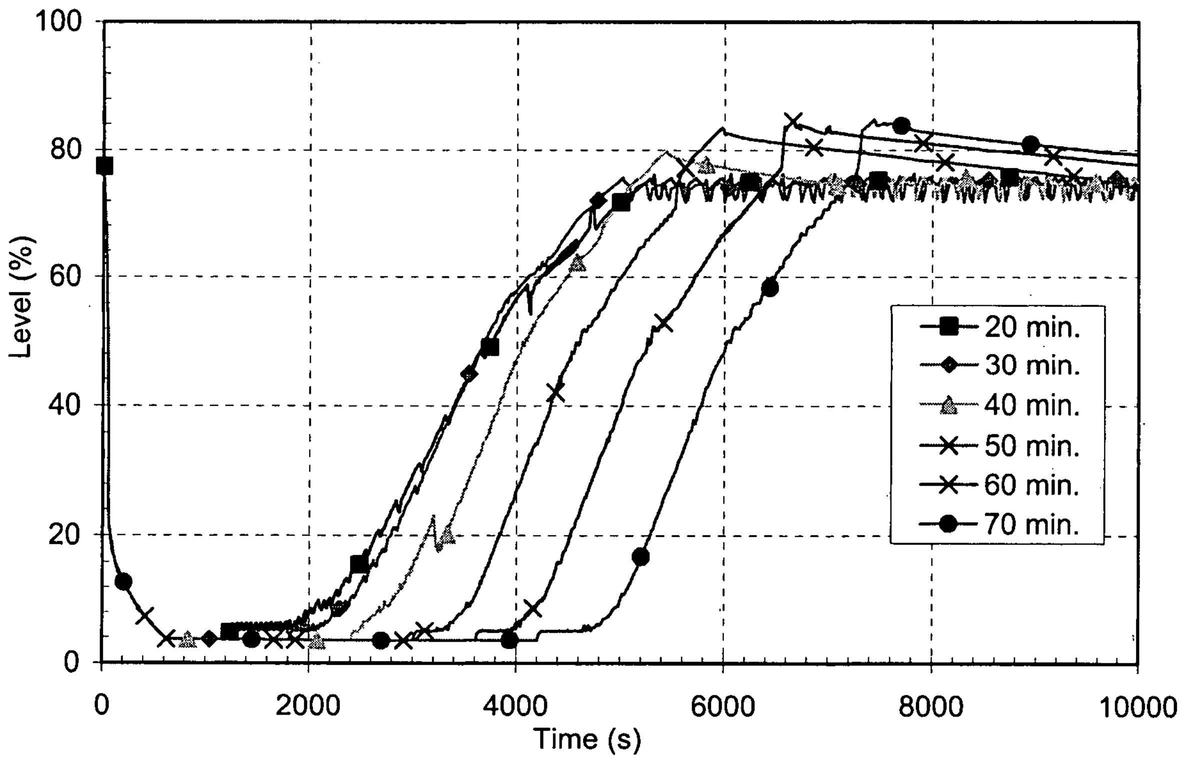

Figure 32 SG1 wide-range level for LOFW with manual actuation of AFW

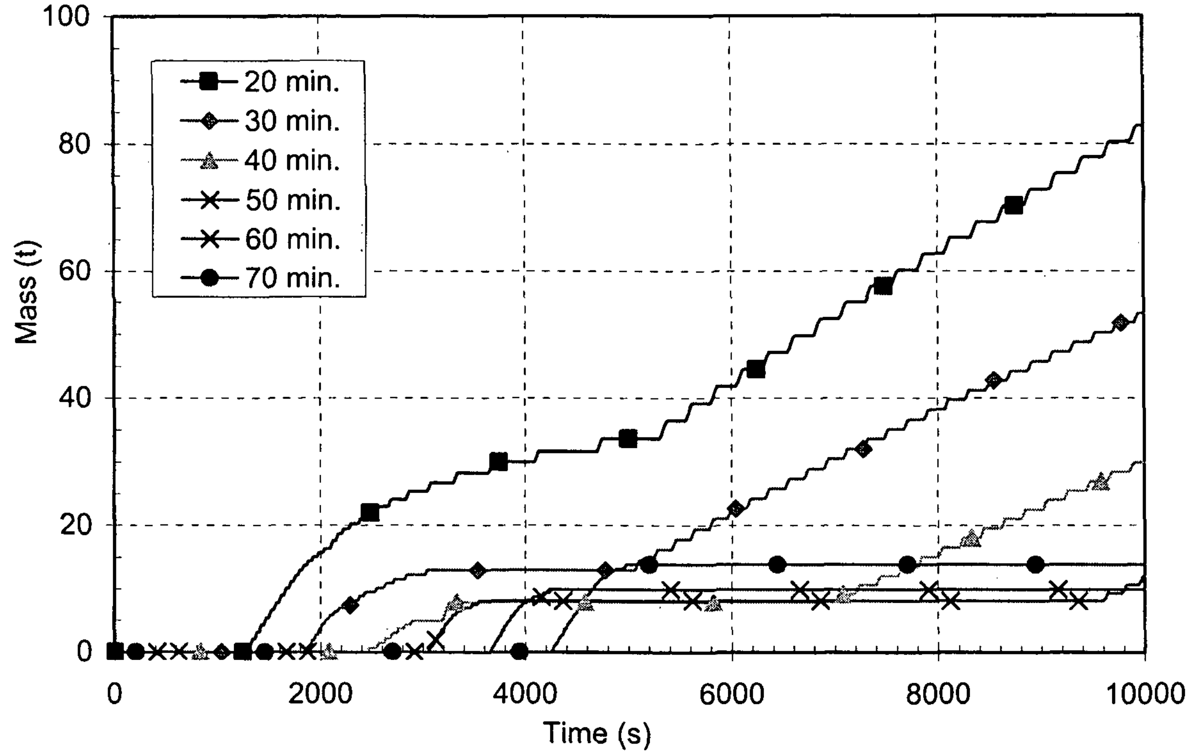

Figure 33 Integrated SG1 PORV flow for LOFW with manual actuation of AFW

5.3 Calculations of Loss-of-Coolant Accidents with Manual Actuation of Safety Injection

Figures 34 and 35 show the results of LOCA calculations with manual actuation of SI. At breaks smaller than 5.08 cm, the RCS was not sufficiently depressurized (Figure 34) to enable accumulator injection, while larger breaks depressurize the RCS. Figure 35 shows that the temperature criterion 1,348 K is first exceeded for a break of 15.24 cm (Case 6"), then for a break of 10.16 cm (Case 4"), 7.62 cm (Case 3"), 1.91 cm (Case 0.75"), and finally for 5.08 cm (Case 2"). This is because for the 5.08-cm (2-in.) break, the accumulators are sufficient to cool the core until they are empty. For breaks larger than 5.08 cm (2 in.), the core begins to significantly heat up after the accumulators empty. In general, the larger the break, the faster the core uncovers. For the 15.24-cm (6-in.) break, the core starts to heat up at 20 minutes. For the 5.08-cm (2-in.) break, the core cladding temperature could exceed the criterion at first peak, if uncertainty is considered. When the SI signal is actuated 20 minutes, further core heatup is prevented (Case 6" SI). This is also true in the case of the 5.08-cm break (Case 2" SI). Therefore, at least 20 minutes are available for operator action. In this scenario, the treatment of uncertainty is unnecessary because the time window is the shortest for the largest break in the spectrum.

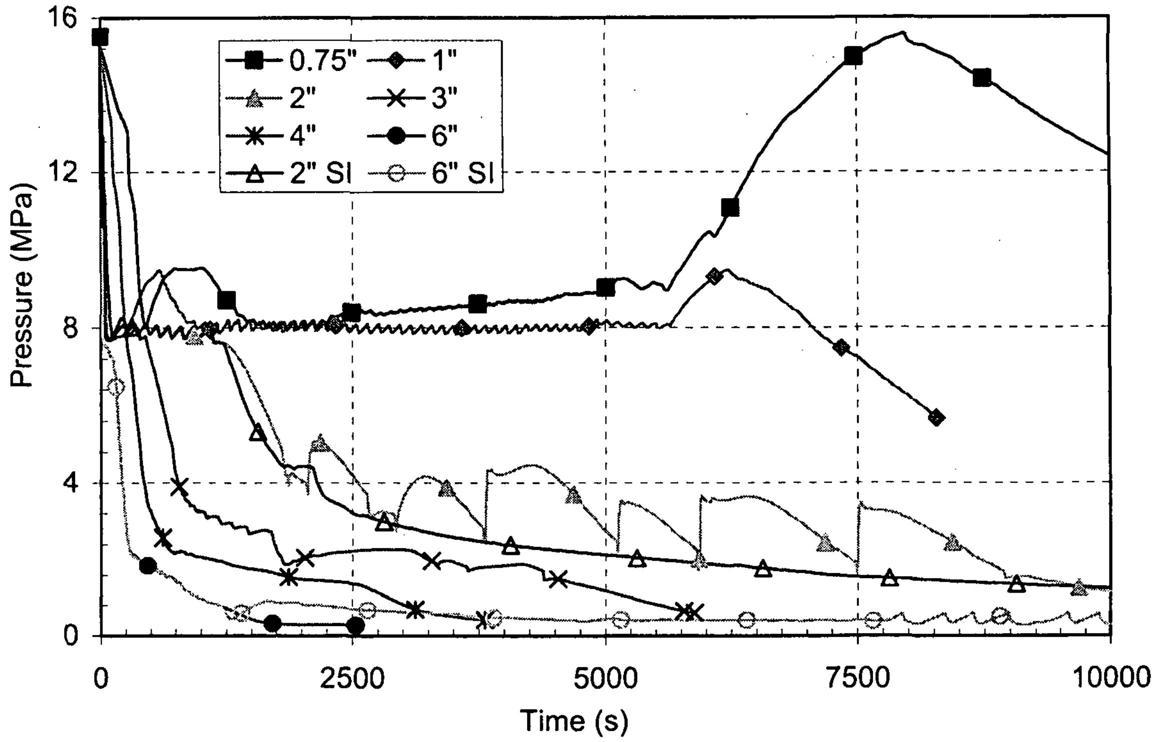

Figure 34 Pressurizer pressure for LOCA with manual actuation of SI

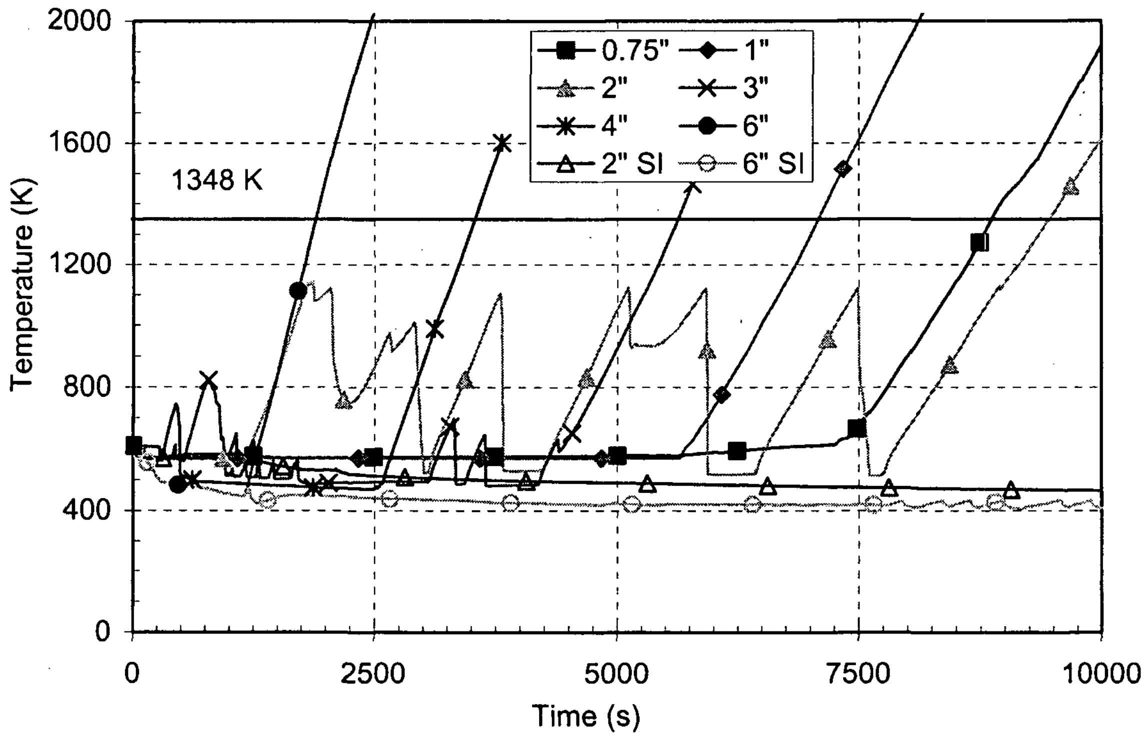

Figure 35 Cladding temperature at 11/12 height of the core for LOCA with manual actuation of SI

5.4 Probabilistic Safety Assessment Results

Institut "Jožef Stefan" – HRA (IJS-HRA) method (Refs. 8, 9) assumes that if the difference between the time window in which the action has to be performed and the actual time needed to perform the action is 10 minutes or more, a recovery can be modeled for the investigated action. If the additional available time for action is shorter than the determined time interval, recovery is not considered.

Additional available time for action (T_a) is defined as the difference between the time window of the action (T_w) and the actual time needed to perform the action (T_p), which is assessed based on real simulator scenarios:

$T_a = T_w - T_p$.

The time window of the human action actually represents the success criteria for the action. It represents the time interval in which operators must perform the action so that the plant can be put in a safer state (i.e., the plant is put into a scenario that leads to a safe state and not to an accident state). The actual time needed to perform the action is the realistic time required for an operator to perform the action, which can be obtained from simulator experience.

The specified time windows are important for HRA because the human error probability (HEP) of a certain operator action is lower if operators have more time available. In the control room of an NPP a team of operators works under the direction of a shift supervisor. If operators have 10 or more minutes of additional time for action, it can be expected that colleagues or the shift supervisor can observe and correct a possible error. Consideration of recovery causes lower HEP and may result in a different impact of human error on the overall PSA results.

5.4.1 Model Description

The PSA model of an NPP is named as HRA_IH_1 and is used for quantification. The characteristics of HRA_IH_1 show that it is a large and detailed model, which includes 4,748 gates; 1,810 basic events; 16 initiating events and main event trees; 738 fault trees, which include 125 human failure fault trees; 57 parameters (failure rate); 418 parameters (probability), which include 55 parameters connected with HEP (those 55 parameters are obtained from 18 different basic HEP parameters, which are expanded to 55 parameters considering different performance shaping factors for basic HEP parameters); 18 groups for parameters of human errors; and 117 groups for human error basic events. Table 3 shows parameters for selected human actions, which are needed for the decision to consider recovery when quantification of HEP is made.

Table 3 Parameters for Selected Human Errors

Human Error	T_w (min)	T_p (min)	T_a=T_w-T_p (min)
Manual actuation of AFW at LOCA	100	1–10	90–99
Manual actuation of AFW at LOFW	60	1–10	50–59
Manual actuation of SI at LOCA	20	2	18

5.4.2 Base Case Results

The results of the PSA include many parameters. Only selected results are mentioned below for the analysis with the following features:

- consideration of internal initiating events
- third-order approximation
- truncation of 2.7×10^{-11}/reactor-year (RY) (Ref. 10)
- consideration of recovery for all selected human actions, because additional available time for action (i.e., the difference between the time in which operators must perform the action so that it meets the success criteria and the actual time needed to perform the action) is more than the determined time interval (e.g., 10 minutes)

The results include the following:

- Core damage frequency of 2.487×10^{-5}/RY.
- No minimal cut set includes manual actuation of AFW during a LOCA. A minimal cut set is a combination of basic events (i.e., component failures, human errors) that may cause an undesired state of the system (e.g., an accident state). This means that manual actuation of AFW during a LOCA is not a safety-significant event as it is not involved in any combination of undesired events.
- Minimal cut set No. 4 (ranked by contribution to core damage frequency) contributes to core damage frequency by 7.136×10^{-7}/RY. It is the highest contributing minimal cut set of those that involve manual actuation of AFW during transients. This means that manual actuation of AFW during transients is a very safety-significant event.
- Minimal cut set No. 1358 (ranked by contribution to core damage frequency) contributes to core damage frequency by 1.088×10^{-9}/RY. This is the highest contributing minimal cut set of those that involve manual actuation of SI during LOCAs. This means that manual actuation of SI during LOCAs is not a very safety-significant event.
- Table 4 presents risk importance factors (i.e., the fractional contribution of considered human errors). The table shows that manual actuation of AFW during LOFW contributes significantly to the core damage frequency, as indicated by the high fractional contribution. The manual actuation of AFW in case of LOCA is not in the list of minimal cut sets, so the risk importance factor cannot be calculated (the event is of no safety significance).

5.4.3 Sensitivity Results of Selected Examples

Sensitivity analysis is performed for each of the selected example actions for a case if recovery would not be considered in the quantification of HEP (i.e., if additional available time for action would be less than the determined time interval). Table 5 shows the results for selected human errors without consideration of recovery.

Results show that consideration of recovery has a significant impact on the HEP. This is evident from a comparison of basic HEPs in Table 4 (in which recovery is considered) and Table 5 (in which recovery is not considered). The change of HEP can significantly impact the core damage frequency and thus the plant risk, if the affected human error is an important contributor to risk, as is the case with manual actuation of AFW in case of transients. For an important human error, it is necessary to determine additional time for action accurately as this may have a significant impact on the assessment of risk.

A comparison of results in Table 4 and Table 5 shows that consideration of recovery leads to a significant change in risk results. For manual actuation of AFW in a LOCA, the change is insignificant, which is expected as the event is not risk important and thus its changes do not significantly affect the results. For manual actuation of AFW during transients (LOFW), this change is significant, as it nearly doubles the core damage frequency and thus the level of risk. For manual actuation of SI in case of a LOCA, the change in risk results is also insignificant.

Table 4 Probabilistic Safety Assessment Results If Recovery Action Considered

Human Error	Basic HEP	Fractional Contribution	Core Damage Frequency	Main Minimal Cut Set and Its Contribution	
Manual actuation of AFW in LOCA	$2.31x10^{-4}$	N/A	$2.487x10^{-5}$ /RY	-	-
Manual actuation of AFW in LOFW	$2.31x10^{-4}$	$6.93x10^{-2}$	$2.487x10^{-5}$ /RY	4	$7.136x10^{-7}$ /RY
Manual actuation of SI in LOCA	$3.99x10^{-5}$	$3.01x10^{-4}$	$2.487x10^{-5}$ /RY	1358	$1.088x10^{-9}$ /RY

Table 5 Probabilistic Safety Assessment Results If Recovery Action Not Considered

Human Error	Basic HEP	Fractional Contribution	Core Damage Frequency	Main Minimal Cut Set and Its Contribution	
Manual actuation of AFW in LOCA	$2.85x10^{-3}$	$7.71x10^{-4}$	$2.494x10^{-5}$ /RY	7875	$8.424x10^{-11}$ /RY
Manual actuation of AFW in LOFW	$2.85x10^{-3}$	$4.80x10^{-1}$	$4.448x10^{-5}$ /RY	1	$8.810x10^{-6}$ /RY
Manual actuation of SI in LOCA	$4.92x10^{-3}$	$3.91x10^{-3}$	$2.496x10^{-5}$ /RY	190	$1.343x10^{-8}$ [/RY]

5.5 Results

The times needed for operators to perform actions were determined on the basis of simulator experience (Ref. 7). To start the AFW system, the operator needs from 1 to 10 minutes, while SI signal actuation requires 2 minutes. When the time window is large, much additional time is available, and the time window does not need to be determined very accurately, even if the human factor event is an important contributor to the risk. For example, the time needed to start the SI signal is 2 minutes, so operators have an additional 18 minutes to perform this action. Considering uncertainties in peak cladding temperatures of 200 K based on previous uncertainty evaluations (Ref. 11) and the adiabatic heatup rate for a 15.24-cm (6-in.) break, the criterion would be reached 3 minutes earlier than in case with not considering uncertainties. Equally important is the uncertainty in the time of reaching maximum temperature, which is approximately 2 minutes (Ref. 12). Even considering the uncertainties, the time window is sufficient.

In the case of small- and medium-break LOCAs with the assumption that HPSI is not available, depressurization is needed for breaks smaller than 5.08 cm (2 in.). The 5.08-cm (2-in.) break is limiting, as for this and larger breaks, the RCS depressurizes by itself. However, when the pressure drops below the accumulator injection point, the core is already heated up in the case of a 5.08-cm (2-in.) break. Considering the typical cladding temperature uncertainty of the best estimate calculation to be 200 K (Ref. 11), the criterion 1,348 K could be exceeded. The possibility of recovery action would then be questionable because of the short time window. The uncertainty analysis is not necessary, as the contribution of this event to plant risk is insignificant.

Establishing AFW at an LOFW event is a significant contributor to the risk, but the calculated time window gives sufficient additional time, even if the HRA uses a conservative time window.

For the case of a LOCA with delayed SI signal actuation, the analysis shows that the additional time available is sufficient. Therefore, uncertainty analysis is not needed even though the event is a contributor to risk.

All these examples show that uncertainty analysis is not needed, because additional time is available and/or the event is not a significant contributor to risk, as determined by the PSA. This finding indicates that uncertainty analysis may be valuable only for significant risk contributors when additional available time is close to the time interval (e.g., 10 minutes) after which recovery would not be considered. For the selected examples, this is not the case. When the additional available time is not so close to the time after which recovery is not considered, the uncertainty of an operator's action can be estimated in the PSA work scope by considering conservative time windows as proposed in Reference 13.

6. RUN STATISTICS

The scenarios were calculated on a Hewlett-Packard personal computer with Intel Core 2 Quad at 2.40 gigahertz under Microsoft Windows XP, Professional Version 2002, Service Pack 3. Table 6 shows the run statistics. For all calculations, the number of volumes was 469. In most cases, the calculations run faster than real time. The exception are the 2" calculations, where the reactor kinetics time step was reduced below the minimum value, and thus the time step was set to small value. Steady-state calculations for all runs lasted 1,000 seconds and required 307.4 seconds of CPU and 32,531 steps. Compared to RELAP5/MOD3.3 steady-state calculations made in 2005 on a SUN FIRE V880 server (with four UltraSPARC III 750-megahertz processors, with 16 gigabytes main RAM, running under the SOLARIS 9 operating system) and requiring 1669.9 seconds, the current calculations ran more than five times faster.

Table 6 Run Statistics

Calculated Case	CPU Time (s)	Transient Time (s)	CPU/Transient Time	Number of Time Steps
	LOCA calculations with manual actuation of manual AFW			
A	1214.2	7840*	0.15	111965
B	2192.8	10000	0.22	261194
C	1826.4	10000	0.18	240354
D	1294.0	10000	0.13	214544
E	990.4	10000	0.10	197818
F	691.7	10000	0.07	180859
	LOFW calculations with manual actuation of AFW			
20 min	3541.0	10000	0.35	323210
30 min	2274.4	10000	0.23	264772
40 min	3299.6	10000	0.33	283196
50 min	3182.7	10000	0.32	310860
60 min	4683.4	10000	0.47	456030
70 min	3331.3	10000	0.33	309882
	LOCA calculations with manual actuation of SI			
0.75"	1645.0	10000	0.16	169956
1"	1014.9	8276*	0.12	125794
2"	24422.3	10000	2.44	2333657
3"	3328.4	5884*	0.57	330534
4"	2522.1	3813*	0.66	264064
6"	1763.0	2532*	0.70	213871
2" SI	64653.6	10000	6.47	6153483
6" SI	5997.3	10000	0.60	625187

* Calculation with clad temperature exceeding criterion

7. CONCLUSIONS

In this study, deterministic safety analyses with RELAP5/MOD3.3 Patch 03 computer code were performed as a support to the HRA. Safety analyses were needed to determine the time parameters, which were inputs for the HRA within the probabilistic safety assessment. The deterministic analyses and results were presented on selected realistic examples representative of typical situations. The results of HRA show that consideration of recovery has a significant impact on human error probability. Changes in HEP can significantly impact the core damage frequency if the affected human error is an important contributor to risk, as it is in one of the three example actions. For important human errors, it is necessary to determine the additional time for action accurately, as this may have a major effect on the assessment of risk. As implied by the HRA, the less time available, the more probable human error becomes.

This study also shows that uncertainty evaluation of the best-estimate calculation is not needed in the cases presented, even though one event is a significant contributor to the risk, because the available time is much greater than the time needed to perform the operator action.

For the LOCA case with manual AFW start, the evaluation of uncertainties would significantly change the time window. The reason is that a larger break size is critical, as it implies faster evolution of the transient and less time for the operator to act. The uncertainty analysis is not needed for the HRA, as the contribution of this event to plant risk is insignificant.

The results suggest that uncertainty analysis of realistic deterministic safety analysis in support of HRA may be needed only for significant risk contributors, when the additional available time for action is close to the time limit for considering the possibility of recovery.

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11. ABSTRACT *(200 words or less)*

This report presents the results of analyses performed for the updated human reliability analysis. The analysis estimates time windows available to perform operator action to satisfy the success criteria to prevent core damage. The best-estimate RELAP5/MOD3.3 computer code was used. In the past, the conventional probabilistic safety assessment used a conservative approach to address this factor. However, the current standard for probabilistic safety assessment recommends the use of best-estimate codes. The RELAP5/MOD3.3 best-estimate code calculations were performed for three selected cases in which human actions supplement safety system actuations: (1) small or medium loss-of-coolant accident requiring a manual start of the auxiliary feedwater system, (2) loss of normal feedwater requiring a manual start of the auxiliary feedwater system, and (3) a loss-of-coolant accident requiring manual actuation of the safety injection signal. The analysis used a qualified RELAP5 input model representing a Westinghouse-type, two-loop pressurized water reactor for the calculations. The results of the deterministic safety analysis were examined to identify the latest time that an operator could act and still satisfy the safety criteria. The results show that the time available to perform operator action (i.e., the time window) is greater than the actual time needed to perform the action.

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