MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

May 31, 2010

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-10148

Subject: **Transmittal of Markups of US-APWR DCD Chapter 15.6.5 Small Break LOCA**

Reference: 1) Mitsubishi Heavy Industries, Ltd., 'MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/9/2010', UAP-HF-10138, May 21, 2010

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") an official document entitled 'Markups of US-APWR DCD Chapter 15.6.5 Small Break LOCA'. MHI provides this document in relation to the M-RELAP5 code modification described in Reference 1, as agreed between the NRC and MHI.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata
General Manager - APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

DO81
NRO

Enclosures:

1. Markups of US-APWR DCD Chapter 15.6.5 Small Break LOCA (non-proprietary)

CC: J. A. Ciocco
C. K. Paulson

Contact Information
C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373 – 6466

ENCLOSURE 1

UAP-HF-10148

Markups of US-APWR DCD Chapter 15.6.5 Small Break LOCA

May 2010
(Non-Proprietary)

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

Loss-of-coolant accidents (LOCAs) are postulated accidents (PAs) that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system. The coolant loss occurs from piping breaks in the reactor coolant pressure boundary (RCPB) up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (RCS).

Various size breaks were examined to determine the conditions of the RCS, reactor core, and containment vessel and to demonstrate that the emergency core cooling system (ECCS) has the capability to mitigate each LOCA. For the US-APWR, the spectrum of breaks is categorized under large break and small break LOCAs, for the purposes of reporting bounding results. A large break is defined as a break with a total cross-sectional area equal to or greater than 1.0 ft^2. A small break is defined as a piping break within the RCPB with a total cross sectional area up to 1.0 ft^2.

The small break LOCA reported in this section is a large enough break that the charging pumps of the chemical and volume control system (CVCS) cannot provide sufficient makeup water to the RCS; therefore, the ECCS would be actuated. For very small breaks where the charging pumps have the capability to make up for leakage, the pressurizer level and pressure would be sustained and the ECCS would not be actuated.

In the transient and accident analyses for the US-APWR, both large break and small break LOCAs are classified as the PAs. They are not expected to occur during the life of the plant, but postulated as a conservative design basis. The event frequency conditions are described in Section 15.0.0.1.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Description of Large Break LOCA

The pipe break for the large break LOCA is assumed to occur in a cold leg piping located between the outlet of the reactor coolant pump (RCP) and the corresponding reactor vessel (RV) inlet nozzle, as this break places the most severe performance requirement on the ECCS. The double-ended cold leg guillotine (DECLG) and split breaks, with a total cross-sectional area equal to or greater than 1.0 ft^2, are analyzed. The RCS loop taken for the break is the one with pressurizer on it.

In this large break LOCA analysis, loss-of-offsite power (LOOP) is assumed. The LOOP occurs coincident with the break. The primary effect on this transient is that AC power will be lost to the RCPs and they will coastdown. The LOOP scenario is more severe as core then sprays it into the containment vessel to maintain the pressure of the containment to be below the design pressure and restore it to approximately atmospheric pressure. The CSS is automatically actuated on the high-3 containment pressure signal. The CS/RHR heat exchangers provide long term cooling by removing heat from the containment to further reduce the pressure.

During a LOCA, the RWSP is well protected against debris wash down. Containment drains (transfer pipes) into the RWSP are protected from large debris by vertical debris bars, capped by a ceiling plate. The suction strainers, and the CSS and SI suctions are located as such that they are protected from clogging. Detailed design descriptions are given in Section 6.2.2.2.

Continued operation of the SI pumps supplies borated water during long term cooling. Core temperatures are reduced to long term, steady state levels associated with the dissipation of residual heat generation. During long term cooling, the HHIS is designed to inject into both the RCS hot legs and the reactor vessel downcomer to avoid an unacceptably high concentration of boric acid (H_3BO_3) in the core.

15.6.5.2.2 Description of Small Break LOCA

The small break LOCA is assumed primarily to occur in a cold leg piping located between the outlet of the RCP and the corresponding RV inlet nozzle, as this break places the most severe performance requirement on the ECCS. The DECLG, split and the direct vessel injection (DVI) line breaks, with a total cross sectional area up to 1.0 ft^2 are analyzed. The RCS loop taken for the DECLG and split breaks is the one with pressurizer on it.

In this small break LOCA analysis, LOOP is assumed to occur in concurrent with the reactor trip. The LOOP scenario is more severe as core flow decreases earlier and the SI pumps start later than in the offsite power available scenario. Because the LOOP cases are more severe, only those results are reported in this document.

Compared with the large break, the phases of the small break LOCA prior to recovery occur over a longer time period. In order to identify various phenomena, the small break LOCA can be divided into five phases: blowdown, natural circulation, loop seal clearance, boil-off, and core recovery. The duration of each phase depends on the break size and the performance of the ECCS. The following discussion of these five phases assumes the small break is located at the cold leg. The phases during small break LOCA can be described as follows:

<u>Blowdown phase</u>

Upon initiation of the break, the RCS primary side rapidly depressurizes until flashing of the hot coolant into steam begins. Reactor trip is initiated on the low pressurizer pressure setpoint of 1860 psia. Closure of the condenser steam dump valves isolates the SG secondary side. As a result, the SG secondary side pressure rises to the safety valve set point of 1296 psia, and steam is released through the safety valves. The ECCS actuation signal is generated at the time the pressurizer pressure decreases to the low pressurizer pressure setpoint of 1760 psia and safety injection initiates, after a time delay. The RCPs trip, after 3 seconds delay, upon the reactor trip signal resulting from the low pressurizer pressure, because the LOOP is assumed for the safety analysis. The coolant in the RCS remains in the liquid phase throughout most of the blowdown period, although toward the end of the period, steam begins to form in the upper head, upper plenum, and hot legs. The rapid depressurization ends when the pressure falls to just above the saturation pressure of the SG secondary side, which is at the safety valve set point. The break flow in the RCS is single-phase liquid throughout the blowdown period.

Natural Circulation phase

When the blowdown phase ends, two-phase natural circulation is established in the RCS loops with the decay heat being removed by heat transfer (condensation and convection) to the SG secondary side. The EFW is initiated to maintain the secondary side inventory. As more coolant is lost from the RCS through the break, steam accumulates in the downhill side of the SG tubes and the crossover leg. The natural circulation phase will continue until there is insufficient driving head on the cold leg side of the loops, due to the accumulation of steam in loops between the top of the steam generator tubes and the loop seals.

Loop Seal Clearance phase

The third phase is the loop seal clearance period. With the loop seals present, the break remains covered with water. The RCS water inventory continues to decrease and steam volume in the RCS increases. The relative pressure in the core increases, which, together with the loss of coolant inventory through the break, causes the liquid levels in the core and the SG to continue to decrease. If, during this process, the core mixture level drops below the top of the core, the cladding will experience a dryout and the cladding temperature in the upper part of the core will begin to rise. When the liquid level of the downhill side of the SG is depressed to the elevation of the loop seals, the seals clear and steam in the RCS is vented to the cold legs. Break flow changes from a low-quality mixture to primarily steam. This relieves the back-pressure in the core and the core liquid level is restored to the cold leg elevation by flow from the downcomer.

Boil-off phase

After the loop seals clear, the RCS primary side pressure falls below that of the secondary side due to the increase of the break flow quality, resulting in a lower mass flowrate but a higher volumetric flow through the break. The vessel mixture level may decrease as a result of the core boiling in this phase, if the RCS pressure is too high for the injection system to make up for the boil-off rate. The core might uncover before the RCS depressurizes to the point where the SI pumps (and accumulator, when the RCS pressure drops to a sufficiently low value) deliver ECCS water to the RCS at a rate higher than the break flow.

Core Recovery phase

As the RCS pressure continues to fall, the combined SI and the accumulator flowrates eventually exceed the break flow. The vessel mass inventory increases and the core recovery is established. In a small break LOCA, the accumulator injection to the core begins before the reactor coolant is completely discharged into the containment vessel, and the RCS pressure is still above the containment pressure. For a small break LOCA, the PCT occurs when the core is at a relatively high pressure, and the break flow is choked. Therefore, the containment pressure in the small break LOCA does not affect the PCT.

TMI action item II.K.3.5 "Automatic RCP Trip during a LOCA" requires RCP trip following all small breaks. In the US-APWR, an automatic RCP trip will actuate on an ECCS actuation signal generated from low pressurizer pressure, or high containment pressure. When the offsite-power is available, the RCPs automatically trip after the ECCS actuation signal. In the case of LOOP, the RCPs trip after the 3-second delay following LOOP which is postulated to occur concurrently with the reactor trip for the safety analysis. Hence, the requirement is met. No operator action is required to trip the RCPs during a LOCA.

In the small break LOCA, the RCS pressure may not fall below the pressure that allows water injection from the accumulators. In this case, the HHIS alone provides the core cooling function. Continued operation of the SI pumps supplies borated-water during long term cooling. Core temperatures are reduced to long term, steady state levels associated with the dissipation of residual heat generation.

15.6.5.2.3 Description of Post-LOCA Long Term Cooling

There are two considerations in the post-LOCA long term cooling that must be addressed: maintaining long term decay heat removal and the potential for boric acid (H_3BO_3) precipitation. After the quenching of the core at the end of reflood phase, continued operation of the ECCS supplies borated water from the RWSP to remove decay heat and to keep the core subcritical. Borated water from the RWSP is initially injected through DVI lines (RV injection mode). If left uncontrolled, boric acid (H_3BO_3) concentration in the core may increase due to boiling and reach the precipitation concentration. Boric acid precipitation in the core could affect the core cooling. To prevent the boric acid precipitation, the operator switches over the operating DVI lines to the hot leg injection line (simultaneous RV and hot leg injection mode).

In the case of a hot leg break, almost all ECCS water injected through DVI lines passes through the core and exits from the break point. As a result, the boric acid concentration in the core does not increase. Even after the switchover, sufficient ECCS water passing through the core for decay heat removal is assured, and that simultaneously prevents any increase in boric concentration in the core.

In the case of a cold leg break, the ECCS water through DVI lines is not effective in flushing the core. As the result, boric acid concentration in the core may increase. After the switchover, almost all ECCS water injected into the hot leg passes the core. Therefore, the boric acid concentration in the core decreases.

The main objective of the post LOCA long term cooling evaluation is to determine the switchover time from RV injection mode to the simultaneous RV and hot leg injection mode to prevent the boric acid precipitation, hence the long-term cooling is assured.

15.6.5.3 Core and System Performance

15.6.5.3.1 Evaluation Model

The reactor is designed to withstand thermal effects caused by a LOCA event including the double-ended severance of the largest RCS pipe. The reactor core and internals together with the ECCS are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core is preserved following the accident. The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the requirements of 10 CFR 50.46. The requirements are:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

b. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

d. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In this best-estimate large break LOCA analysis, the analysis method and inputs are identified and assessed to estimate the uncertainty of the calculated results. This uncertainty is accounted for, in order to obtain a high probability that the criteria (a) through (c) above are not exceeded.

15.6.5.3.1.1 Large Break LOCA Evaluation Model

<u>Large Break LOCA Calculation Methodology</u>

The 10 CFR 50.46 permits the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. In particular, best estimate thermal-hydraulic models may be used to predict the peak cladding temperature (PCT), local maximum cladding oxidation (LMO), and maximum core wide cladding oxidation (CWO). The regulation requires an assessment of the uncertainty of the best estimate

initial conditions, power distributions, and global and local parameters for the DECLG break. Some bounding parameters are fixed and selected to obtain a conservative estimate of PCT. One such parameter is the containment pressure, which affects the PCT and contains an uncertainty, as described in Chapter 6, Section 6.2.

Applying the Wilks' equation (Ref. 15.6-26), it needs 59 ASTRUM runs to obtain the 95th percentile for one parameter (i.e. PCT) with 95% confidence. The number of runs (N) for three parameters (i.e., PCT, LMO and CWO) with 95th percentile and 95% confidence is 124, obtained using the following equation:

$$\beta \leq 1-\sum_{k=0}^{2} {}_{N}C_{k}\alpha^{N-k}(1-\alpha)^{k}$$

where: α = 0.95 (95th percentile) β= 0.95 (95% confidence), k is the number of evaluation parameter, and N is the number of runs. The detail procedure to yield the 124 runs is described in the Topical Report (Ref.15.6-9) and Reference 15.6-15.

Applying ASTRUM to calculate the total uncertainty in the PCT and other parameters, all the uncertainty parameters are sampled simultaneously in random in the WCOBRA/TRAC runs. Local parameters are those that affect the local fuel response at the hot spot. The local uncertainty is incorporated in the HOTSPOT code (Ref.15.6-8) to evaluate the PCT.

15.6.5.3.1.2 Small Break LOCA Evaluation Model

The small break LOCA analysis is performed using the M-RELAP5 code (Ref. 15.6-14), a modified version of the RELAP5-3D, which has multi-dimensional thermal-hydraulics and kinetic modeling capability. One-dimensional modeling with M-RELAP5 is used for LOCAs with break sizes less than 1.0 ft^2.

The following modifications were made to the M-RELAP5 code to incorporate 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements that are also in accordance with the TMI Action Item II.K.3.30 and II.K.3.31.

- Addition of ANS-1971 x 1.2 fission product decay curve
- Addition of Baker-Just correlation (not steam-limited) for metal-water reaction rate calculations
- Addition of ZIRLO™ burst model
- For choked-flow calculation, the Moody model (steam quality > 0.01) and the Henry-Fauske model (steam quality < 0.01) are incorporated to model the discharge
- Return to nucleate and transition boiling heat transfer modes are prevented for the initial blowdown phase

Several M-RELAP5 modeling techniques are used to address specific US-APWR design features:

- Empirical correlations to model the advanced accumulator characteristics are included.

- Safety injection (SI) water temperature rises because the makeup water from the RWSP is recirculated. Temperature rise in the RWSP water is modeled.

A full spectrum of break sizes up to 1.0 ft^2 and various locations are analyzed (Ref. 15.6-16). The spectrum analysis is performed to find out the limiting PCT break size.

15.6.5.3.1.3 Post-LOCA Long term Cooling Evaluation Model

An analysis method with appropriate evaluation model is applied to control the boric acid precipitation and to assure post long term cooling after small and large break LOCAs. Figure 15.6.5-41 shows the evaluation models of post-LOCA long term cooling. These models are similar to the model described in References 15.6-10 through 15.6-13

Fundamental Calculation Method

The fundamental method of boric acid concentration evaluation during the post-LOCA long term cooling is as follows:

(1) Assumptions

- Only cold-leg break is modeled, because boric acid precipitation would not occur in the case of a hot leg break.
- Boric acid only flows in liquid phase. Vapor phase does not contain any boric acid.
- Two volumes are modeled. The first volume includes the core, lower plenum and upper plenum as boric acid condensation volume. The second volume is the RWSP volume as the main source of borated water.

  In this evaluation, the first volume is defined as the "Mixing Volume".
- Void fraction is considered in estimating the inventory of mixing volume.
- The void fraction in the mixing volume is calculated by the modified Yeh's correlation (Ref. 15.6-27).
- Boric acid mixes uniformly.
- Core decay heat is modeled to calculate core evaporation and void fraction.
- Two modes are simulated. The first is RV injection mode. The second is the simultaneous RV and hot leg injection mode.

(2) Initial Conditions

- Calculation is initiated at the beginning of reflood phase.
- The inventory of mixing volume contains a portion of the injected borated water from accumulators.
- The remaining portion of the accumulators inventory spills out into the RWSP.
- The volume of RWSP consists of: its original inventory, accumulators' spillage and RCS coolant.

latent heat.

- SI system injection flow rate decreases with an increase in system pressure.

The first item implies that a higher system pressure reduces boric acid concentration in the mixing volume, while the second one yields a reverse effect. In the evaluation, the atmospheric pressure is assumed for the large break LOCA and a higher pressure for the small break.

Criterion of Boric Acid Precipitation

From Reference 15.6-28, the boric acid precipitation criterion is conservatively assumed to be 29.27 wt.%, which is the precipitation concentration in the atmospheric pressure. Core pressure is higher than the atmospheric pressure, due to the downcomer head and the flow-resistances around the loop. Therefore, the core boiling temperature and the boric acid solubility will be higher than the assumed values. Furthermore, no credit is taken for the RWSP pH additive that increases the boric acid solubility. Hence, this criterion is conservative.

15.6.5.3.2 Input Parameters and Initial Conditions

15.6.5.3.2.1 Large Break LOCA

Table 15.6.5-1 lists the major plant parameter inputs identified for use in the large break LOCA analysis. An initial transient run was made with mostly nominal values, or in some cases, a conservative one. Confirmatory WCOBRA/TRAC runs were performed by varying these limiting parameters over their normal operational ranges to determine the limiting value. The limiting values were used for the reference transient. The other parameters, which are not limiting parameters, are treated as randomly sampled over their operating range in the ASTRUM calculations. Table 15.6.5-1 also lists the major uncertainty parameters and ranges to perform the ASTRUM runs for large break LOCA of the US-APWR based on the operating ranges and other aspects.

- The limiting single failure in the large break LOCA analysis is assumed, which is the loss of one train of ECCS and a second train out of service for maintenance; In this case, only two SI pumps are available.
- Minimum ECCS safeguards are assumed, which results in the minimum delivered ECCS flow available to the RCS.
- Minimum containment pressure is applied for conservatism as described in Section 6.2.1.5.

15.6.5.3.2.2 Small Break LOCA

Spectrum analysis is performed to determine a limiting break size within the small break LOCA category. In addition, sensitivity analyses are reported in Reference 15.6-16, which covers the entire spectrum of break size, break orientation and break location, also noding, time-step size and input sensitivity studies. The sensitivity analyses are performed by complying with the requirements set forth in 10 CFR Appendix K to Part 50 on ECCS Evaluation Models. The objective is performed to determine the effects of various modeling assumption on the calculated PCT, LMO and CWO. Three small break LOCA cases are reported in this section. They are as follows:

- 7.5-inch upside break, which is the limiting break for PCT during the loop-seal clearance phase.
- 1-ft^2 upside break, which is the limiting break for PCT during the boil-off phase.
- 3.4-inch break, which is a DVI line break, with only 1 train of SI system is assumed to operate.

The major plant parameters inputs used in the Appendix-K based small break LOCA analysis are listed in Table 15.6.5.2. The top-skew axial power shape is chosen because it provides the distribution of power versus core height that maximizes the PCT. Figure 15.6.5-13 shows the hot rod power shape used to conduct the small break LOCA analysis. The hot rod power shape considers the axial off-set limits of the core design, and is conservative compared to the limiting large-break LOCA power shape. The beginning of life (BOL) hot assembly burnup provides the maximum (conservative) initial stored energy in the fuel for the SBLOCA event. In addition, for the hot rod, an initial highest pellet temperature is also assumed for conservatism.

In addition to the conditions in Table 15.6.5-2, the following conditions are also applicable to the SBLOCA.

- The limiting single failure in the small break LOCA analysis is assumed, which is the loss of one ECCS train, with one additional train out of service for maintenance; In this case, only two SI pumps are available.
- Minimum ECCS safeguards are assumed, which results in the minimum delivered ECCS flow available to the RCS.
- LOOP is assumed to occur simultaneously with the reactor trip, resulting in the delay of SI pumps and EFWS operations. RCP trip is assumed to occur 3 seconds after the reactor trip, as described in Section 15.0.0.7.
- Shutdown reactivities resulting from fuel temperature and void are given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors as shown in Table 15.6.5-2. Control rod insertion is considered to occur and assumed in the analysis.

15.6.5.3.2.3 Post-LOCA Long Term Cooling

The major input parameters used in the long term cooling evaluation are listed in Table 15.6.5-3. In this evaluation, atmospheric pressure is assumed as the lowest possible system pressure during a large break LOCA. The pressure of 120 psia, which corresponds to the boric acid congruent melting temperature of 339.8°F, is assumed as the highest possible system pressure during a small break LOCA. The initial boric acid concentrations in the RWSP, accumulator, and RCS are assumed to be maximum. Water inventory of RWSP and accumulator are assumed to be maximum because much mass of borated water source makes the concentration in mixing volume higher. RCS water mass is assumed to be minimum because RCS boric acid concentration is lower than RWSP and accumulator.

Safety injection temperature is assumed to be maximum to maximize the core evaporation rate. For a large break LOCA, the assumed injection temperature is the saturation temperature at atmospheric pressure. In the case of a small break LOCA, this temperature is assumed as the RWSP maximum temperature reached during a LOCA. In the post-LOCA long term cooling analysis, the limiting single failure is assumed, which is the loss of the entire train of one ECCS train, with one additional train out of service for maintenance; In this case, only two SI pumps are available.

Operator actions are credited to perform the switchover from the RV injection mode to the simultaneous RV and hot-leg injection mode. The timing of operator action is determined by the solubility limit of boric acid concentration in the core.

15.6.5.3.3 Results

15.6.5.3.3.1 Large Break LOCA Analysis Results

<u>The Result of Reference Transient Calculation</u>

The reference transient calculation is performed based on the confirmatory calculation results in order to obtain the conservative estimation. Figures 15.6.5-1 through 15.6.5-7 present the results of the reference case for the best estimate large break LOCA analysis. The transient is initiated from the end of a steady-state run. The sequence of events for the reference case large break LOCA is listed in Table 15.6.5-6, which shows the plant actions (e.g. trips, etc) and those phenomena observed in the calculation (e.g., end of blowdown, etc).

(1) Blowdown phase

During the first few seconds of the transient, the core water inventory decreases rapidly. During the blowdown phase, the initial stored energy is the main contributor to the temperature rise and boiling. The decay heat is a secondary contributor. The RCPs are presumed to trip concurrent with the break in the LOOP scenario. Consequently, DNB occurs and the cladding temperature rises quickly even though the core power decreases. The hot rod cladding temperature at the limiting elevation for large break LOCA is shown in Figure 15.6.5-1. At six seconds into the transient, an ECCS actuation signal is generated due to the low pressurizer pressure. In the early blowdown phase, an upward flow takes place in the core removing the core decay heat by way of two-phase heat transfer. About 13 seconds into the transient, the accumulator begins to inject water at a high rate into the cold leg regions.

Figure 15.6.5-2 shows the hot assembly exit vapor, entrainment, and liquid flowrates transients. This figure displays the flow rates for the vapor, entrained liquid and continuous liquid at the top of the hot assembly.

that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The analyses are carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling is achieved.

Based on the analysis, the application of ASTRUM for the best-estimate analysis of the large break LOCA shows that the acceptance criteria of 10 CFR 50.46 are satisfied for the US-APWR. In addition, it is confirmed that 2 (two) safety injection trains are capable of satisfying the design cooling function for any large break LOCA, assuming a single failure of one train, and another train out of service for maintenance.

15.6.5.3.3.2 Small Break LOCA Analysis Results

Details for the limiting small break LOCA are presented in this section. The results for other cases are documented in detailed in Technical Report (Ref. 15.6-16).

Results of 7.5-inch Small Break LOCA Analysis

The sequence of events for the 7.5-inch small break LOCA is presented in Table 15.6.5-9. Depressurization of the RCS (Figure 15.6.5-14) causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated when the low pressurizer pressure setpoint of 1860 psia is reached. The reactor trips at 9.3 seconds, then the power decreases (Figure 15.6.5-15). Control rod insertion starts at 11 seconds, which is concurrent with the turbine trip and main steam isolation. Voiding in the core also causes the reactor power to decrease.

The liquid and vapor discharges out of the break are shown in Figure 15.6.5-16. During the earlier part of the transient, the effect of the break flow is not strong enough to overcome the upward flow through the core that is maintained by the coasting RCPs. The ECCS actuation signal occurs at 12 seconds when the low pressurizer pressure setpoint is reached. This is immediately followed by the RCPs trip just before 13 seconds. The main feedwater flow is isolated at 17 seconds. To limit the pressure build up in the secondary system, the main steam safety valves open at 81 seconds. The upper region of the core begins to uncover at 122 seconds. Figure 15.6.5-17 shows the accumulator and safety injection mass flowrates transient. The HHIS begins to inject borated water to the reactor core at 130 seconds. The accumulators begin injecting borated water into the cold-leg at about 300 seconds.

As a result of the loop-seal clearance, the core is recovered at ~~142~~ 143 seconds. Figure 15.6.5-18 shows the RCS inventory transient. The downcomer liquid collapsed level and core/upper plenum liquid collapsed level are shown in Figures 15.6.5-19 and 15.6.5-20, respectively.

Figure 15.6.5-21 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and the average rod in the hot assembly that contains the hot rod. The PCT of ~~773~~775°F occurs at 136 seconds. This figure demonstrates that the PCT is substantially lower than 2200°F.

Figure 15.6.5-22 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-10 presents the calculated PCT, LMO, and CWO results for the limiting 7.5-inch small break LOCA. This case is the limiting break for PCT during the loop-seal clearance phase.

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F. The PCT of ~~773~~775°F presented in Table 15.6.5-10 indicates that this regulatory limit has been met.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-10 indicates that this regulatory limit has been met.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. The maximum core wide cladding oxidation is lower than 0.2 % as presented in Table 15.6.5-10 in compliance with regulatory limit.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met since the PCT does not exceed 2200°F. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Results of 1-ft^2 Small Break LOCA Analysis

The sequence of events for the 1-ft^2 break, which is a 13.5-inch equivalent diameter small break LOCA is presented in Table 15.6.5-11. This is the limiting break for PCT during the boil-off phase.

Figure 15.6.5-23 depicts the pressure transient in the pressurizer. Depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated at 6.9 seconds when the low pressurizer pressure setpoint is reached. LOOP is assumed at the same time with the reactor trip. The reactor power then decreases (Figure 15.6.5-24) following the reactor trip. Control rod insertion and main steam flow isolation occur at 8.7 seconds. The RCPs trip at 9.9 seconds, indicating 3 seconds delay from the reactor trip. Main feedwater flow is isolated at 15 seconds. Because secondary system pressure build up does not occur, the main steam safety valves remain closed.

The liquid and vapor discharges from the break are shown in Figure 15.6.5-25. Early in the transient, the effect of the break flow is not strong enough to overcome the upward flow through the core that is maintained by the coasting RCPs. Upward flow through the core is maintained. However, the flow rate is not sufficient to prevent partial uncovery in the core.

The ECCS actuation signal is generated when the low pressurizer pressure setpoint is reached at 8 seconds.

Figure 15.6.5-26 shows the accumulator and safety injection mass flow rates. The accumulators begin injecting borated water into the cold-leg at ~~90~~ 89 seconds. The HHIS begins to inject borated water to the reactor core at 126 seconds. As a result of ECCS injection, the mass inventory is recovered. Figure 15.6.5-27 shows the RCS inventory transient. The downcomer liquid collapsed level and core/upper plenum liquid collapsed level transients are shown in Figures 15.6.5-28 and 15.6.5-29, respectively. Figure 15.6.5-30 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and for the average rod in the hot assembly that contains the hot rod. This figure shows that the PCT of ~~1323~~1088°F occurs at ~~169~~ 151 seconds. The PCT is significantly lower than 2200°F.

Figure 15.6.5-31 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-12 presents the 1-ft^2 upside break, which is a 13.5-inch equivalent diameter small break LOCA.

1. The PCT of ~~1323~~1088°F presented in Table 15.6.5-12 indicates that this regulatory limit has been met.

2. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-12 indicates that this regulatory limit has been met.

3. The maximum core wide cladding oxidation is lower than 0.2% as presented in Table 15.6.5-12, in compliance with regulatory limit.

4. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Results of the DVI-Line Small Break LOCA Analysis

The sequence of events for the DVI-line break, which is a 3.4-inch equivalent diameter small break LOCA is presented in Table 15.6.5-11. This case assumes the injection of only one SI pump.

Depressurization of the RCS (Figure 15.6.5-32) causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated when the low pressurizer pressure setpoint is reached at 26 seconds. The reactor power then decreases (Figure 15.6.5-33) following the reactor trip. Control rod insertion starts at 28 seconds, simultaneous with the turbine trip and main steam isolation. The RCP trips at 29 seconds, which is 3 seconds after the reactor trip.

The liquid and vapor discharges out of the break are shown in Figure 15.6.5-34. Downward flow does not occur in this particular case. Upward flow through the core is maintained. The core flow is sufficient to prevent any uncovery of the core.

The ECCS actuation signal is initiated when the low pressurizer pressure setpoint of 1760 psia is attained at 35 seconds. In this case, the HHIS alone provides the core cooling function. Figure 15.6.5-35 shows the accumulator and safety injection mass flow rates. Figure 15.6.5-36 shows that the RCS inventory increases. The downcomer liquid collapsed level transient and core/upper plenum liquid collapsed level transient are shown in Figures 15.6.5-37 and 15.6.5-38, respectively.

Figure 15.6.5-39 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and the average rod in the hot assembly that contains the hot rod. This figure shows that the PCT does not occur in the DVI-line break, indicating that the core keeps covered throughout the transient.

Figure 15.6.5-40 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-14 presents the DVI-line break, which is a 3.4-inch equivalent diameter small break LOCA.

1. For the DVI-line break, no heatup occurs. This obviously demonstrates that the regulatory limit has been met.

2. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-14 indicates that this regulatory limit has been met.

3. The maximum core wide cladding oxidation is not observable because core uncovery does not even occur.

4. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Based on the analysis, the acceptance criteria of 10 CFR 50.46 are satisfied for the US-APWR. In addition, it is confirmed that two safety injection trains are capable of satisfying the design cooling function for any small break LOCAs, assuming a single failure of one train, and another train out of service for maintenance. Concluding the small break LOCA analysis, Table 15.6.5-15 lists the spectrum of peak cladding temperatures.

15.6.5.3.3.3 Post-LOCA Long Term Cooling Evaluation Results

Results of the Large Break LOCA

Figure 15.6.5-42 shows the calculated time-history of the core boric acid concentration and the solubility limit used for this calculation. In the figure, the solid line indicates that the boric acid concentration gradually increases as time advancing. The dotted line imposes the criterion of boric acid precipitation. This implies that the switchover to the hot leg injection mode must be performed before the precipitation limit is reached. The calculation indicates that a switchover at around four hours after the LOCA assures that the boric acid concentration remains below the solubility limit. After the switchover, the boric acid concentration decreases. In contrary, the dashed line shows that the concentration would increase beyond the precipitation limit if the switchover were not performed. Figure 15.6.5-42 also shows the dilution effect of the hot leg injection flow after the switchover.

Results of the Small Break LOCA

In the case of a small break LOCA, the SI flowrate is relatively small compared with the large break LOCA because RCS pressure remains high. The simultaneous RV and hot leg injection may affect the dilution behavior of the boric acid in the core. In the small break LOCA, two cases are considered with regard to the break area.

If the break size is small, the RCS pressure is maintained high and retained in a subcooled condition due to the SI system operation. In this case, the boiling of core may not occur and two-phase natural circulation is established. This situation prevents the boric acid build up in the core.

If the break size is relatively large, RCS depressurizes to relatively low pressure. Therefore, it is necessary to calculate the boric acid concentration in the core for the long term cooling evaluation in this case.

The congruent melting temperature of boric acid is 339.8°F, which is slightly lower than the saturation temperature at 120 psia (341.3°F). Therefore, cases at pressures higher than 120 psia need not be considered and the bounding case for boric acid precipitation is

15.6.5.6 Conclusions

The US-APWR satisfied all criteria for the postulated LOCA transient:

- The best-estimate analysis of the large break LOCA demonstrates that the acceptance criteria of 10 CFR 50.46 are satisfied.
- The conservative analysis of the small break LOCA, which is based on the Appendix K, demonstrates that the acceptance criteria of 10 CFR 50.46 are satisfied.
- The switchover to the simultaneous RV and hot leg injection mode at four hours after a LOCA prevents boric acid precipitation in the core, and the post-LOCA long term cooling is assured.
- The EAB and LPZ doses are shown to meet the 10 CFR 50.34 dose guidelines.
- The dose for the MCR personnel is shown to meet the dose criteria given in GDC 19.
- The requirements of the TMI Action Plan items are met.

Table 15.6.5-2

US-APWR Major Plant Parameter Inputs Used in the Appendix-K based Small Break LOCA Analysis

Parameters	Values
Core and Fuel Rod Condition	
Core Power	102% of rated power (4540 MWt)
Peaking factor	$F_Q = 2.6$
Axial power shape	Top-skew (double humps), as shown in Figure 15.6.5-13.
Hot assembly burnup	Beginning of life (BOL)
Fuel assembly type	17 X 17 ZIRLO™ cladding
Plant Operating Condition	
Fraction of SG tube plugged	10% (maximum)
RCS average temperature	Nominal value + 4°F (587.8°F)
Pressurizer pressure	Nominal value + 30 psia (2280 psia)
Primary coolant flow	Thermal design flow (112,000 gpm/loop)
RV upper head temperature	Nominal (T_{cold})
Pressurizer level	Nominal
Accumulator temperature	Maximum (120°F)
Accumulator pressure	Minimum (600 psia)
Accumulator volume	Nominal (2152 ft^3)
Accident Boundary Condition	
Break location	Cold leg
Break type	Split
Break sizes	• 7.5-inch diameter break • 1.0 ft^2 break • 3.4-inch diameter DVI-line break
Offsite power	Not available
Reactor trip signal	Low pressurizer pressure
Reactor trip signal delay time	1.8 seconds
RCP trip (at LOOP)	3 seconds after reactor trip
ECCS actuation	Low pressurizer pressure
Safety injection delay	Maximum (118 seconds)
Number of available SI pumps	2 pumps for cold leg break 1 pump for DVI line break
Safety injection flow	Minimum
Safety injection water temperature	RWSP temperature rise is modeled

Table 15.6.5-9

Sequence of Events for 7.5-inch Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	9.3
Control rod insertion starts	11.1
Main steam isolation	11.1
ECCS actuation signal	11.9
RCP trip	12.3
Main feedwater isolation	17.3
Main steam safety valve open	81
Emergency Power Source initiates	115
Core upper region uncovery	122
High Head Injection System begins	130
Peak Cladding Temperature occurs	136
Core upper region recovery	~~142~~143
Emergency feedwater flow begins	145
Accumulator injection begins	~~299~~315

Table 15.6.5-10

Core Performance Results for 7.5-inch Small Break LOCA

	Values
Peak Cladding Temperature (°F)	~~773~~775
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	less than 0.2

Table 15.6.5-11

Sequence of Events for 1-ft^2 Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	6.9
ECCS actuation signal	8.3
Control rod insertion starts	8.7
Main steam isolation	8.7
RCP trip	9.9
Main feedwater isolation	14.9
Main steam safety valve open	not actuated
Accumulator injection begins	~~90~~89
Core upper region uncovery	~~96~~95
Emergency Power Source initiates	111
High Head Injection System begins	126
Emergency feedwater flow begins	141
Peak Cladding Temperature occurs	~~169~~151
Core upper region recovery	~~339~~169

Table 15.6.5-12

Core Performance Results for 1-ft^2 Small Break LOCA

Items	Values
Peak Cladding Temperature (°F)	~~1323~~1088
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	less than 0.2

Table 15.6.5-13

Sequence of Events for DVI-line Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip, (LOOP is assumed)	~~25.9~~25.8
Control rod insertion starts	~~27.7~~27.6
Main steam isolation	~~27.7~~27.6
RCP trip	~~28.9~~28.8
Main feedwater isolation	~~33.9~~33.8
ECCS actuation signal	35.4
Main steam safety valve open	57
Emergency Power Source initiates	138
High Head Injection System begins	153
Emergency feedwater flow begins	168
Core upper region uncovery	not occur
Peak Cladding Temperature	lower than the initial value
Core upper region recovery	N/A

Table 15.6.5-14

Core Performance Results for DVI-line Small Break LOCA

Items	Values
Peak Cladding Temperature (°F)	lower than the initial value
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	N/A

Table 15.6.5-15

Spectrum of Peak Cladding Temperatures for Small Break LOCA

Break size and orientation	PCT
1-ft^2 at cold leg (bottom)	~~1174~~1029°F
13-inch at cold leg (bottom)	~~1154~~971°F
12-inch at cold leg (bottom)	~~938~~741°F
11-inch at cold leg (bottom)	lower than the initial temperature
10-inch at cold leg (bottom)	lower than the initial temperature
9-inch at cold leg (bottom)	lower than the initial temperature
8-inch at cold leg (bottom)	~~lower than the initial temperature~~696°F
7.5-inch at cold leg (bottom)	~~761~~758°F
7-inch at cold leg (bottom)	~~756~~759°F
6.5-inch at cold leg (bottom)	~~lower than the initial temperature~~718°F
6-inch at cold leg (bottom)	~~lower than the initial temperature~~719°F
5-inch at cold leg (bottom)	lower than the initial temperature
4-inch at cold leg (bottom)	lower than the initial temperature
3-inch at cold leg (bottom)	lower than the initial temperature
2-inch at cold leg (bottom)	lower than the initial temperature
1-inch at cold leg (bottom)	lower than the initial temperature

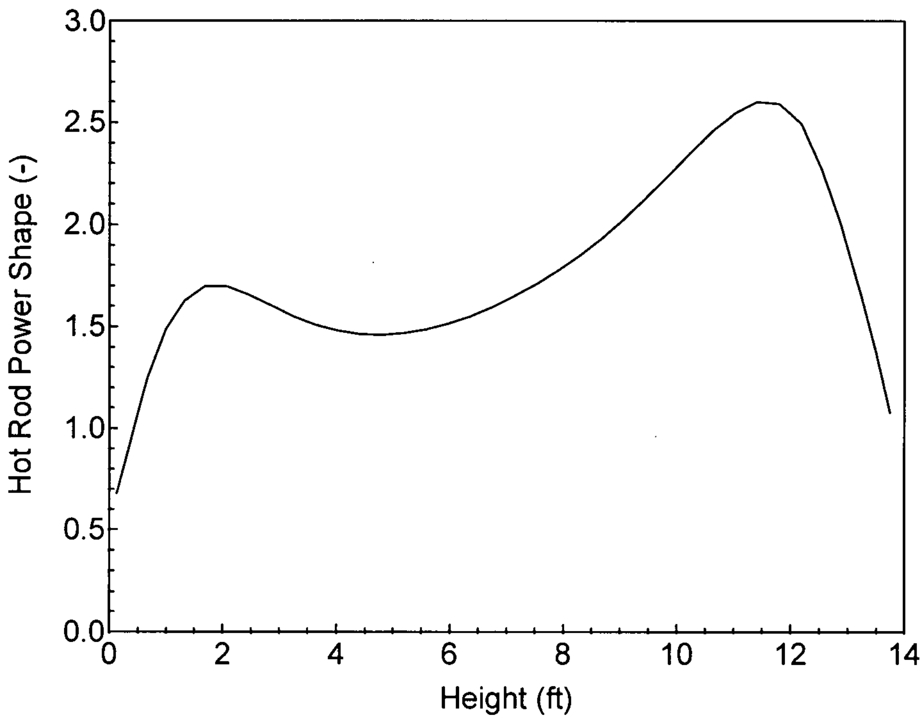
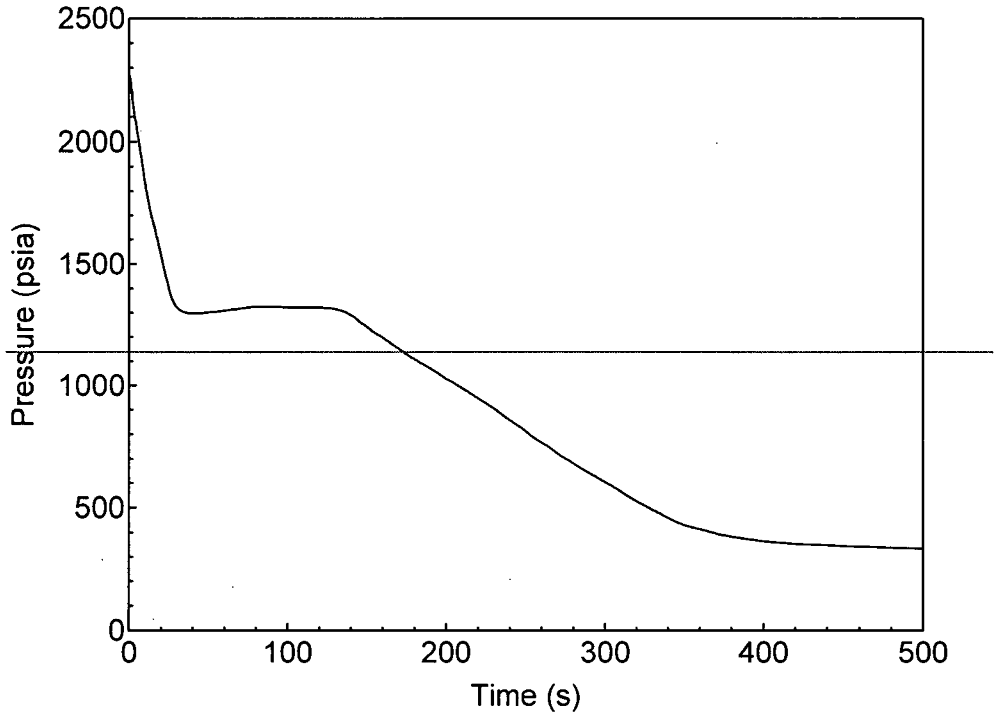


Figure 15.6.5-13 Hot Rod Power Shape Used for Small Break LOCA analysis



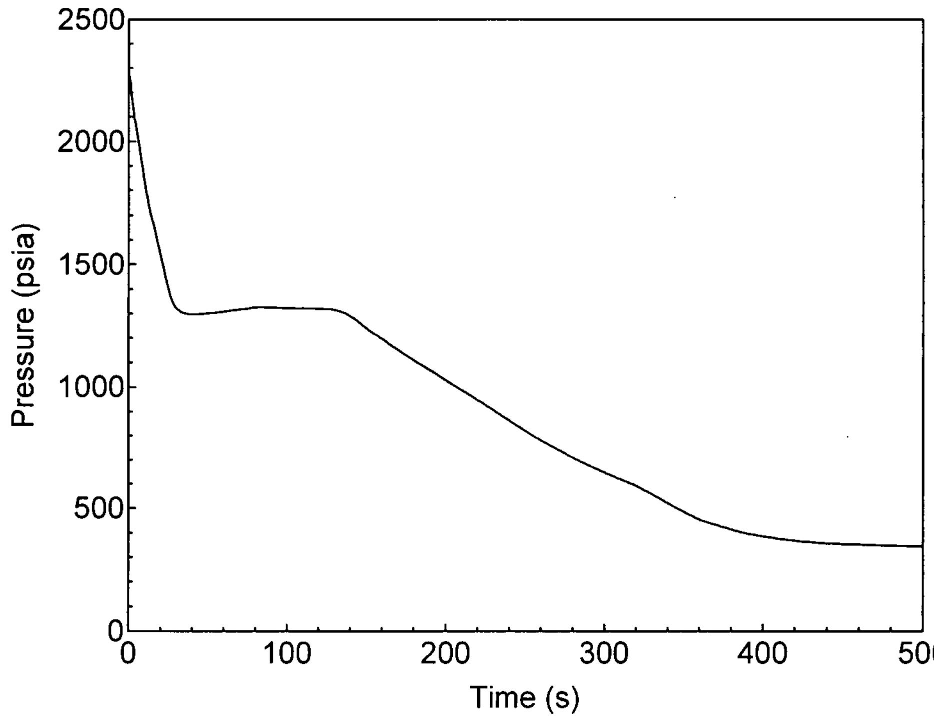


Figure 15.6.5-14 RCS (Pressurizer) Pressure Transient for 7.5-inch Small Break LOCA

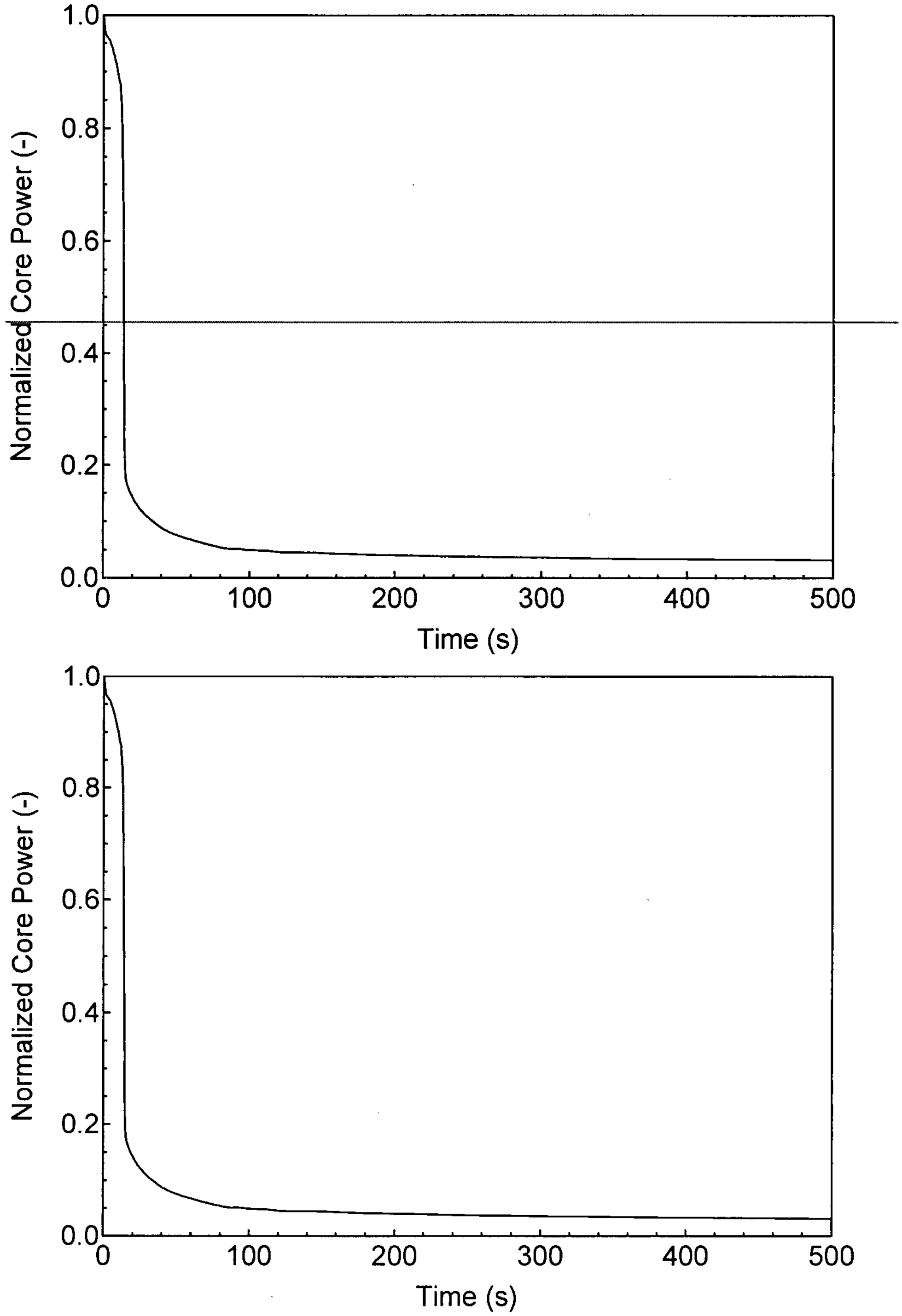


Figure 15.6.5-15 Normalized Core Power for 7.5-inch Small Break LOCA

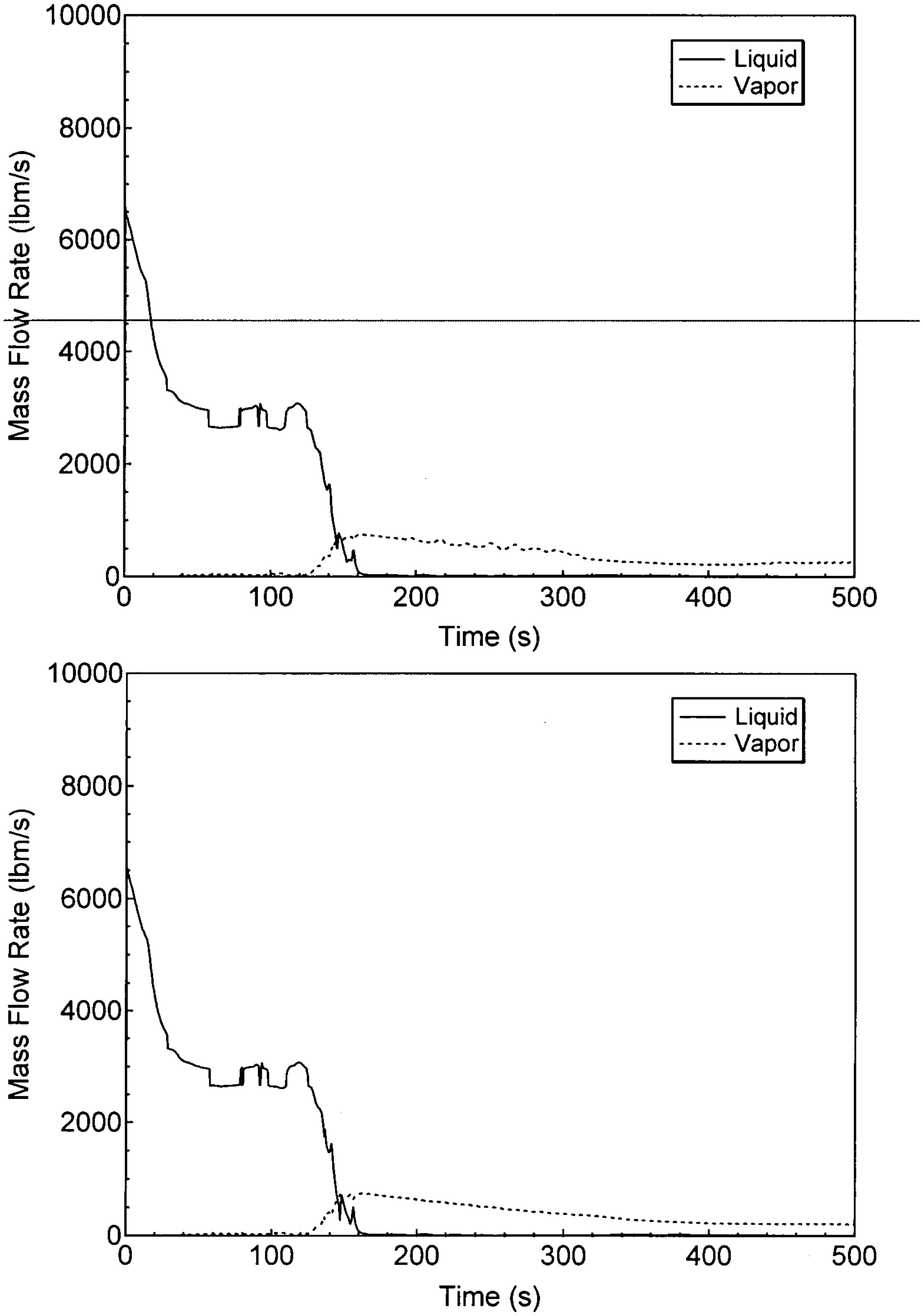


Figure 15.6.5-16 Liquid and Vapor Discharges through the Break for 7.5-inch Small Break LOCA

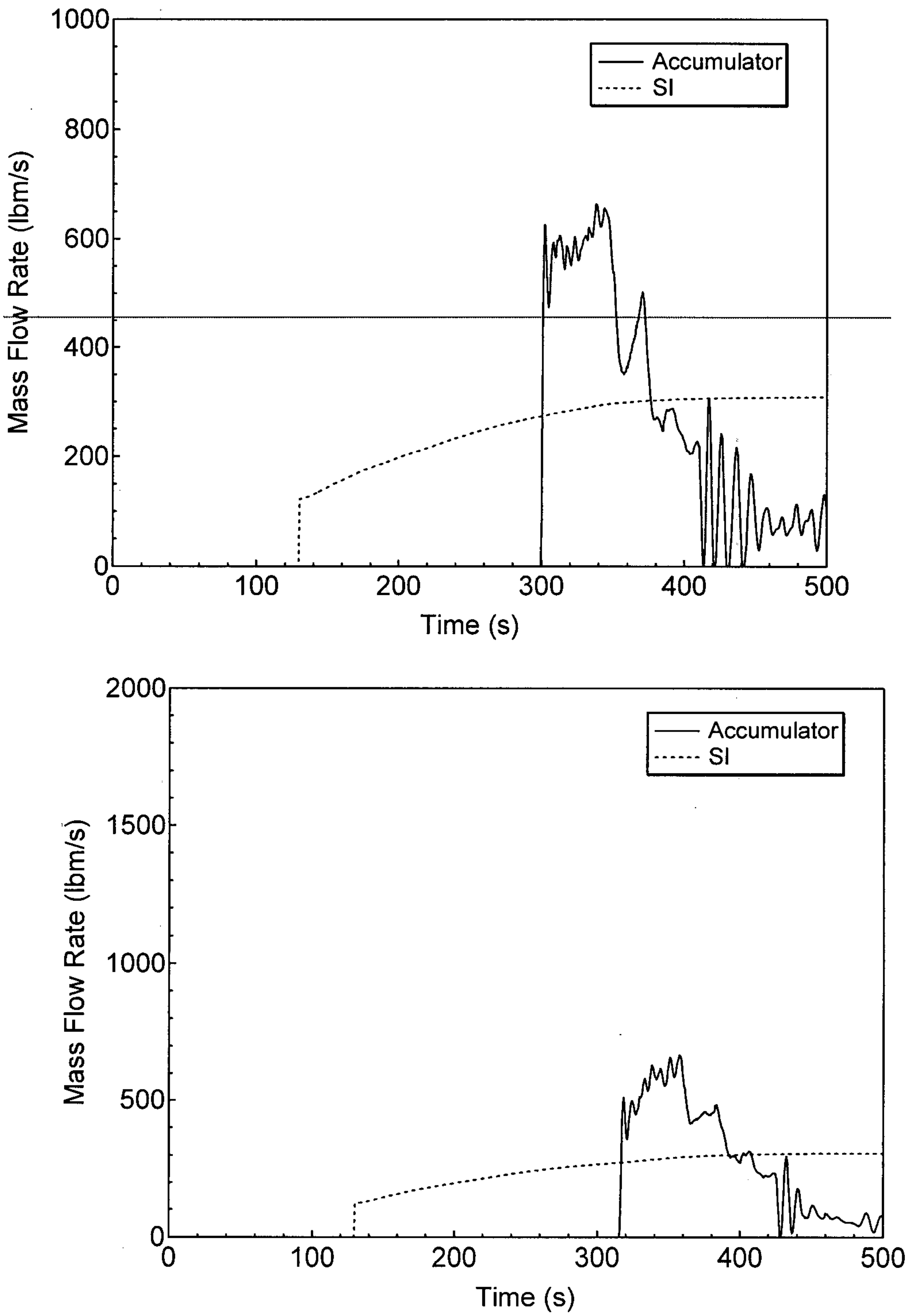


Figure 15.6.5-17 Accumulator and Safety Injection Mass Flowrates for 7.5-inch Small Break LOCA

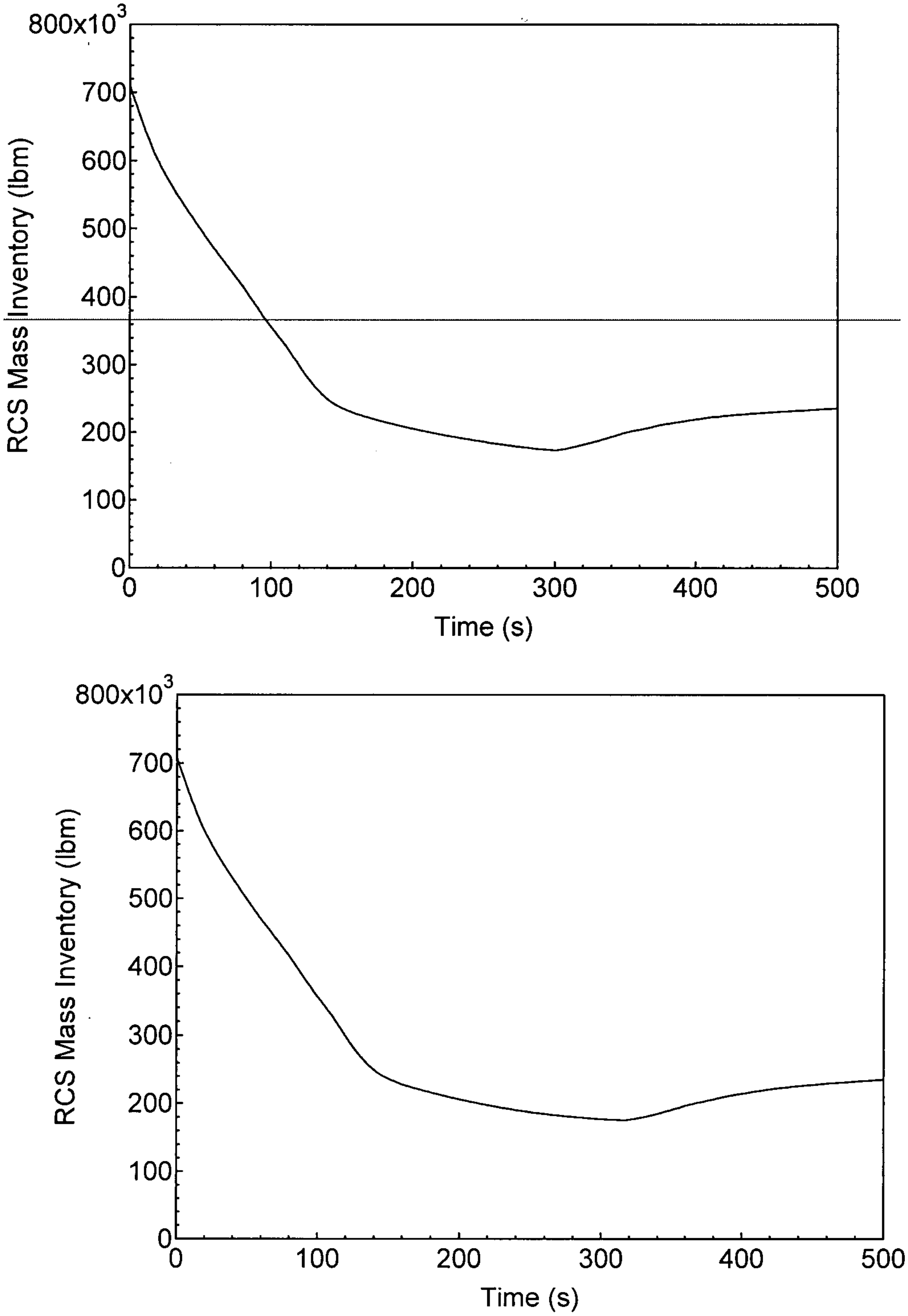
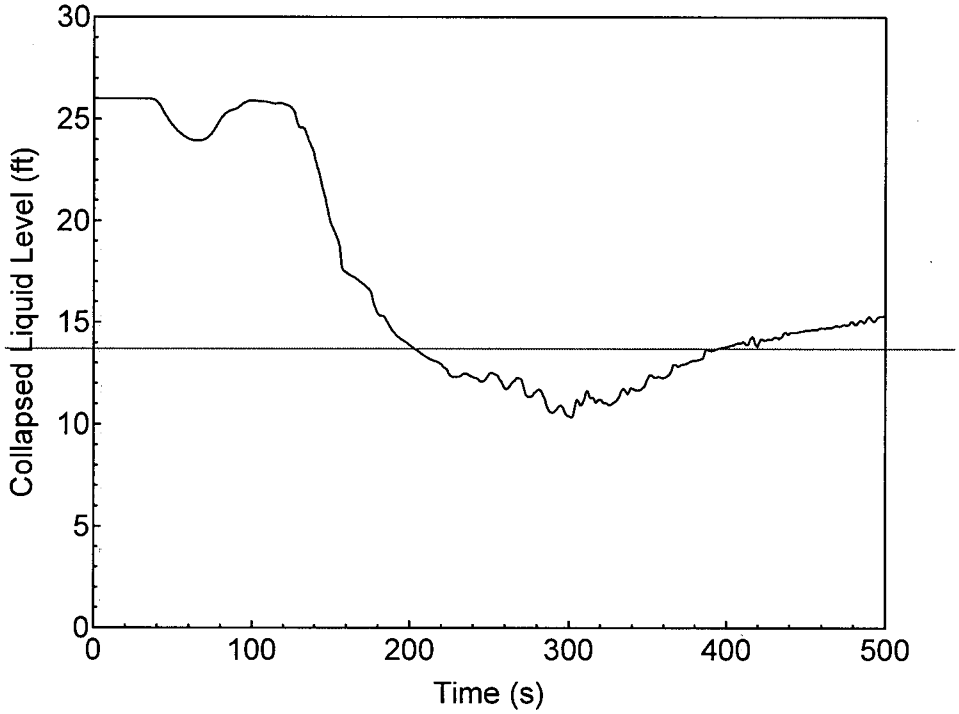


Figure 15.6.5-18 RCS Mass Inventory for 7.5-inch Small Break LOCA



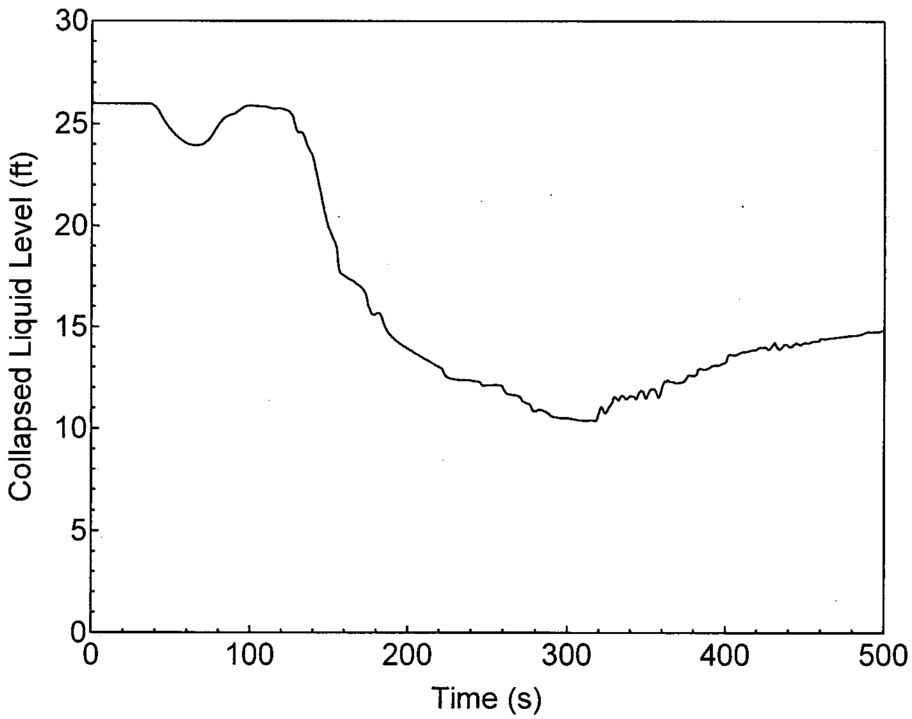
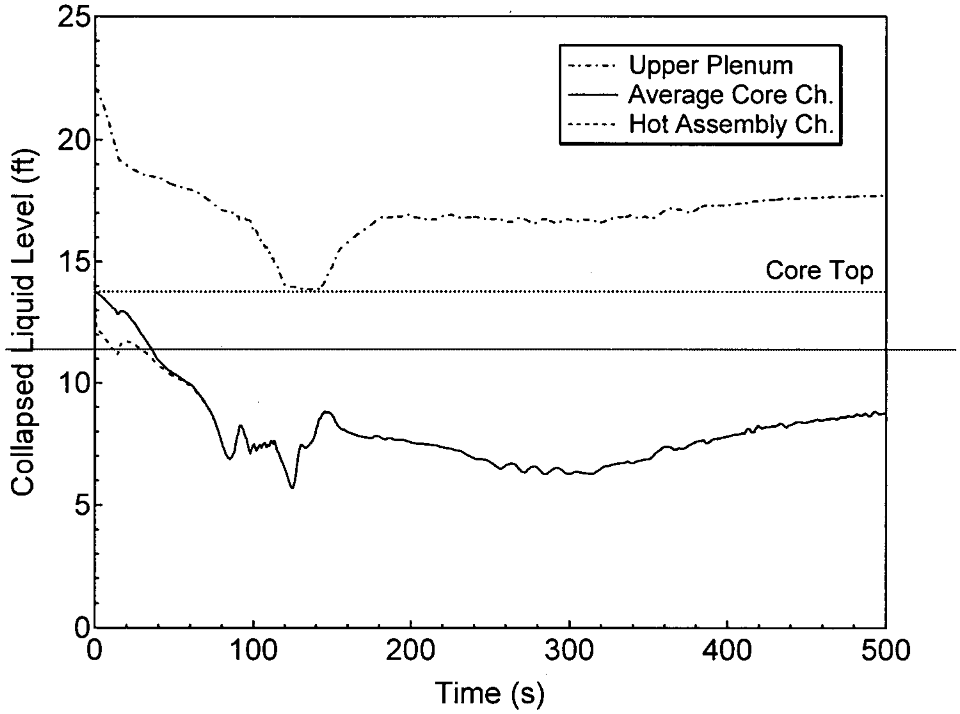


Figure 15.6.5-19 Downcomer Collapsed Level for 7.5-inch Small Break LOCA



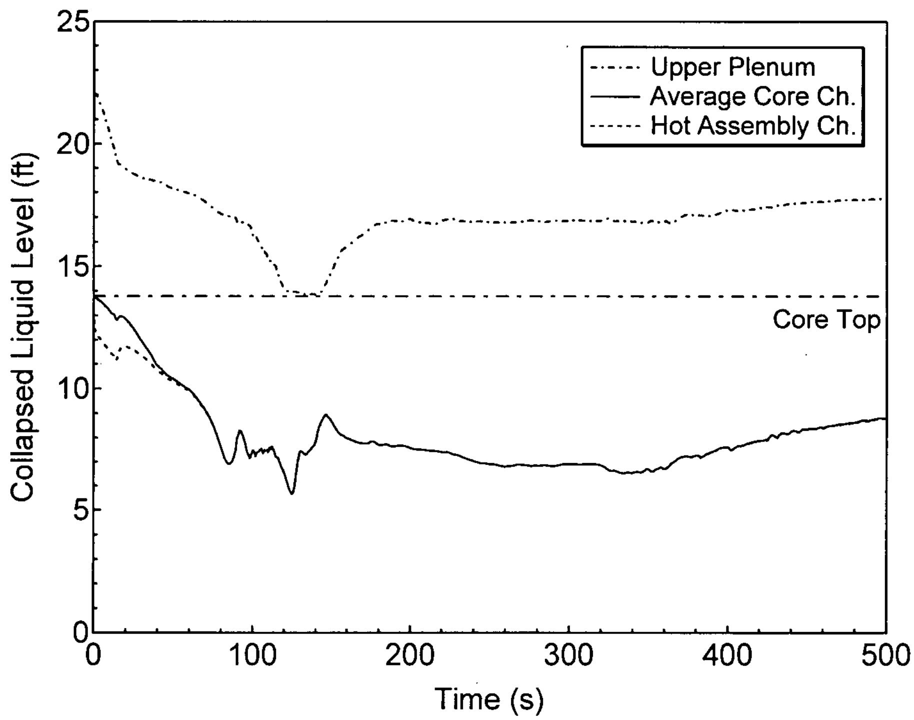
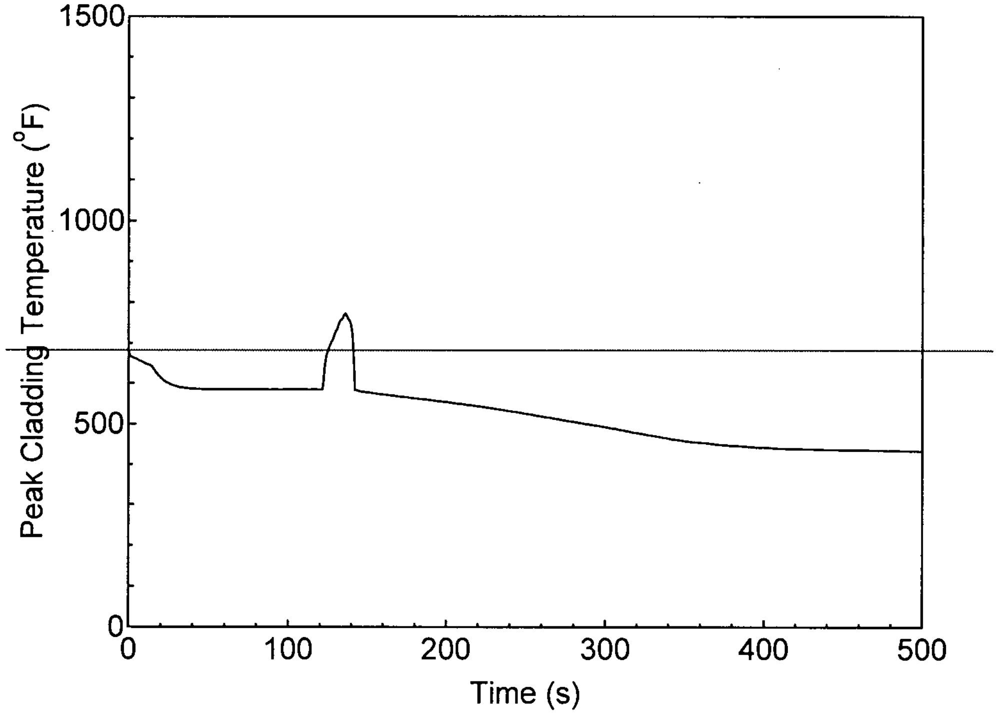


Figure 15.6.5-20 Core/Upper Plenum Collapsed Level for 7.5-inch Small Break LOCA



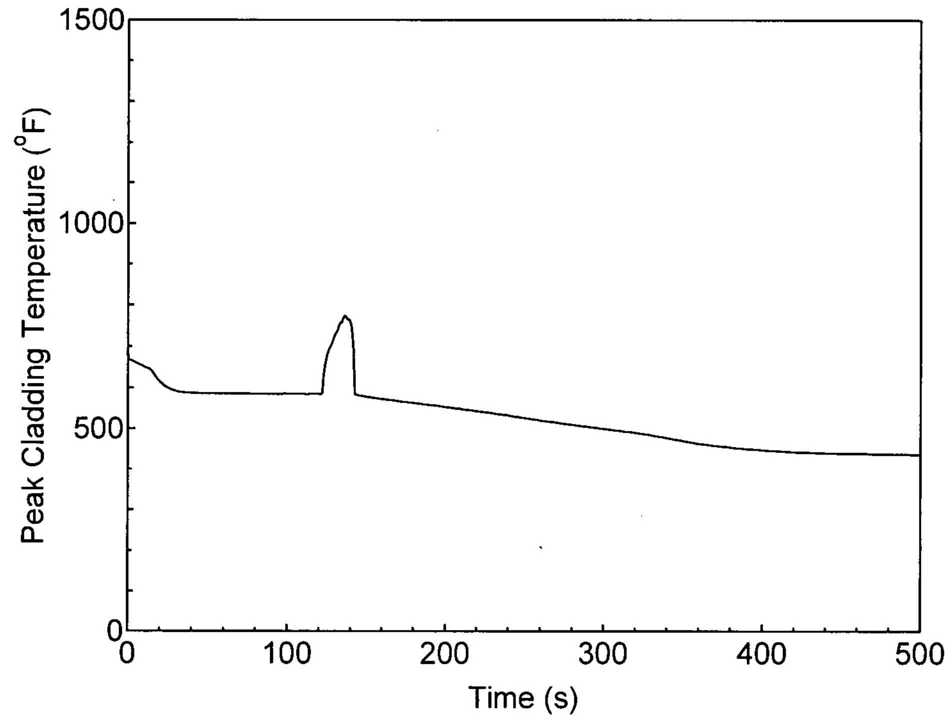
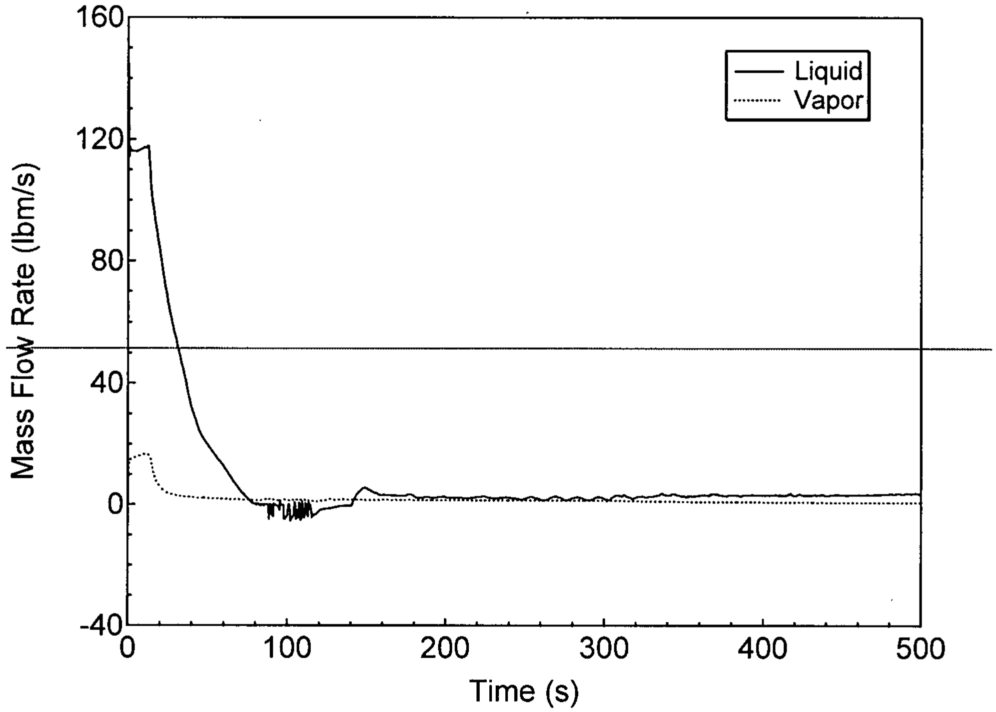


Figure 15.6.5-21 PCT at All Elevations for Hot Rod in Hot Assembly for 7.5-inch Small Break LOCA



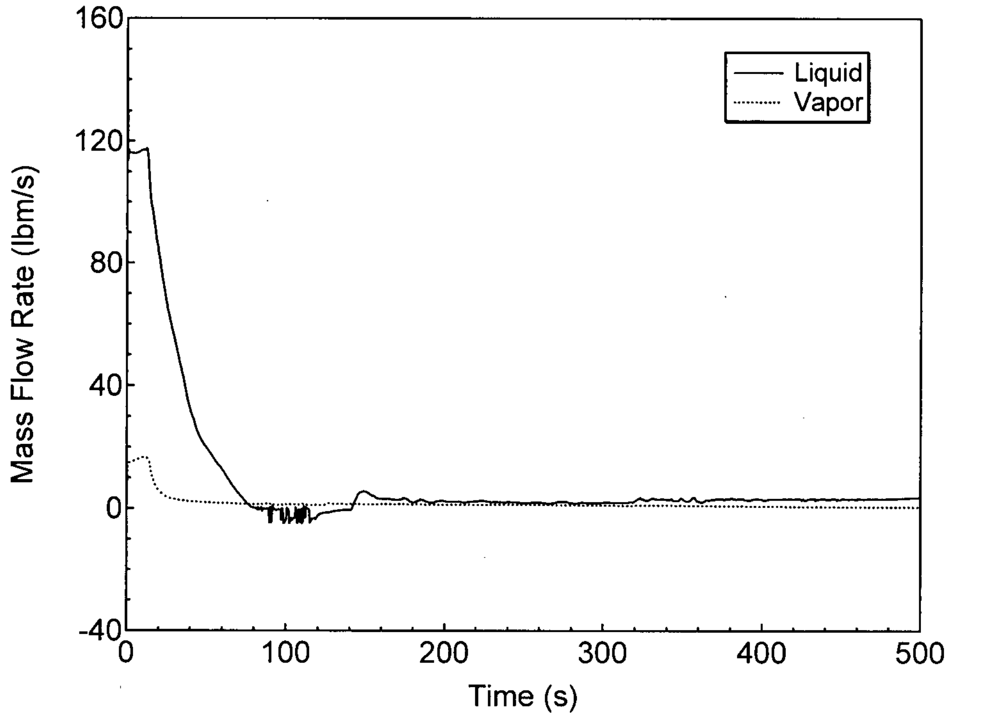


Figure 15.6.5-22 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 7.5-inch Small Break LOCA

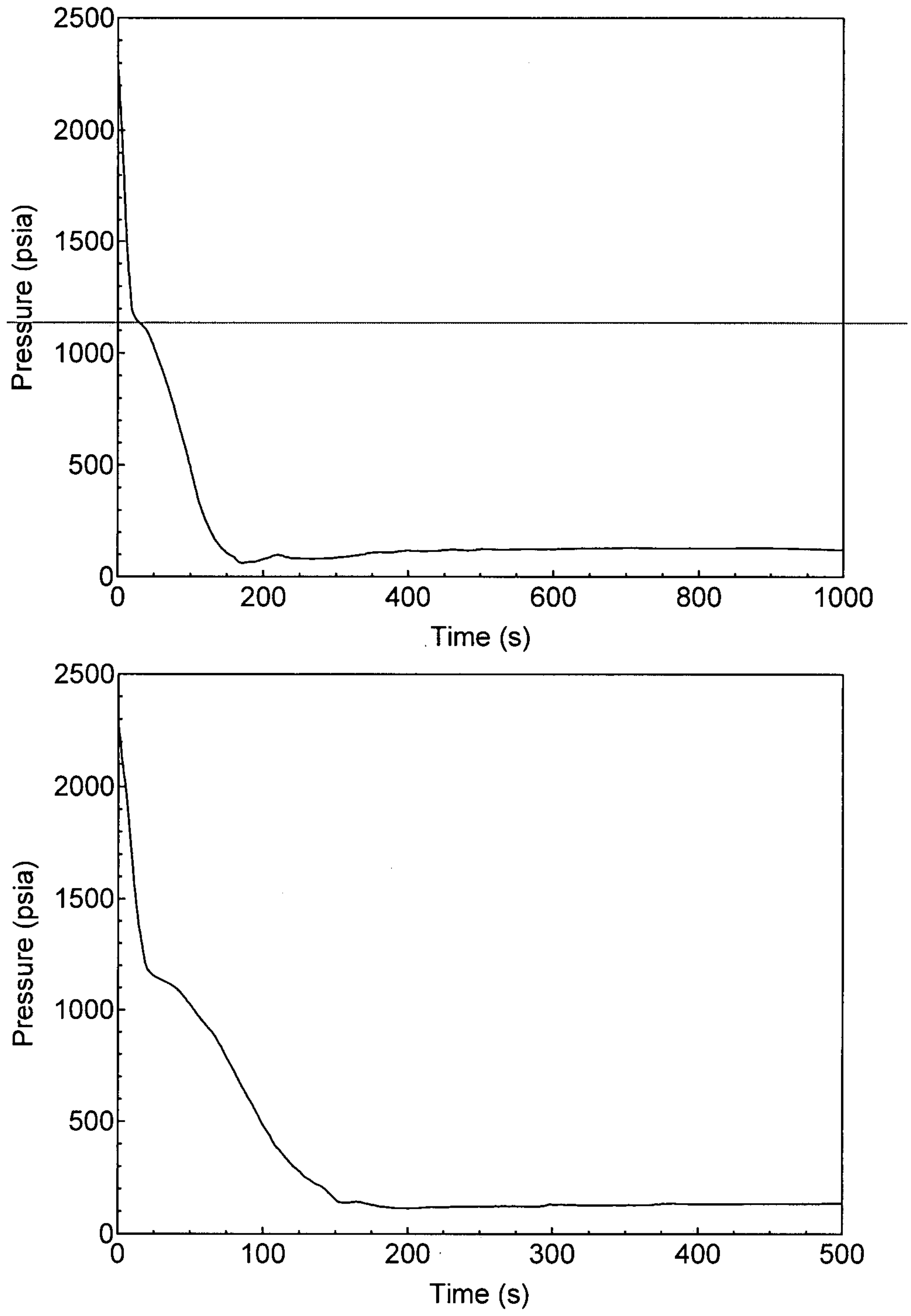


Figure 15.6.5-23 **RCS (Pressurizer) Pressure Transient for 1-ft^2 Small Break LOCA**

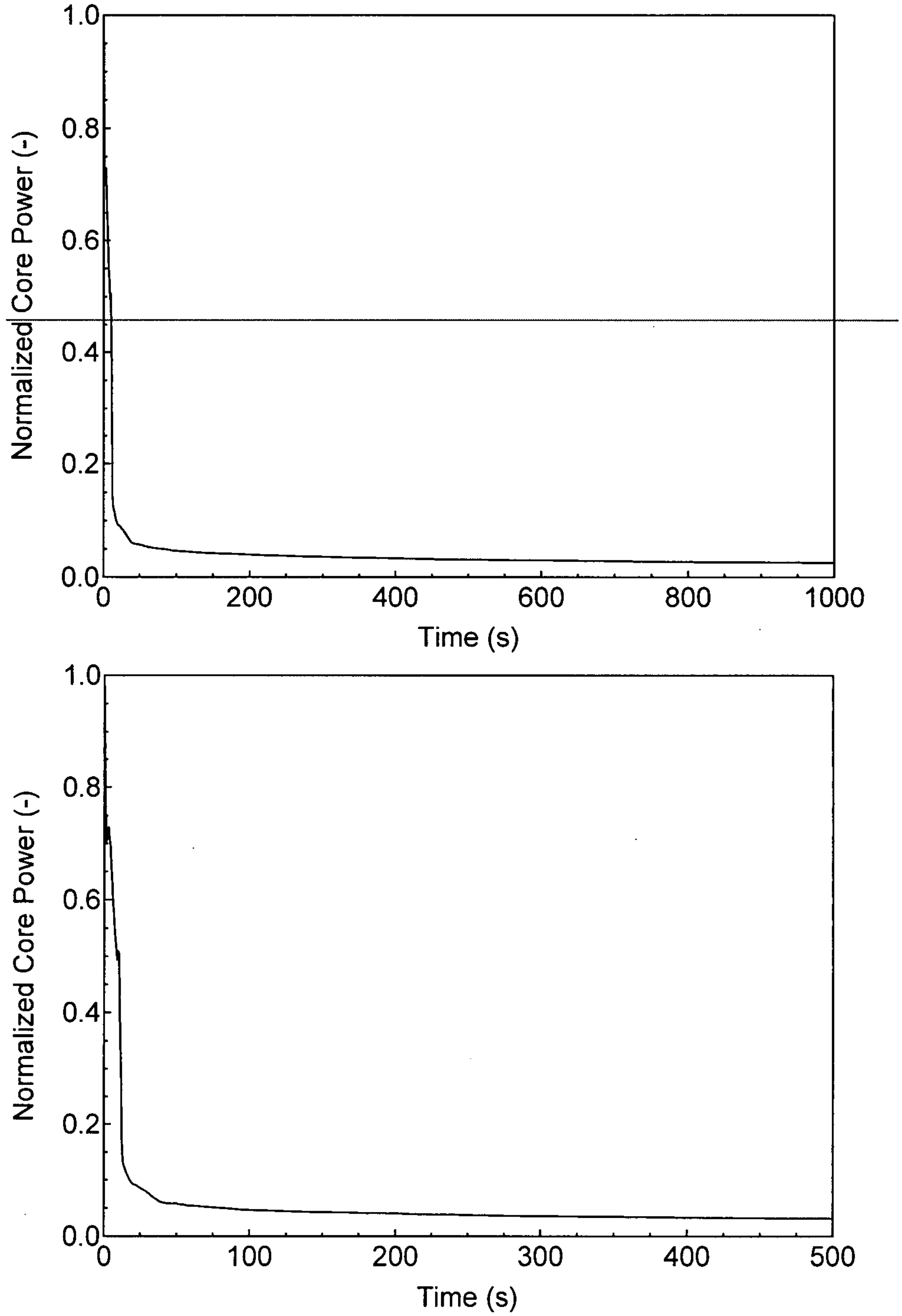


Figure 15.6.5-24 Normalized Core Power for 1-ft^2 Small Break LOCA

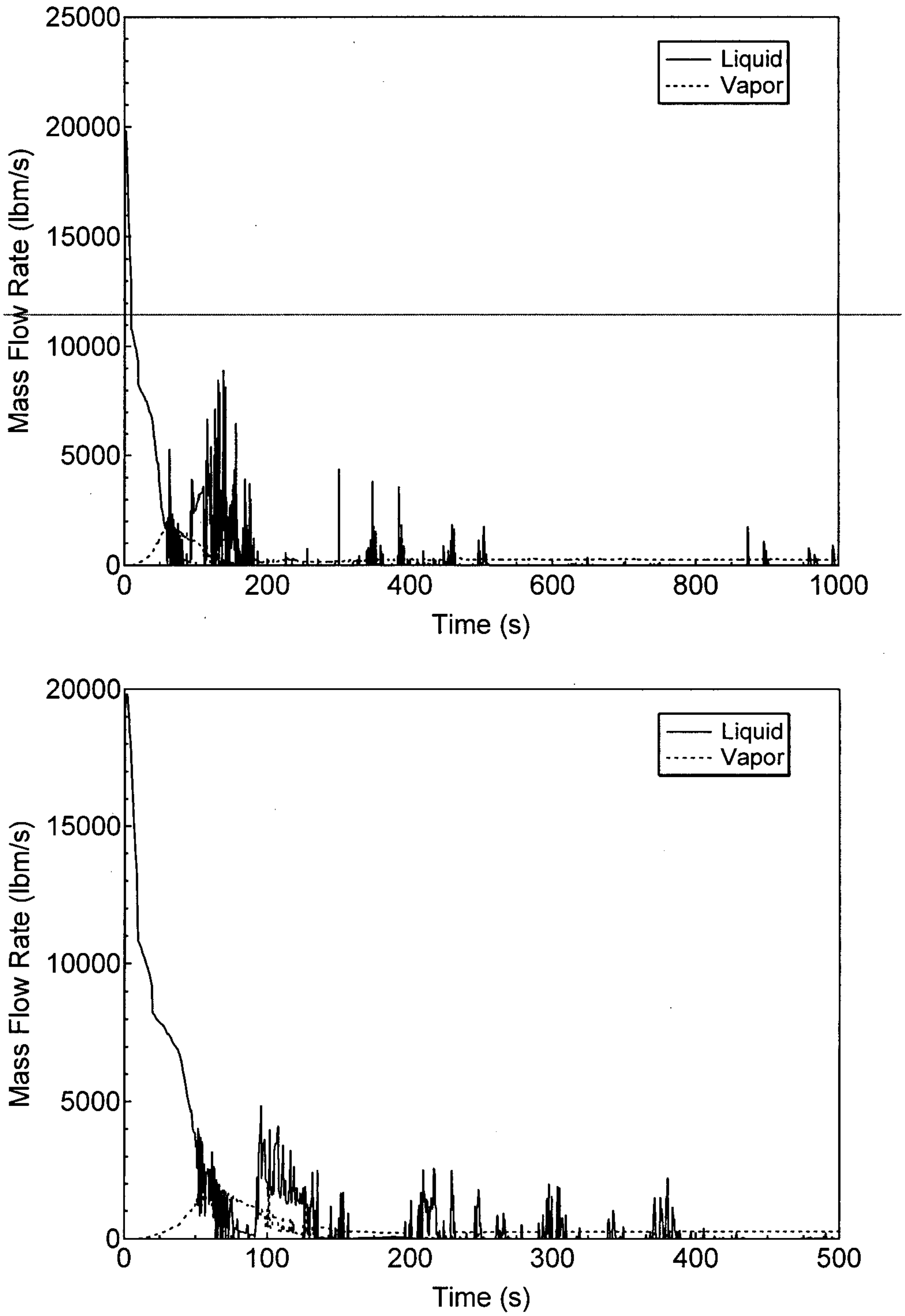


Figure 15.6.5-25 Liquid and Vapor Discharges through the Break for 1-ft^2 Small Break LOCA

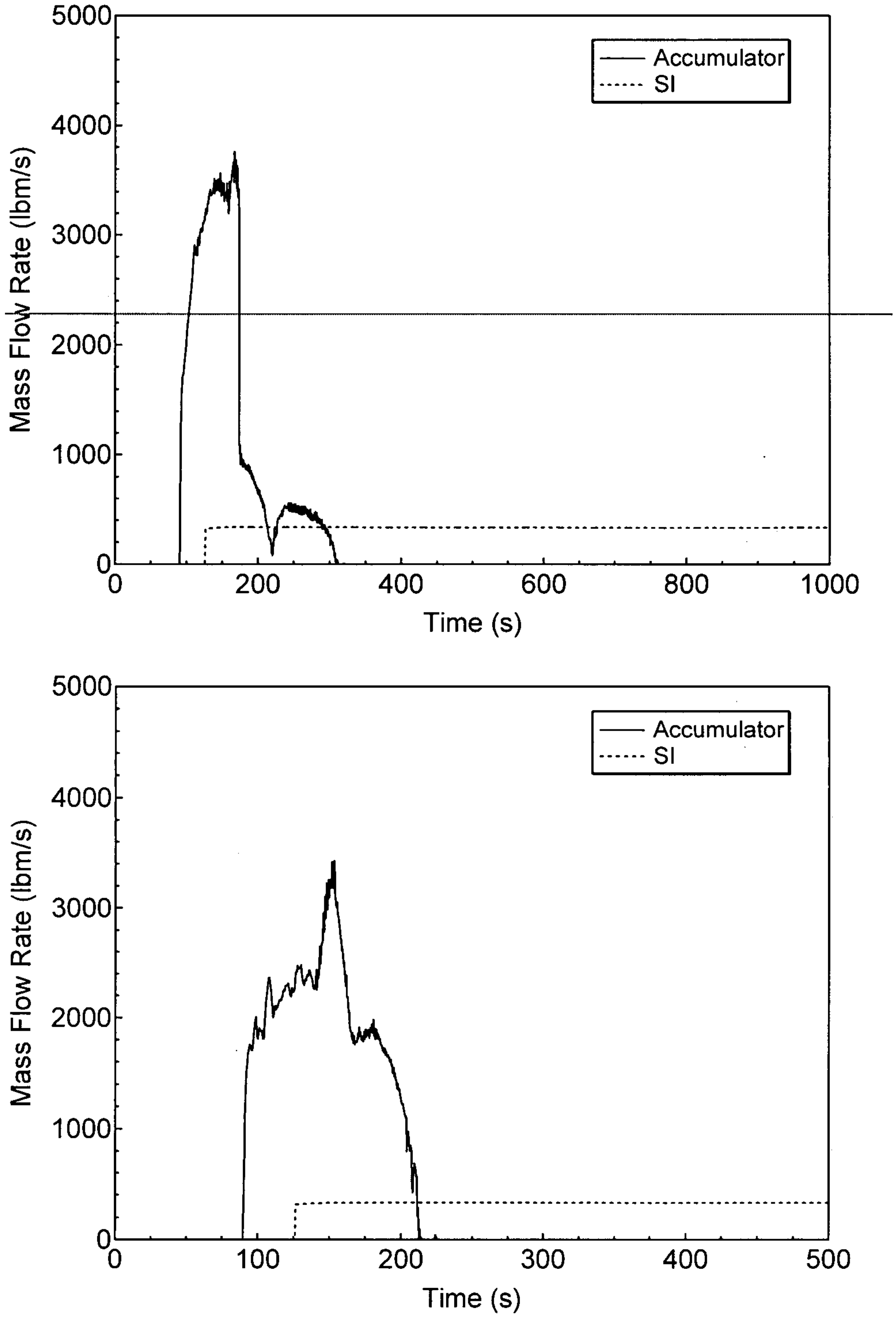


Figure 15.6.5-26 Accumulator and Safety Injection Mass Flowrates for 1-ft^2 Small Break LOCA

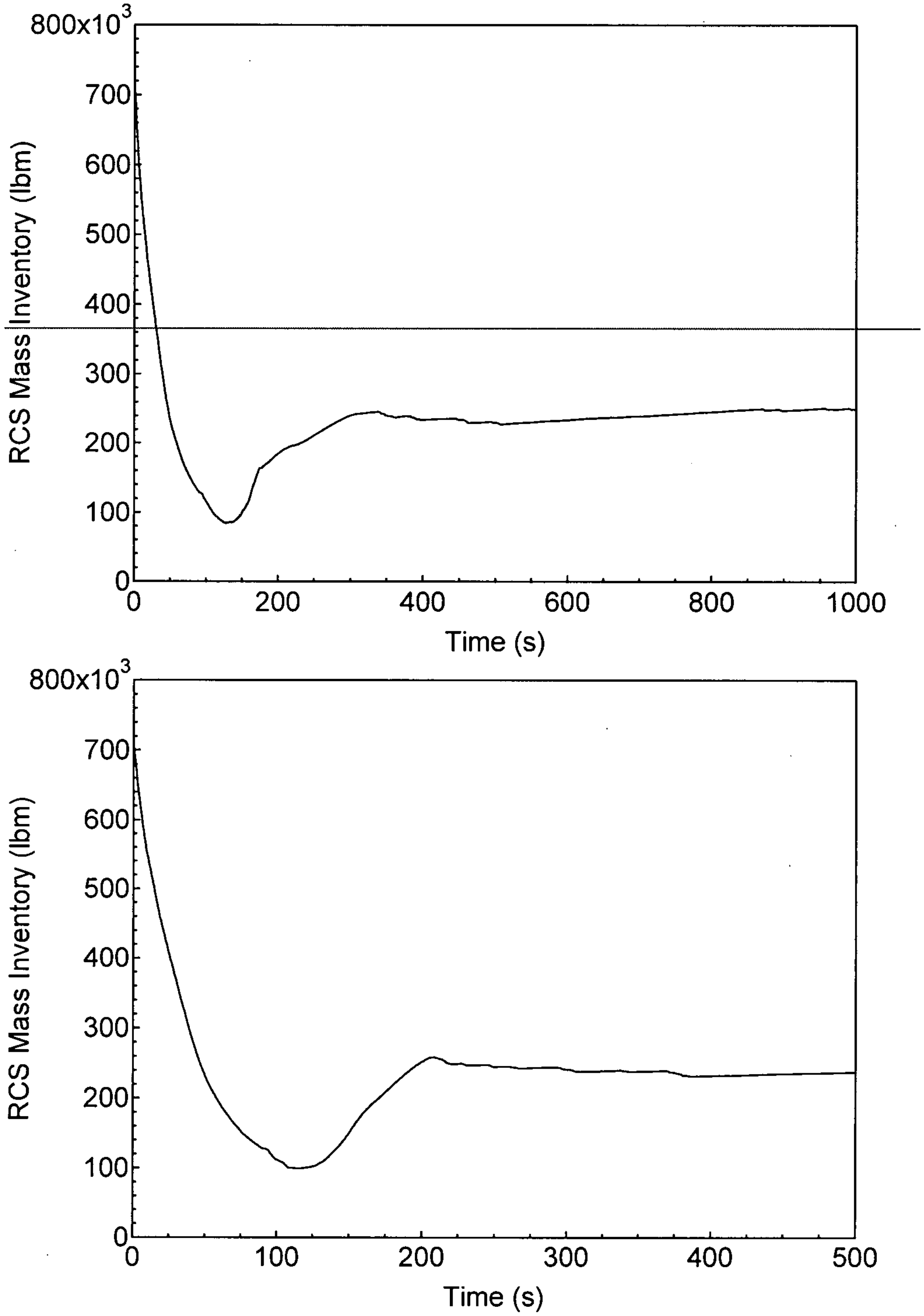


Figure 15.6.5-27 RCS Mass Inventory for 1-ft^2 Small Break LOCA

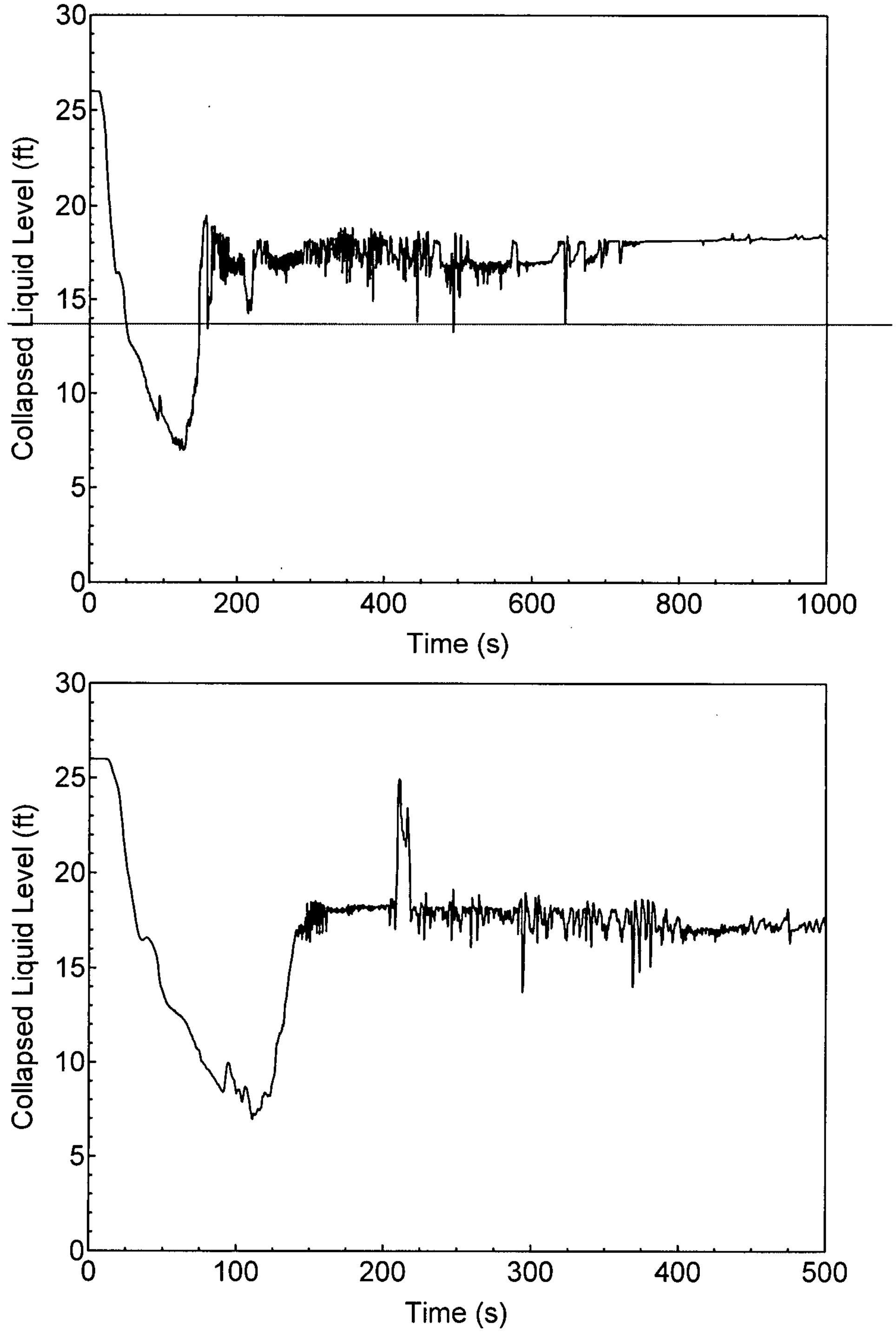
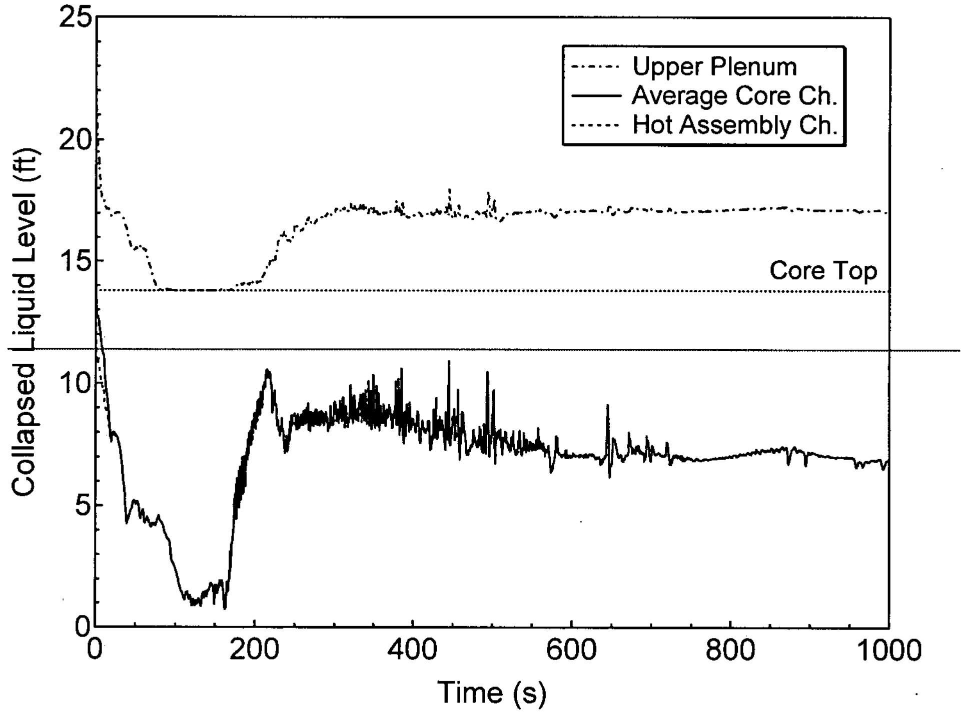


Figure 15.6.5-28 Downcomer Collapsed Level for 1-ft^2 Small Break LOCA



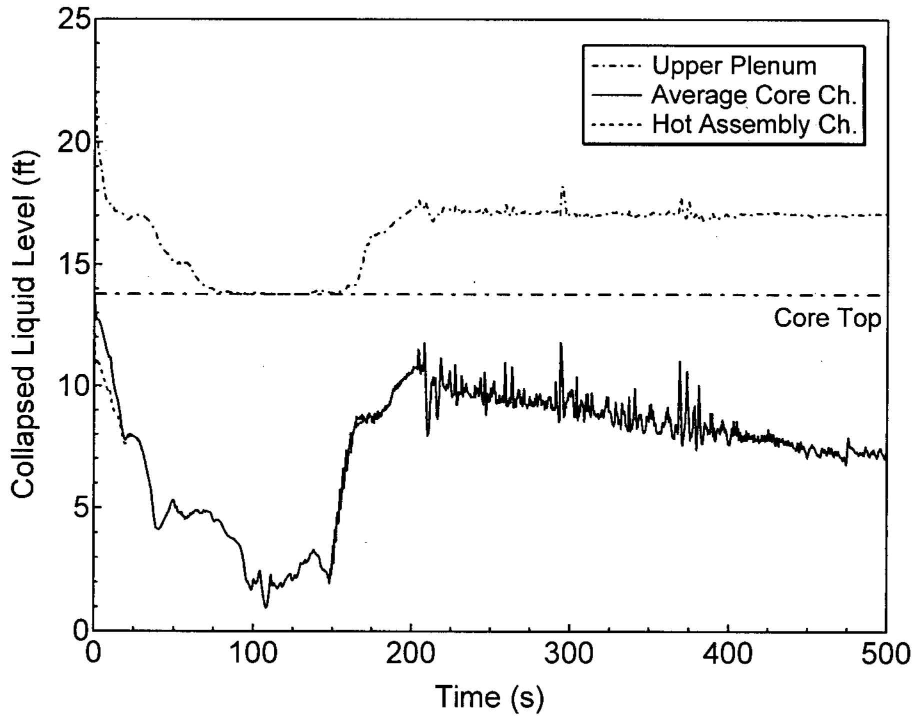


Figure 15.6.5-29 Core/Upper Plenum Collapsed Level for 1-ft^2 Small Break LOCA

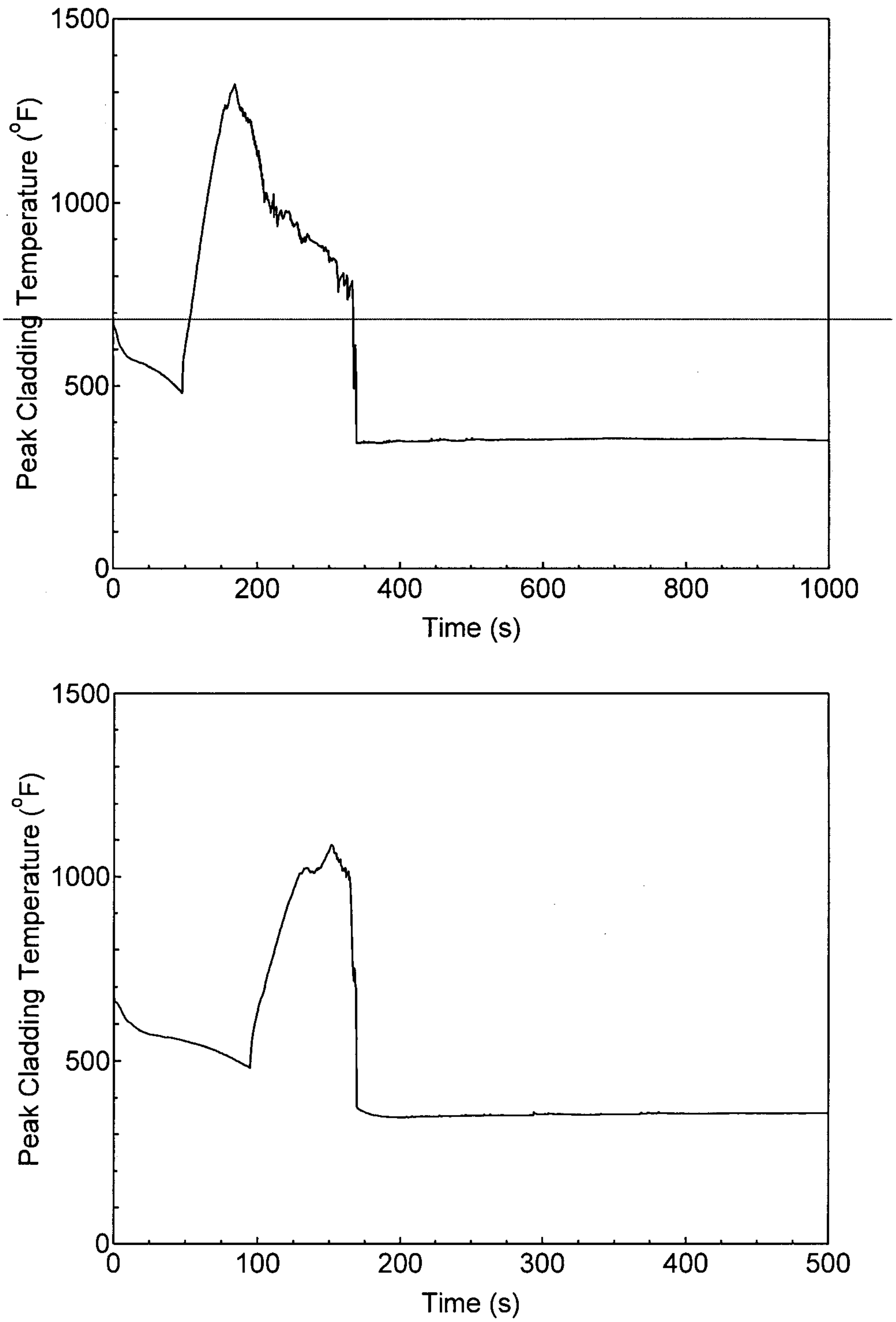


Figure 15.6.5-30 PCT at All Elevations for Hot Rod in Hot Assembly for 1-ft² Small Break LOCA

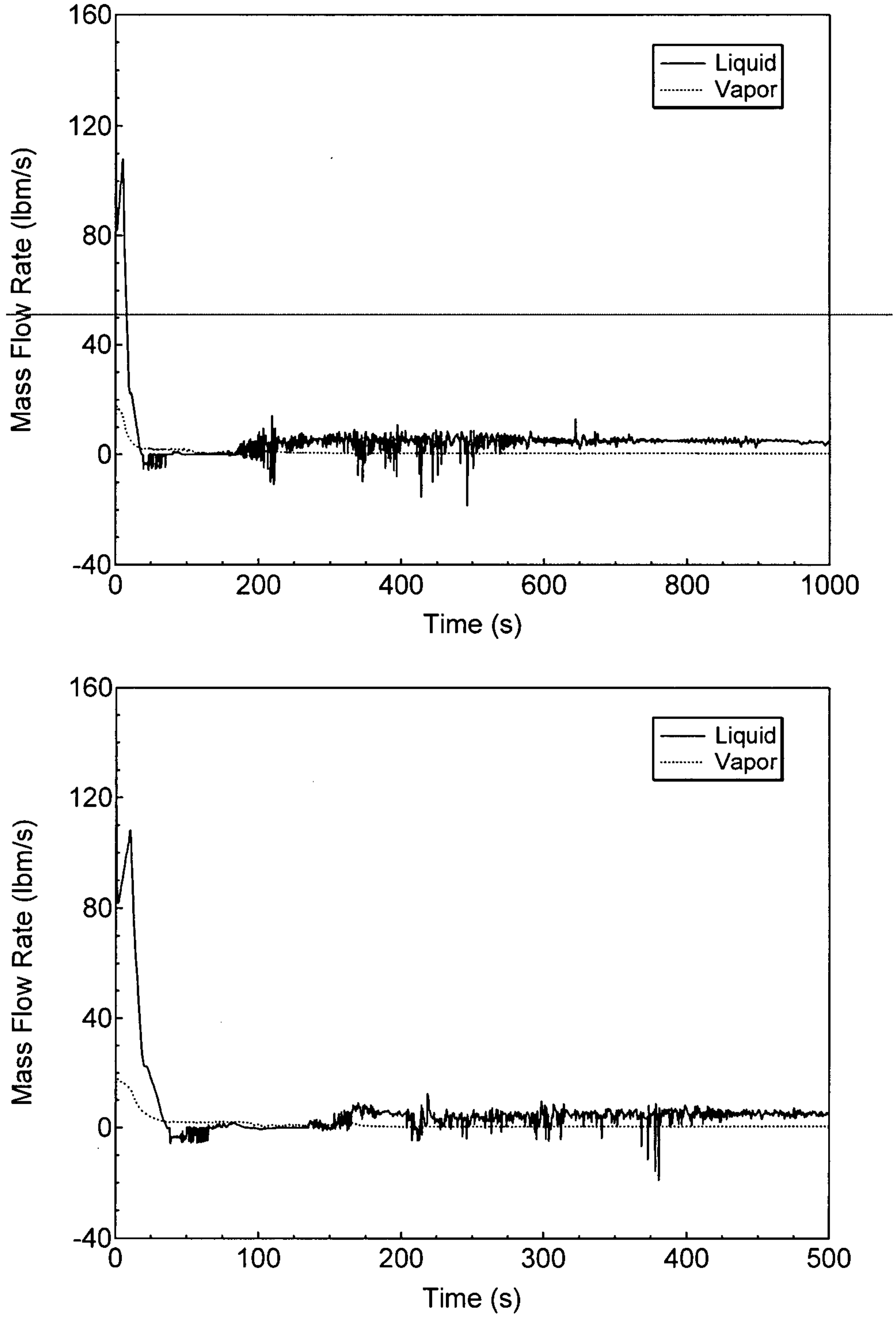


Figure 15.6.5-31 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 1-ft^2 Small Break LOCA

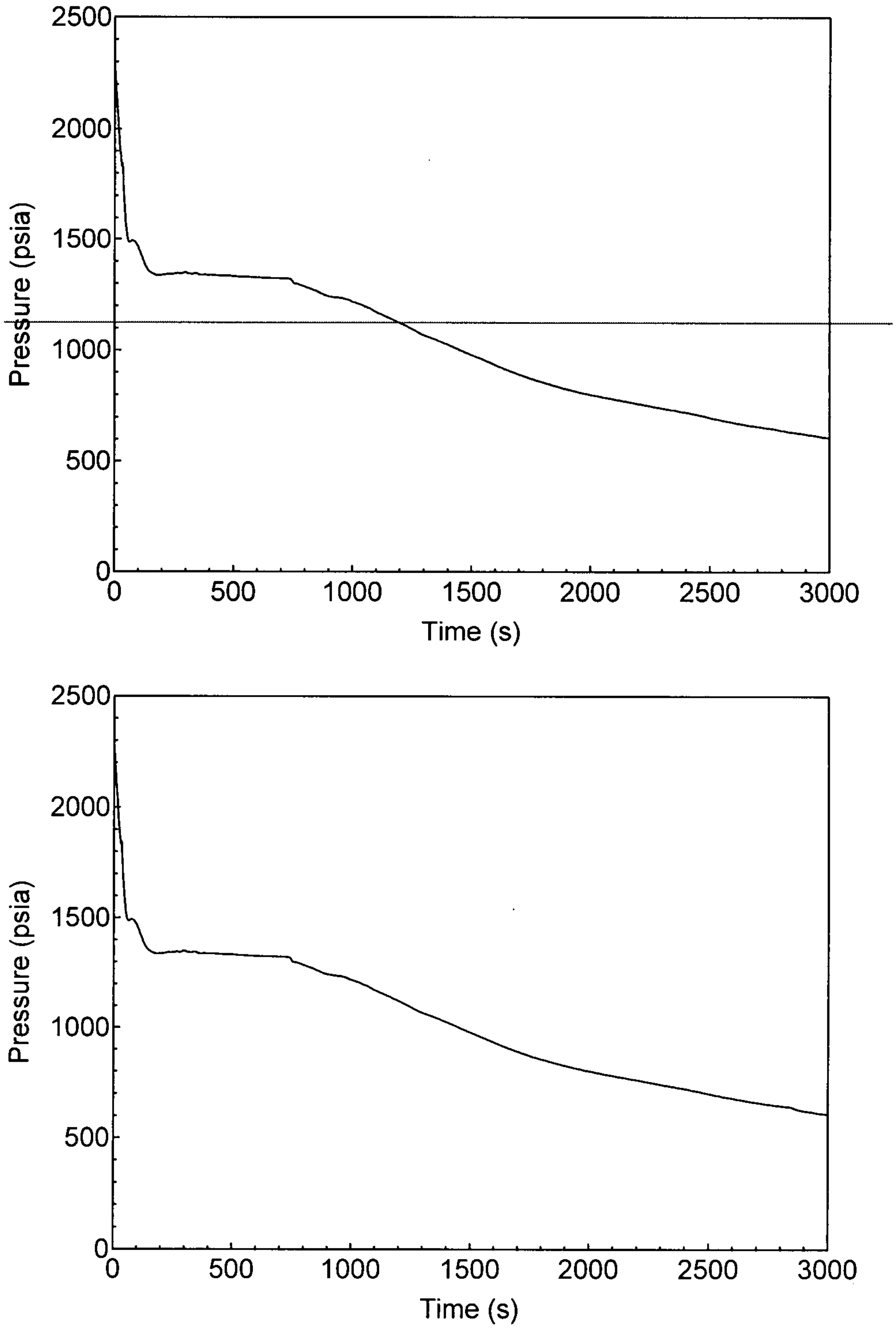
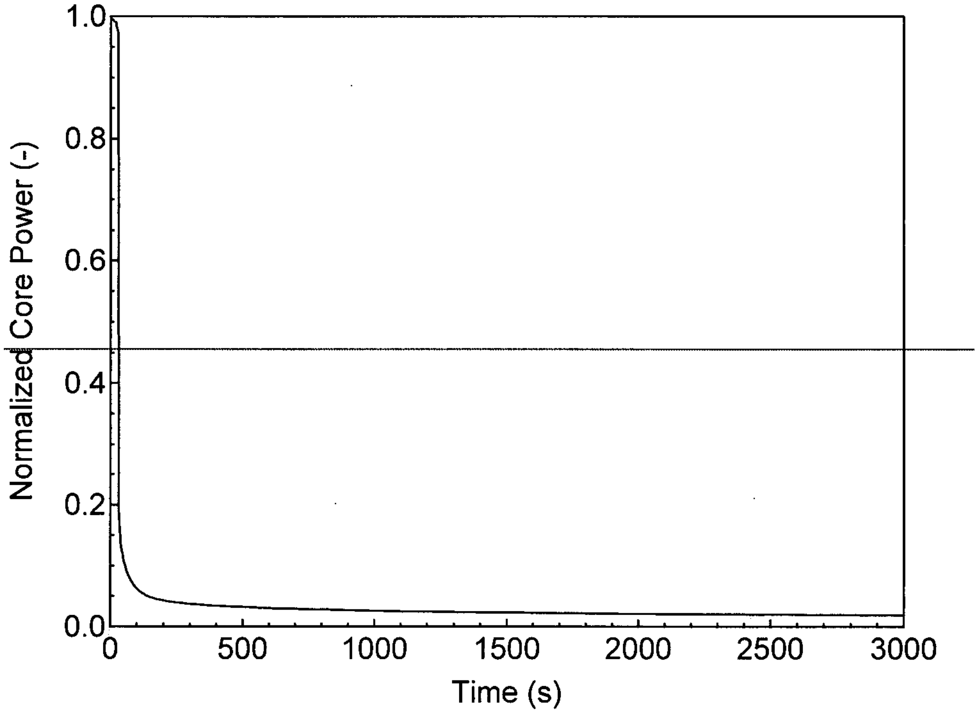


Figure 15.6.5-32 **RCS (Pressurizer) Pressure Transient for DVI-line Small Break LOCA**



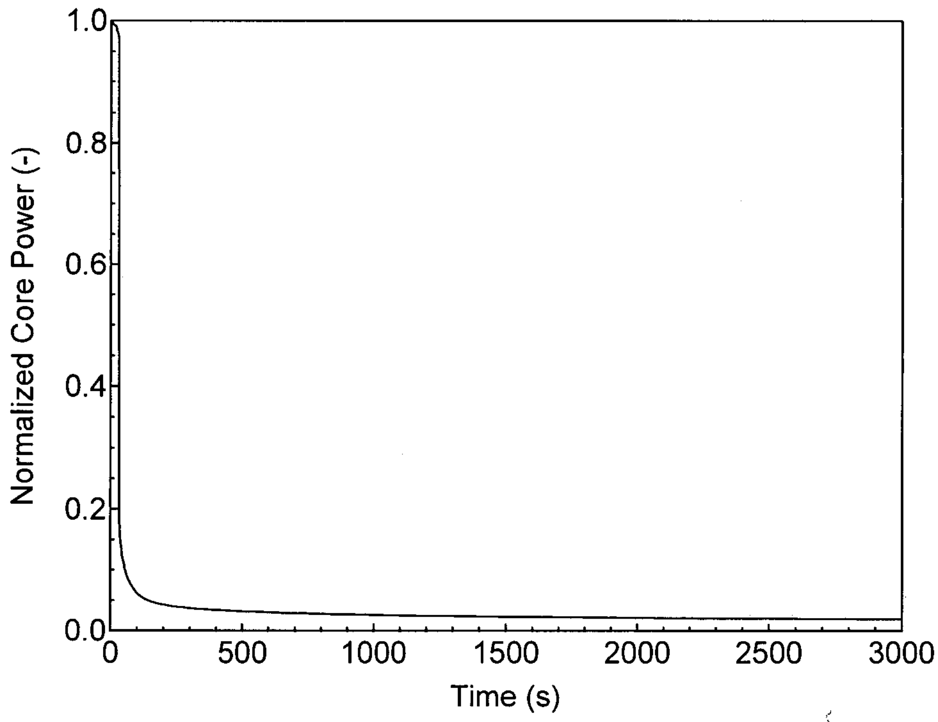
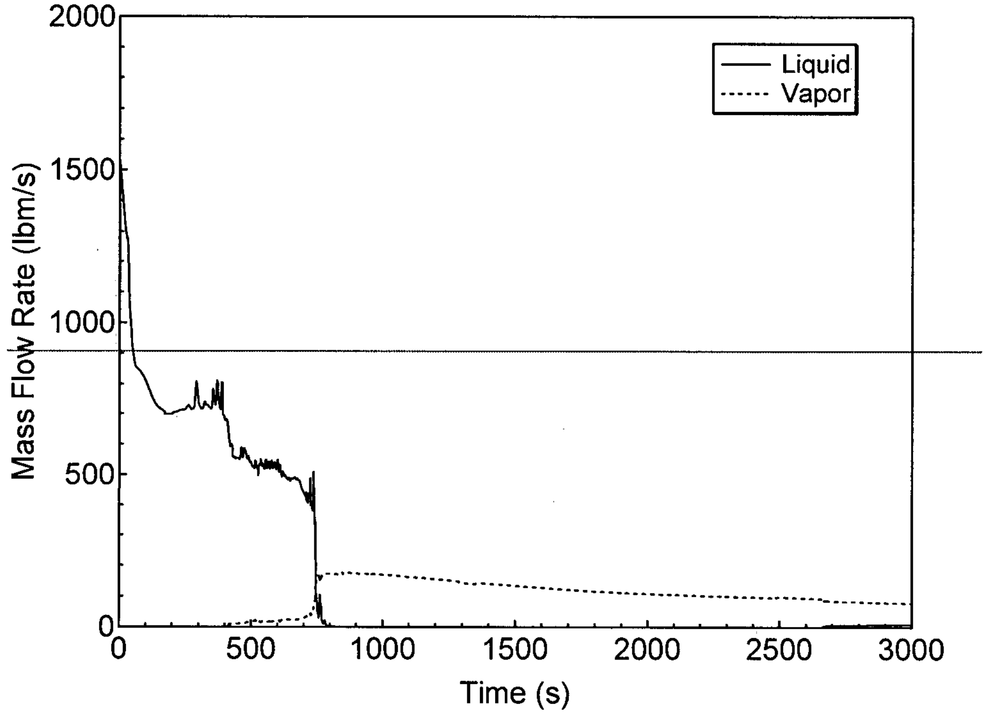


Figure 15.6.5-33 Normalized Core Power for DVI-line Small Break LOCA



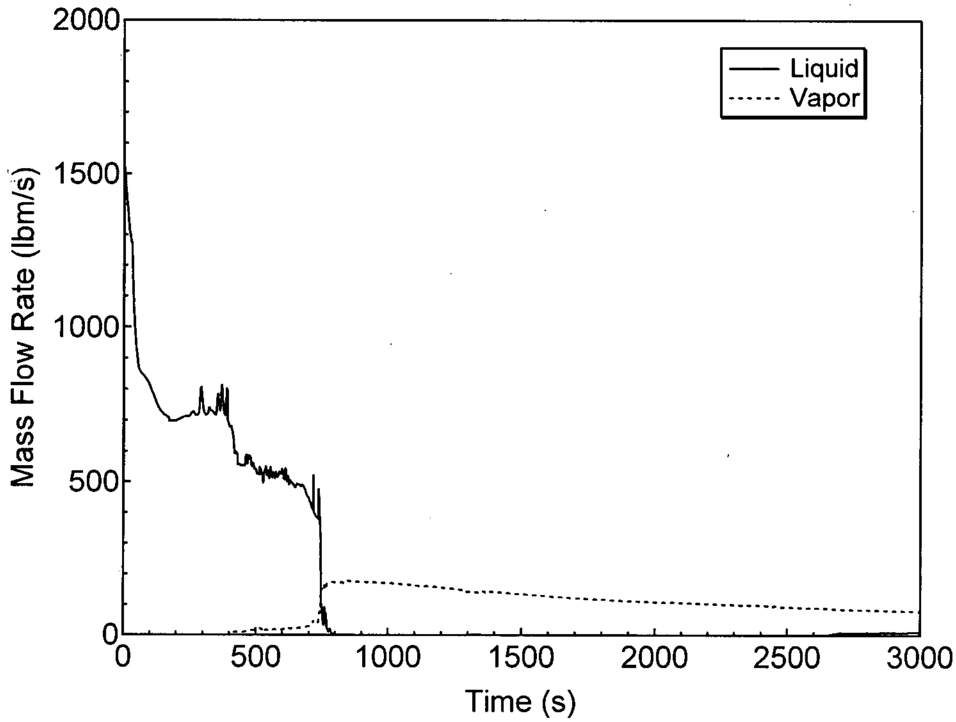


Figure 15.6.5-34 Liquid and Vapor Discharges through the Break for DVI-line Small Break LOCA

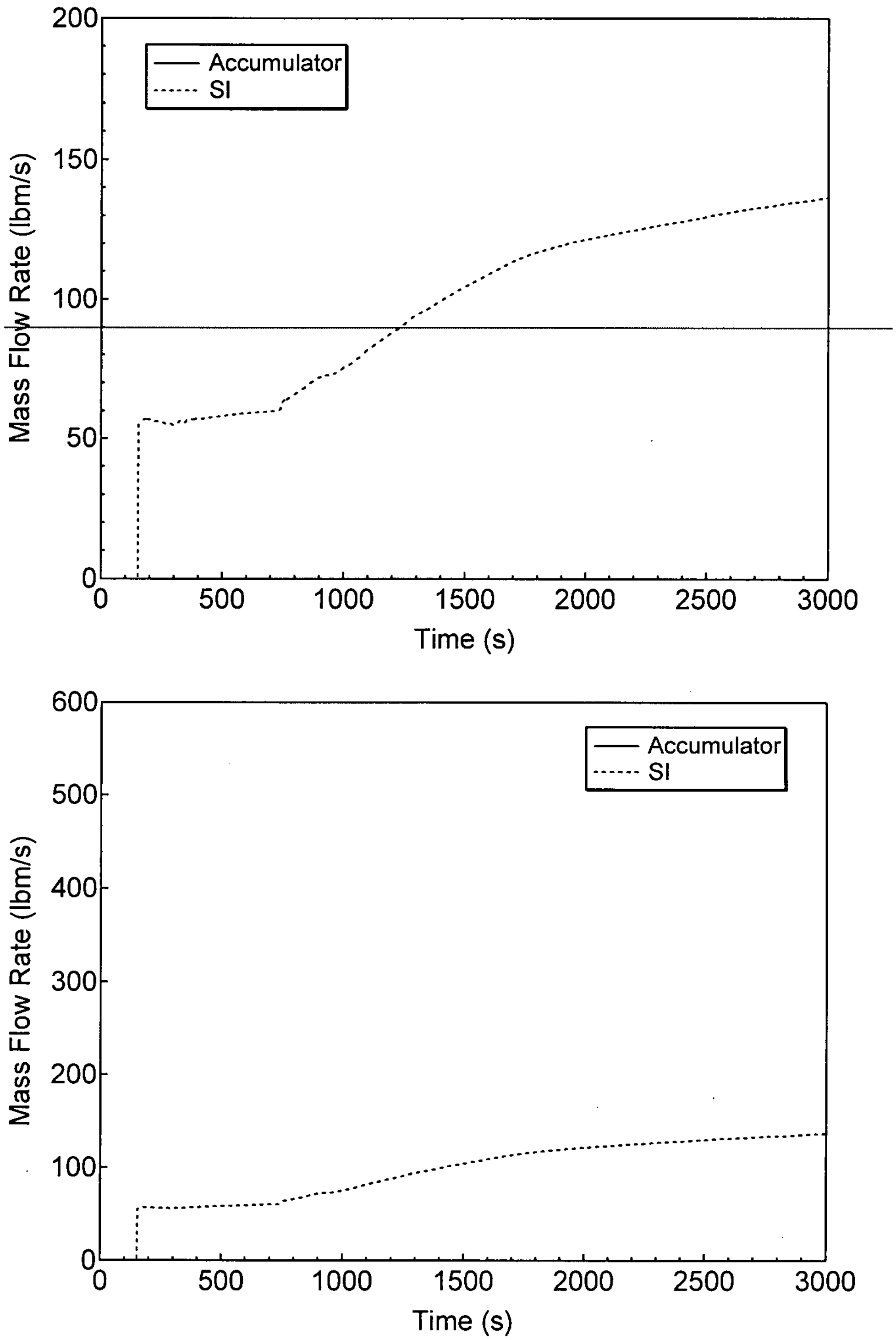
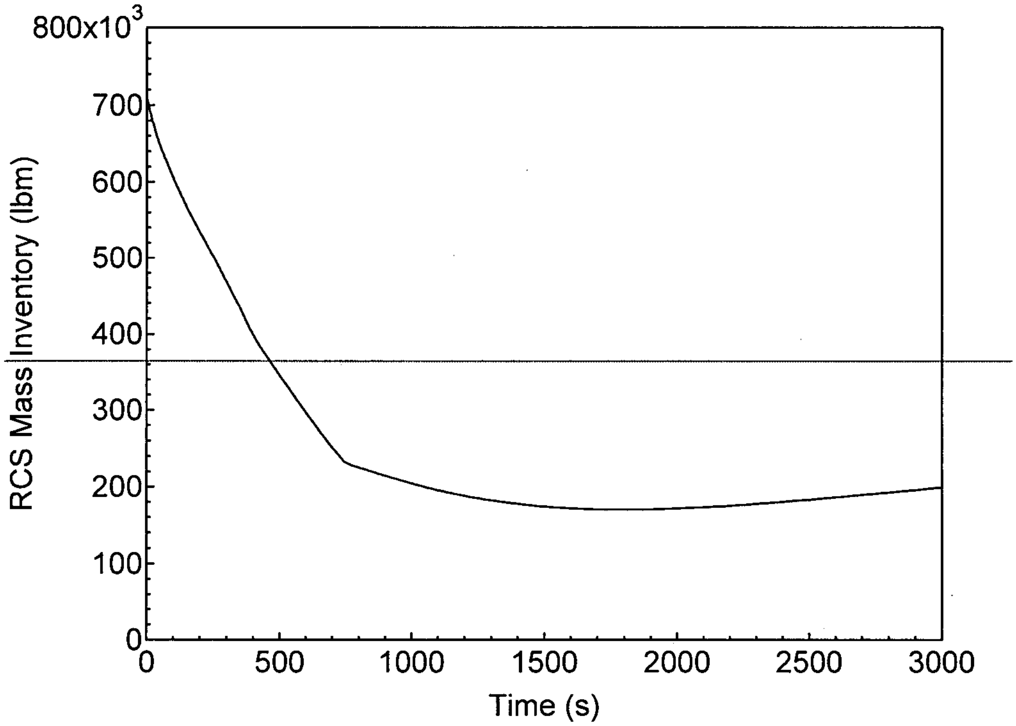


Figure 15.6.5-35 Accumulator and Safety Injection Mass Flowrates for DVI-line Small Break LOCA



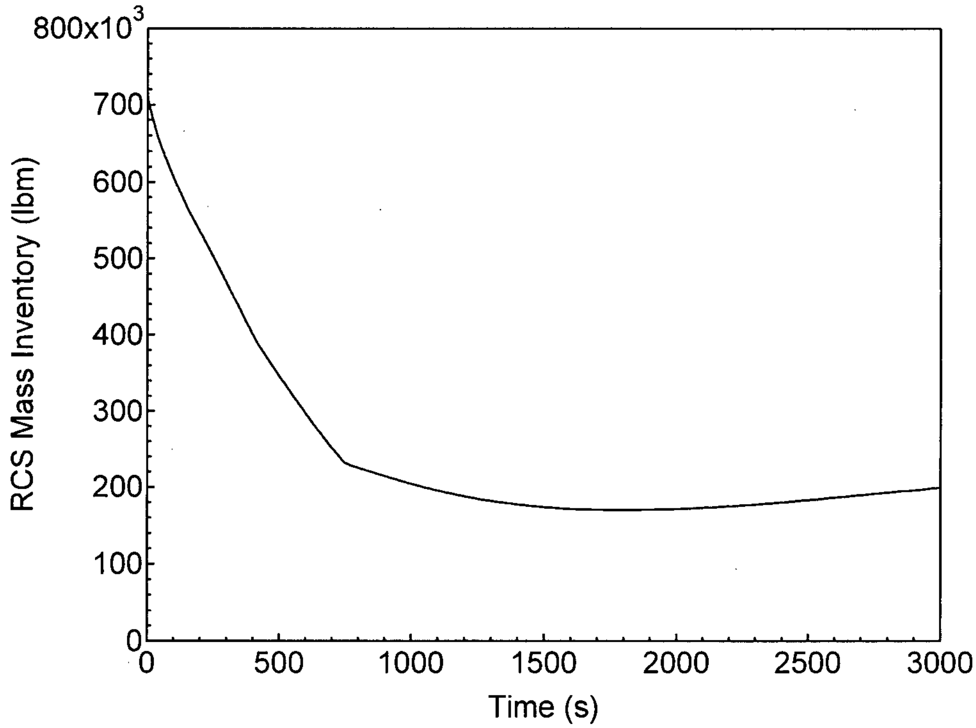
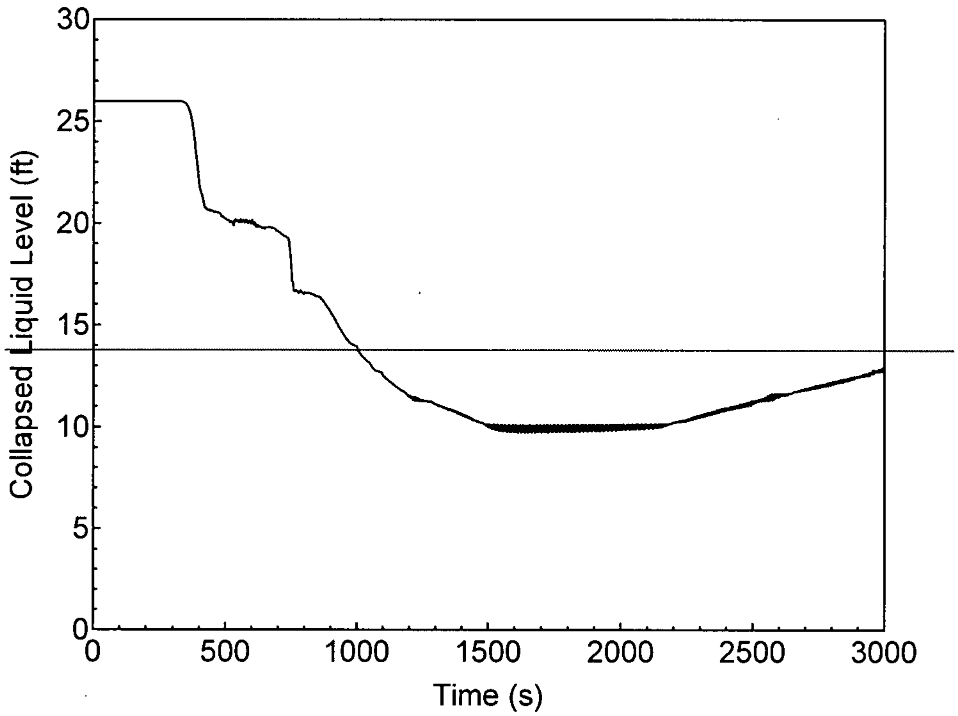


Figure 15.6.5-36 RCS Mass Inventory for DVI-line Small Break LOCA



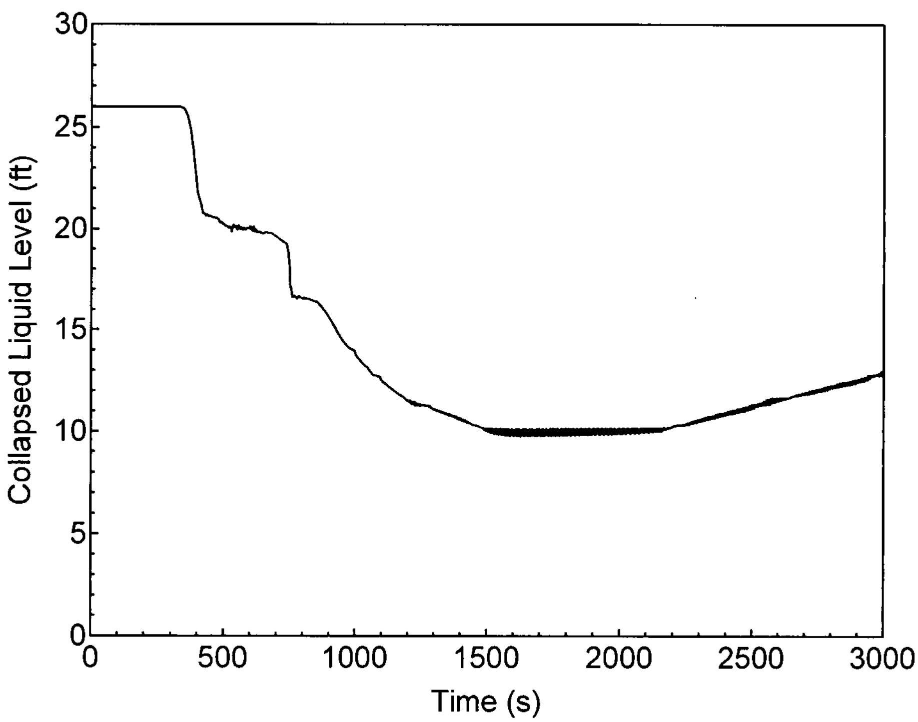


Figure 15.6.5-37 Downcomer Collapsed Level for DVI-line Small Break LOCA

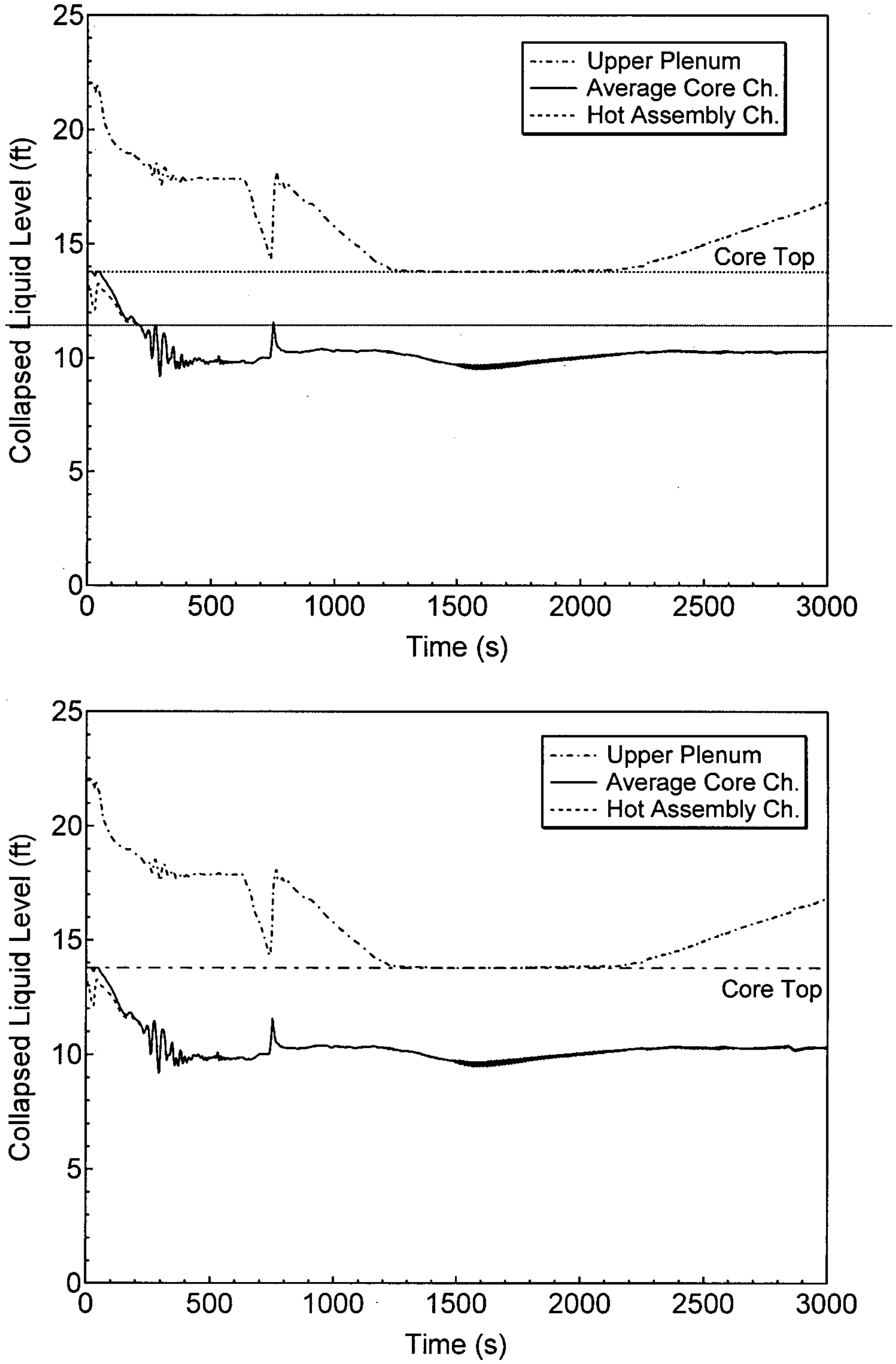


Figure 15.6.5-38 Core/Upper Plenum Collapsed Level for DVI-line Small Break LOCA

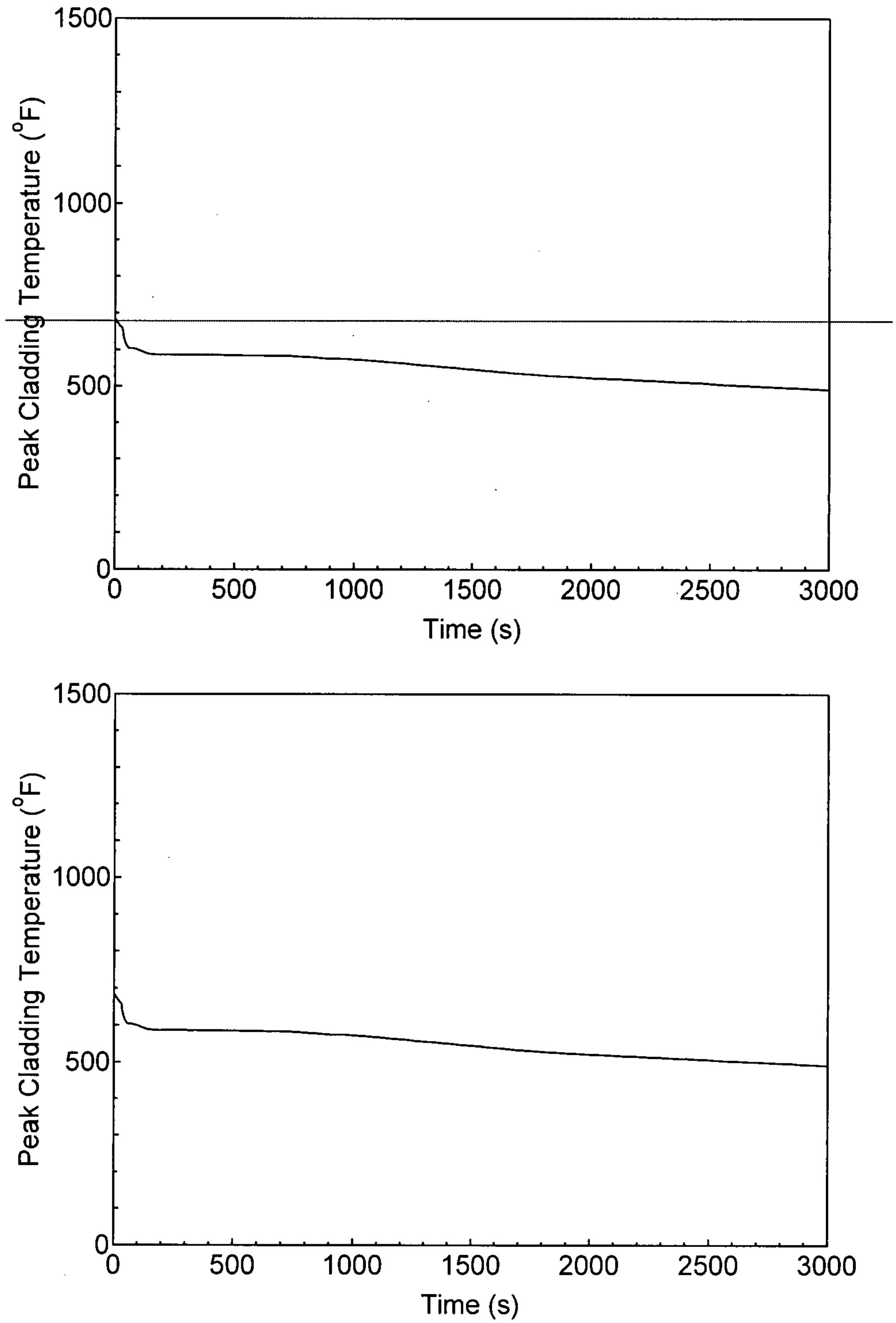


Figure 15.6.5-39 PCT at All Elevations for Hot Rod in Hot Assembly for DVI-line Small Break LOCA

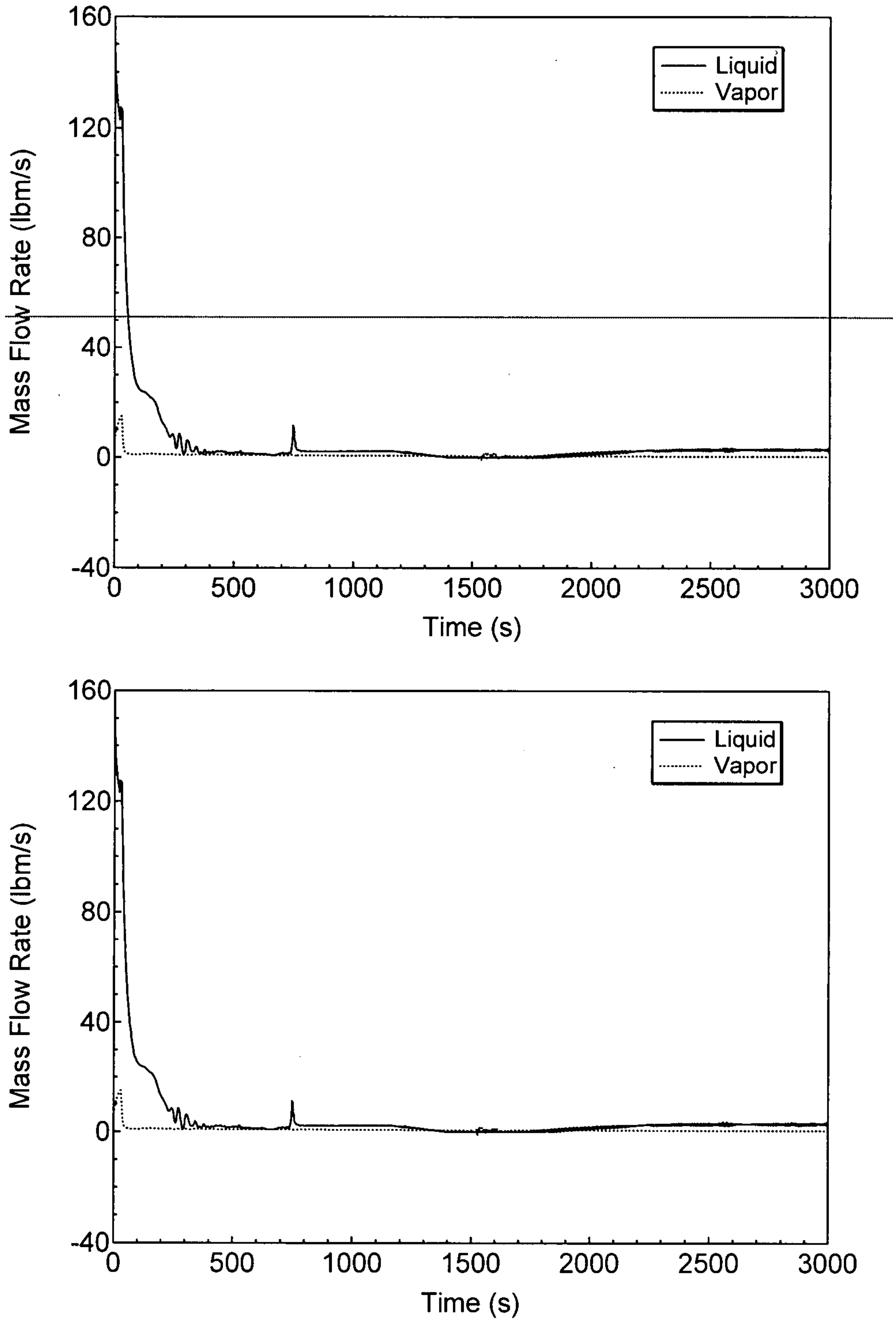


Figure 15.6.5-40 Hot Assembly Exit Vapor and Liquid Mass Flowrates for DVI-line Small Break LOCA

15.6.7 References

15.6-1 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.6-2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.6-3 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.6-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.6-5 Best-Estimate Calculations of Emergency Core Cooling System Performance, NRC Regulatory Guide 1.157, May 1989.

15.6-6 Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident, NUREG/CR-5249, 1989.

15.6-7 Code Qualification Document for Best Estimate LOCA Analysis, WCAP-12945-P-A (Proprietary), Volume I, Revision 2, and Volumes II-V, Revision 1, and WCAP-14747 (Non-Proprietary), Westinghouse Electric Company, March 1998.

15.6-8 Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method, WCAP-16009-NP (Non-Proprietary) Revision 0, Westinghouse Electric Company, May 2003.

15.6-9 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011 (Proprietary) and MUAP-07011-NP (Non-Proprietary), July 2007.

15.6-10 Beaver Valley Power Station, Unit Nos.1 and 2 Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173, FENOC letter dated July 8, 2005, to NRC.

15.6-11 Beaver Valley Power Station, Unit Nos.1 and 2 Responses to a Request for Additional Information (RAI dated September 30, 2005) in Support of License Amendment Request Nos. 302 and 173, FENOC letter dated November 21, 2005, to NRC.

15.6-12 Summary of August 23, 2006 Meeting with the Pressurized Water Reactor Owners Group (PWROG) to discuss the Status of Program to establish consistent Criteria for Post LOSS-OF-COOLANT (LOCA) Calculations, NRC Memorandum dated October 3, 2006, S.L. Rosenberg to S.E. Peters.

15.6-13 Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, "Post LOCA Long Term Cooling Model" due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit, NRC letter dated November 23, 2005, D.S. Collins to G.C. Bischoff.

15.6-14 Small Break LOCA Methodology for US-APWR, MUAP-07013-P (Proprietary) and MUAP-07013-NP (Non-Proprietary), ~~July 2007~~May 2010.

15.6-15 Guba, A., Makai, M., Lenard, P., Statistical Aspects of Best Estimate Method-I, Reliability Engineering and System Safety, Vol. 80, 217-232, 2003.

15.6-16 Small Break LOCA Sensitivity Analyses for US-APWR, MUAP-07025-P, ~~July 2007~~May 2010.

15.6-17 RSIC Computer Code Collection CCC-371, ORIGEN 2.2 Isotope Generation and Depletion Code - Matrix Exponential Method, June, 2002.

15.6-18 Containment Spray as a Fission Product Cleanup System, U.S. Nuclear Regulatory Commission, NUREG-0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants), Chapter 6.5.2 Revision 4, March 2007.

15.6-19 A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments, U.S. Nuclear Regulatory Commission, NUREG/CR-6189, July 1996.

15.6-20 Powers, D.A. and S.B. Burson, A Simplified Model of Aerosol Removal by Containment Sprays, U .S. Nuclear Regulatory Commission, NUREG/CR-5966, June 1993.

15.6-21 S.L Humphreys, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, U .S. Nuclear Regulatory Commission, NUREG/CR-6604, April 1998.

15.6-22 Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Regulatory Guide 1.52, Revision 3, June 2001.

15.6-23 Laboratory Testing of Nuclear Grade Activated Charcoal, U.S. Nuclear Regulatory Commission, Generic Letter 99-02, June 3, 1999.

15.6-24 ESF Atmosphere Cleanup System, U.S. Nuclear Regulatory Commission, NUREG 0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants), Chapter 6.5.1 Revision 3, March 2007.

15.6-25 Grove Software, Inc. MicroShield: A comprehensive photon/gamma ray shielding and dose assessment program.

15.6-26 Wilks, S.S, Determination of Sample Sizes for Setting Tolerance Limits, The Annals of Mathematical Statistics, Vol.12, pp. 91-96, 1941.

15.6-27 H. C. Yeh, Modification of Void Fraction Calculation, Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, June 6,1988.

15.6-28 Cohen, P., Water Coolant Technology of Power Reactors, Chapter 6, Chemical Shim Control and pH Effect, ANS-USAEC, 1969.