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 flow decreases earlier and the safety injection (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.

As a result of the break, the coolant from RCS is rapidly lost, the cooling capability for the reactor core is reduced, and the RCS pressure decreases rapidly. The reactor trip and subsequent borated water injection from the accumulator complement void formation in causing a rapid reduction to a power level corresponding to fission product decay heat.

The large break LOCA is generally divided into three phases in which specific phenomena are occurring. They are the blowdown phase, refill phase, and reflood phase.

Blowdown phase

The blowdown phase covers the period from the initial pipe break to the time where the RCS pressure is equal to the containment pressure.

Initially, subcooled liquid is discharged through the break at a rate that exceeds the capacity of the RCPs in coast down mode. As a result, core flow reverses and the fuel rods go through DNB resulting in rapid cladding heat up. Reactor power decreases due to voiding in the core. Water flashes to steam starting in the upper plenum and core and continuing to the lower plenum and downcomer.

In the early stage of the blowdown, the RCPs in the intact loops are still delivering single-phase liquid to the core. As a result, there will be a temporary upward flow through the core. As the loops become two-phase, the RCP performance degrades. The cooling effect due to upward flow may not be sufficient if the break is large and the pumps performance degrades rapidly.

As the RCPs driven flow decreases, the break flow begins to dominate and core flow reverses again. Liquid, entrained liquid, and steam flows provide core cooling. As the RCS pressure continues to fall to the containment pressure, the break flow and core flow are reduced. Consequently, the core begins to heat up. When the RCS pressure drops below the accumulator injection pressure, borated water is injected into the vessel.

Refill phase

The refill phase starts from the end of blowdown phase until the lower plenum is refilled up to the bottom of the core. During this phase the core experiences a nearly adiabatic heatup as the lower plenum is filled with borated water supplied by the accumulators of the safety injection system (SIS). The accumulators operate in the high flow rate mode in this phase, which is similar to existing conventional PWR. The accumulator flow is sufficient to fill the downcomer and initiate reflood of the core.

Reflood phase

The reflood phase covers the period from the end of the refill phase to final quenching of the core. The accumulators automatically switch from the high to low injection rate as the water level in the accumulators fall. Core cooling function is maintained by the small injection flow rate and flow from the SI pumps. The injected borated water begins to quench the lower part of the core. As the quench front progresses, the location of the highest cladding temperature moves higher in the core. Eventually, the entire core is covered with a two-phase mixture and cooled.

(1) Reactor Trip Signals

A reactor trip signal occurs due to one of the following signals for this event:

- Low pressurizer pressure
- Low reactor coolant flow
- Over temperature ΔT

(2) Engineered Safety Features Actuation Signals

The engineered safety features (ESF) actuation signals are comprised of the ECCS actuation signals, the main steam line isolation signals, the containment vessel isolation signals, and the containment vessel spray actuation signals.

The ECCS actuation signal is actuated on one of the following signals for this event:

- High containment pressure
- Low pressurizer pressure

The main steam line is isolated on one of the following signals for this event:

- High main steam line pressure negative rate
- High-high containment pressure

The containment spray system (CSS) is actuated on the high-3 containment pressure signal.

The containment is isolated on one of the following signals for this event:

- ECCS actuation signal
- Containment spray actuation signal

The turbine trips automatically following the reactor trip. The RCPs trip automatically, initiated by both the ECCS actuation signal and the reactor trip with a delay time. RCP coastdown occurs in the blowdown phase.

After the reactor and turbine trips, heat from the core, hot internals, and the vessel continue to be transferred to the coolant and then to the secondary system. Since the secondary system heat sink is temporarily lost due to the turbine trip, the secondary system pressure increases. In the case of LOOP, the emergency power source (EPS) supplies electrical power to the essential components of the ECCS. Hence, the design functions are maintained.

As a result of a high containment pressure, the main steam lines are automatically isolated. After the isolation of main feedwater system, the ECCS actuation signal initiates flow to the secondary side by starting the emergency feedwater (EFW) pumps.

(3) Emergency Core Cooling System Functions During a LOCA

The US-APWR ECCS consists of the accumulator system, the high head injection system (HHIS) and emergency letdown system. The ECCS injects borated water into the RCS following a postulated LOCA to cool the reactor core, to prevent damage to the fuel cladding, and to limit the zirconium-water reaction of the fuel cladding to a very small amount.

Each of the four RCS loops has an accumulator connected to the respective cold leg. When the RCS pressure falls below the accumulator initiating pressure of 600 psia, the accumulators begin to inject borated water into the RCS cold legs. Each accumulator has an internal passive flow damper, which automatically switches the injection flowrate. When the water level is above the top of a standpipe within an accumulator, water enters the flow damper through both inlets at the top of the standpipe and at the side of the flow damper and thus the accumulator injects water with a large flowrate. When the water level drops below the top of the standpipe, the water enters the flow damper only through the side inlet and thus injects water at a lower flowrate. The accumulators are attached to the cold legs.

The HHIS consists of four independent safety trains, each containing an SI pump and the associated valves and piping. The SI pumps are aligned to take suction from the refueling water storage pit (RWSP) and deliver borated water directly to the downcomer through the direct vessel injection (DVI) nozzles located below the cold leg inlet nozzles on the RV. The RWSP is located within the lowest portion of the containment vessel and collect the water from the break and the containment sprays. The RWSP provides a continuous borated water source for the SI pumps avoiding the need to switch the pump suction from a storage water tank to the containment recirculation sump. The SI pumps start automatically upon receipt of the ECCS actuation signal.

The accumulators initially inject large flow rate, then automatically reduced to lower flow rate as the water level in the accumulators drop below the level of the internal standpipe. The reduced flow from the accumulators, together with the DVI flow from the SI pumps is sufficient to maintain the downcomer level provide flow to the core during the reflood phase. The combined performance of the accumulator system and the HHIS is sufficient to eliminate the need of low head injection pumps.

(4) Containment Spray System Functions During a LOCA

The containment spray system (CSS) consists of four independent trains, each containing a containment spray/residual heat removal (CS/RHR) heat exchanger, a CS/RHR pump, spray nozzles, piping and valves. The CSS takes borated water taken from the RWSP 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 calculations and that this uncertainty be included when comparing the results of the calculations to the acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 (Ref. 15.6-5).

The code scaling, applicability, and uncertainty (CSAU) evaluation methodology (Ref. 15.6-6) presented an approach for applying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis. This methodology has been applied to three and four-loop PWR plants (Ref. 15.6-7) using a response surface technique for the uncertainty treatment.

The Automated Statistical Treatment of Uncertainty Method (ASTRUM), was developed using the WCOBRA/TRAC code (Ref. 15.6-8). This methodology uses a statistical sampling method, in which all parameters are simultaneously varied. The necessary number of cases to calculate the 95th percentile PCT, LMO, or CWO with 95% confidence is determined based on statistical theory. The ASTRUM is used for the US-APWR large break LOCA analysis.

WCOBRA/TRAC (M1.0) Evaluation Model

The WCOBRA/TRAC (M1.0) code is a modified version of WCOBRA/TRAC. The applicability of the modified code for the US-APWR large break LOCA analysis is discussed in the Topical Report (Ref. 15.6-9).

The WCOBRA/TRAC code combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift flux equations used in the loops to allow a complete and detailed simulation of a PWR. Also the WCOBRA/TRAC code has ability to represent important reactor components such as fuel rods, steam generators, RCPs and so on.

The main confirmation points of applicability to the US-APWR are as follows:

- Empirical correlations to model the advanced accumulator characteristics are included.
- Metal heat release and bypass flow within the neutron reflector is modeled as a separate channel with heat structure.

ASTRUM Analysis Process

The confirmatory calculations are performed before the uncertainty evaluation to set the limiting data for some parameters based on a plant-specific basis. These parameters are identified from the conventional PWR sensitivity studies as potential contributors to uncertainty. They are categorized into three groups: nominal without uncertainty, bounded, and nominal with uncertainty. The results of the confirmatory calculations are used to define conditions for reference transient calculation.

The reference transient calculation is performed to evaluate the typical large break LOCA characteristics. The reference transient calculation incorporates the nominal values for 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

<u>**Fundamental Calculation Method**</u>

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.

(3) Calculation Procedure

Boric acid concentration is calculated by the following procedure:

- RV injection mode

a. Core evaporation rate and void fraction are calculated.
b. The mixing volume makeup flow rate that compensates for the core evaporation and reduces void fraction is calculated.
c. Boric acid concentration in the mixing volume is calculated by the following equation:

$$CB_{MV} = \frac{MB_{MV} + \left(W_{makeup} \times CB_{RWSP}\right) \times dt}{MF_{MV} + \left(W_{makeup} - W_{boil}\right) \times dt}$$

where

CB_{MV}	Boric acid concentration in the mixing volume
MB_{MV}	Boric acid mass in the mixing volume
MF_{MV}	Boric acid solution mass in the mixing volume
CB_{RWSP}	Boric acid concentration in the RWSP volume
W_{makeup}	Mixing volume makeup flow rate
W_{boil}	Core evaporation rate
dt	Time step

d. Then, the boric acid concentration in RWSP volume is calculated by

$$CB_{RWSP} = \frac{MB_{RWSP} - \left(W_{makeup} \times CB_{RWSP}\right) \times dt}{MF_{RWSP} - \left(W_{makeup} - W_{boil}\right) \times dt}$$

Where

MB_{RWSP}	Boric acid mass in the RWSP
MF_{RWSP}	Boric acid solution mass in the RWSP

At a certain time, this RV injection mode is switched over into the simultaneous RV and hot leg injection mode.

- Simultaneous RV and hot leg injection mode

a. Core evaporation rate and void fraction are calculated.
b. All hot leg injection water flows into the mixing volume. Then, boric acid concentration of the mixing volume is calculated as follows:

$$CB_{MV} = \frac{MB_{MV} + \left(W_{hotleg} \times CB_{RWSP}\right) \times dt}{MF_{MV} + \left(W_{hotleg} - W_{boil}\right) \times dt}$$

where

W_{hotleg}: Hot leg injection flow rate

c. Mixing volume flushing flow rate is calculated by

$$W_{flush} = W_{hotleg} - \left\{W_{boil} + \left(dMF_{MV} / dt\right)\right\}$$

where

W_{flush} Mixing volume flushing flow rate

dMF_{MV} The increase of liquid mass caused by the reduction of void fraction in one time step.

d. Next, boric acid concentration in the RWSP is calculated by

$$CB_{RWSP} = \frac{MB_{RWSP} - (W_{hotleg} \times CB_{RWSP} - W_{flush} \times CB_{MV}) \times dt}{MF_{RWSP} - (W_{hotleg} - W_{flush} - W_{boil}) \times dt}$$

Boric acid concentration in the core flushing flow is the same as that in the mixing volume.

Range of Mixing Volume

To specify the mixing volume, the following assumptions are used:

- Mixing volume consists of core, upper plenum and lower plenum.
  - All volumes of the core region are included in the mixing volume.
  - Upper plenum volume below hot leg bottom elevation is included in the mixing volume.
  - Half of the lower plenum is included in the mixing volume.
- Mixing volume does not include any volume of neutron reflector region.

Decay Heat

The decay heat of 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards: "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors", October 1971) is used in accordance with 10CFR50 Appendix K requirements.

Borated Water Source

The RWSP, accumulator, and RCS are considered as the sources of borated water. The initial boric acid concentration is assumed to be the maximum allowed for operating conditions. The water mass in the RWSP and accumulators is assumed to be the maximum allowed for operating conditions, because large quantity of borated water in the sources yield higher concentration of boric acid in the mixing volume. A minimum amount of RCS water mass is assumed because boric acid concentration in the RCS is lower than that in the RWSP and accumulators. Boric acid concentration is also considered in calculating liquid mass density.

Effects of System Pressure

The effects of system pressure are as follows.

- Higher system pressure gives a lower void fraction in the core and consequently, more water mass in the mixing volume.
- Higher system pressure increases the boiling rate because of the decrease in the

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

The Result of Reference Transient Calculation

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.

The core pressure transient is illustrated in Figure 15.6.5-3. Following the break, the vessel rapidly depressurizes during the subcooled break flow. The pressure reduction rate then decreases as boiling begins in the vessel and the break flow becomes two-phase. As the RCS pressure falls and approaches the containment atmosphere pressure, the break and core flows reduce accordingly. The blowdown phase ends at 34 seconds.

(2) Refill phase

Figure 15.6.5-4 presents the transient of liquid level in the lower plenum. During the refill phase, core heat up occurs because the primary heat transfer mechanism is convection to steam. The lower plenum is filled with borated water supplied by the accumulators. At approximately 37 seconds, the lower plenum fills to the bottom of the core, which ends the refill period and begins the reflood period.

(3) Reflood phase

The reflood phase starts 37 seconds from the beginning of the break. In this phase, coolant enters the core from the bottom, and the core collapsed liquid level increases. The transient of collapsed liquid level in each of the four downcomer quadrants is presented in Figure 15.6.5-5. The collapsed liquid level in each of four core channels is shown in Figure 15.6.5-6.

The accumulator water level drops below the top of the internal standpipe at 56 seconds and switches from high to low flowrate injection. The accumulator and SI system flow rate transients are shown in Figure 15.6.5-7. PCT occurs at 64 seconds at about 10-ft elevation. Steam generation and liquid entrainment in the flooded portion of the core help to cool the upper part of the core and reduce the cladding temperature.

At 124 seconds, the SI pumps start to inject water into the vessel. By 190 seconds, the cladding temperature at the PCT location has gradually decreased to the point of minimum film boiling temperature. Then, the temperature rapidly decreases to the saturation temperature at about 190 seconds. This ends the reflood phase at about 220 seconds, in which the core is recovered by water.

<u>ASTRUM Results and Comparison with the 10 CFR 50.46 Criteria</u>

A series of WCOBRA/TRAC calculations are performed to determine the 95th percentile PCT, LMO and CWO with 95% confidence. This PCT and other parameters are accomplished by performing 124 ASTRUM runs by randomly selecting from those parameters that are allowed to vary in the reference transient (see Table 15.6.5-1). The same WCOBRA/TRAC runs are used to obtain the 95th percentile at 95% confidence for the LMO and the CWO.

Figures 15.6.5-9 through 15.6.5-12 depict the limiting case values of PCT, LMO, and CWO with 95th percentile and 95% confidence.

Figure 15.6.5-8 shows the axial power shape operating space envelope used by the ASTRUM methodology. In the figure, PBOT is the integrated power fraction in the lower 3rd of the core, while PMID means the integrated power fraction in the middle 3rd of the core.

Figure 15.6.5-9 shows the PCT scatter plot as a function of the effective break area. The effective break area is calculated by multiplying the coefficient of discharge (C_D) with the sample value of the break area, normalized to the cold leg cross sectional area. The C_D is implemented to account for the uncertainty of the break flow model. The PCT is a conservative estimate of the 95th percentile PCT with a 95% confidence level. The figure shows cases for both DECLG and split breaks. The limiting PCT transient corresponds to the DECLG breaks. The figure clarifies that the DECLG break is found to be more limiting than the limiting size split break.

Figure 15.6.5-10 shows the cladding temperature transient of limiting PCT case, which is predicted with Run 48 and is equal to 1758°F. Table 15.6.5-7 lists the sequence of events for the limiting case large break LOCA. The PCT occurs at 75 seconds during the reflood phase. After reaching the PCT, core reflooding progresses and cladding temperature decreases. The cladding at the PCT location is quenched at 170 seconds. Finally, the whole core quenching is established at 220 seconds.

Figure 15.6.5-11 shows the cladding temperature transient at the limiting elevation for the LMO limiting case. The PCT values corresponding to the CWO is plotted against time in Figure 15.6.5-12. As a conservative approach, the value of CWO is selected as the most limiting oxidation value for the rod within the hot-assembly. Table 15.6.5-8 presents the calculated 95th percentile PCT, LMO, and CWO.

Based on the above analysis, the requirements of 10 CFR 50.46 are satisfied, and summarized as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F. The 95th percentile result of 1758°F (Run 48) presented in Table 15.6.5-8 indicates that this regulatory limit is met.

2. The calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation. The 95th percentile result of 2.8% (Run 103) maximum local cladding oxidation presented in Table 15.6.5-8 indicates that this regulatory limit is 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 95th percentile result of below 0.2% (Run 101) maximum core wide cladding oxidation presented in Table 15.6.5-8 indicates that this regulatory limit is met.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. 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 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~~ 78 seconds. The upper region of the core begins to uncover at ~~122~~ 124 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~~ 145 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~~761°F occurs at ~~136~~ 137 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~~761°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~~1302°F occurs at ~~169~~ 161 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~~1302°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~~of 789°F occurs at 1505 seconds.~~in the DVI-line break, indicating that the core keeps covered throughout the transient.~~ The PCT is significantly lower than 2200°F.

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. The PCT of 789°F presented in Table 15.6.5-14 indicates that this regulatory limit has been met.~~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 at 120 psia. Small break evaluation is the same as that used for the large break LOCA , except for the assumed system pressure.

Figure 15.6.5-43 shows the calculated time-history of the core boric acid concentration. The solid line indicates that the gradual increase of boric acid concentration is terminated by the switchover performed at four hours, before the precipitation limit is reached. Accordingly, the boric acid concentration reduces. The dashed line implies that the boric acid concentration continues to increase if the switchover were not carried out at four hours. Figure 15.6.5-43 also shows the dilution effect of the hot leg injected flow after the switchover.

<u>**Core Cooling after Switchover to Hot Leg**</u>

Evaluation is also performed to clarify the effect of early switchover from RV injection mode to the simultaneous RV and hot leg injection mode. If switchover is performed too early, then the injected water to the hot legs is circulated around the RCS loops by entrainment and there may not be sufficient water for core cooling and boron dilution in the core. Entrainment threshold calculations similar to those reported in Reference 15.6-10 demonstrates that significant hot leg entrainment will not occur after 100 minutes. Therefore, the evaluation demonstrates that both hot leg injection and DVI are sufficient to provide core-cooling flow at four hours after the LOCA.

15.6.5.4 Barrier Performance

The Barrier Performance is discussed in detail in the Chapter 6, Section 6.2 on the Containment System. In general, it discusses the evaluation of the containment vessel pressure and temperature transients that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public during and after a LOCA.

15.6.5.5 Radiological Consequences

The radiological consequences evaluation for this event is based on the alternative source term (AST) guidance documented in Reference 15.6-4. The large break LOCA is the design basis case for determining radiological consequences for LOCA transients.

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity as fuel damage occurs due to the loss of coolant.

15.6.5.5.1 Evaluation Model

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code. Direct radiation doses in the main control room (MCR) and the technical support center

(TSC) from the containment, radioactive plume and the MCR or TSC emergency filtration unit are calculated with the MicroShield code (Ref. 15.6-25). Assumed source information of source for direct radiation doses in the MCR is described in Section 6.4.2.5.

- The χ/Q values used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ) are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

The potential release paths to the environment are from:

- The containment low volume purge system until the purge valves are closed
- Containment leakage from accident initiation
- ESF system leakage from accident initiation

Figure 15A-2 depicts the leakage sources to the environment modeled in the dose computation.

Additionally, radionuclide decay of the nuclides is credited prior to release to the environment. No decay is credited for activity in environment.

15.6.5.5.1.1 LOCA Consequence Model

Source Terms

All of the reactor coolant inventory are assumed to be released to the containment at the initiation of the LOCA. The reactor coolant is assumed to have initial concentration levels at the Technical Specification limits of 300 μCi/g dose equivalent (DE) Xe-133 and 1.0 μCi/g DE I-131. Iodine spikes are not considered per Reference 15.6-4.

For the design-basis accident (DBA) LOCA, all fuel assemblies in the core are assumed to be affected. The release of activity from the damaged fuel takes place in two stages. First is the gap release, which is assumed to occur at 30 second after the initiation of the accident. The early in-vessel phase immediately follows the gap release phase and is the phase when the bulk of the activity releases associated with the accident occur.

The initial fission product inventory in the core is given in Table 15.0-14. The core inventory release fractions into the containment are prescribed by Reference 15.6-4 and are shown in Table 15.0-15. The release durations are also prescribed by Reference 15.6-4 and are shown in Table 15.0-16. Onset is the time following the initiation of the accident (i.e., time =0) and is immediately followed by the early in-vessel phase. The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase.

The chemical forms of the iodine released from the fuel to the containment are prescribed by Reference 15.6-4 to be:

- Cesium iodide (Particulates) 0.95
- Elemental iodine 0.0485
- Organic iodide 0.0015

With the exception of elemental iodine and organic iodide and noble gases, fission products are assumed to be in particulate form.

The pH of the RWSP water is assumed to be maintained at 7.0 or greater. By maintaining the pH above 7.0, the assumed iodine species split fractions given above remain valid. Several pH adjustment baskets containing sodium tetraborate decahydrate are placed in the containment to maintain the desired post-accident pH conditions in the RWSP water. (See Section 6.3.2.2.5.)

The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment as it is released. This distribution is adjusted for internal compartments that have limited containment spray. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

Airborne Radioactivity Removal

Expected radioactivity removal mechanisms that are credited in the analysis are:

- Noble gases – radioactive decay
- Elemental iodine – radioactive decay, natural deposition, charcoal filter
- Organic iodide – radioactive decay, charcoal filter
- Particulates – radioactive decay, natural deposition, CSS, high-efficiency particulate air (HEPA) filter

Radioactive decay is credited for fission products remaining within containment. If a fission product escapes to the environment, no credit is taken for radioactive decay.

Reduction in airborne radioactivity in the containment by natural deposition and by the CSS is credited. Acceptable natural deposition and CSS models for removal of iodine and aerosols are described in References 15.6-18, 15.6-19, and 15.6-20. These natural deposition and CSS models are incorporated into the analysis code RADTRAD (Ref. 15.6-21). RADTRAD is used to calculate the removal of airborne radioactivity in the US-APWR containment.

Elemental iodine is removed by natural deposition on the containment wall and other objects in containment. However, natural deposition is conservatively credited to occur on the inside surface of containment only. A conservative natural deposition removal coefficient calculation is used and is based on NUREG-0800, SRP 6.5.2 (Ref. 15.6-18).

Removal of particulate iodine by natural deposition is determined based on the Powers model (10th percentile), as shown in NUREG/CR-6189 (Ref. 15.6-19). Also, the containment spray "washout" removal coefficient for particulate iodine is calculated using NUREG-0800, SRP 6.5.2 (Ref. 15.6-18).

The evaluation of the containment sprays should address areas within the containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour.

Decontamination Factor (DF) for the containment atmosphere achieved by the containment spray system is time dependent and is determined based on NUREG-0800, SRP 6.5.2 (Ref. 15.6-18). Credit for elemental iodine removal is assumed to continue until the DF of 200 (See Appendix 15A.1.2) is reached in the containment atmosphere.

Radioactivity removal by containment spray and natural deposition is discussed additionally in Section 6.5.2 and Appendix 15A.1.2.

In addition to removal of airborne radioactivity by natural deposition and by sprays, removal of airborne activity by filters is considered. Decay of fission products and in-growth of daughter products are also considered. The transport pathway models include filters, and air leakage. Doses at the EAB, LPZ, and the MCR are also calculated.

Release paths

Radioactive material can escape from the containment to the environment by three different pathways for the large break LOCA. Releases occur from the containment purge line prior to containment isolation, containment leakage, and ECCS equipment leakage outside the containment. Containment leakage releases consist of unfiltered leakage and leakage filtered by the annulus emergency exhaust system. The doses from these release paths are summed to obtain the total dose for the LOCA. The releases are assumed to be ground level releases.

It is assumed that containment purge is in operation when the LOCA occurs. Prior to containment isolation, radionuclides released into the containment from the break can escape through this pathway until the purge system isolation valves are closed. The volume of gas escaping is calculated based on a release rate of 20,700 cfm and a valve closure time of 15 seconds from accident initiation. Radionuclides release from the containment low volume purge system assumes that 100% of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the LOCA. No credit is taken for the filters in this purge line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours. Per Reference 15.6-4, the leak rate may be reduced to 50% of the peak leak rate after the first 24 hours. The containment integrated leak test verifies that the leak rate is less than the allowable leakage rate specified in 10 CFR 50, Appendix J.

The annulus emergency exhaust system prevents uncontrolled radioactive release from the containment penetrations and safeguard components to the environment. This system has two annulus emergency exhaust filtration units, which maintain the penetration areas and safeguard component areas at a negative pressure, during accident conditions. The annulus emergency exhaust system automatically initiates on a ECCS actuation signal. This system has HEPA filters and particulates are removed by the filters.

ESF systems that recirculate RWSP water outside of the containment are assumed to leak during their intended operation. These ESF systems include the containment spray system, residual heat removal system, and the safety injection pump. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The only borated water source for the US-APWR ESF recirculation systems is the RWSP, which is located within the containment. The following assumptions are used for evaluating the consequences of leakage from ESF components outside the containment.

With the exception of noble gases which are assumed to escape to the containment atmosphere, all the fission products released from the fuel to the containment (as defined in Table 15.0-15) are assumed to instantaneously and homogeneously mix in the RWSP water at the time of release from the core.

- The leakage is taken as two times (Ref.15.6-4) the sum of the simultaneous leakage from all components in the ESF recirculation systems above. The leakage is assumed to start at the earliest time the recirculation flow occurred in these systems and ended at the latest time the releases from these systems are terminated.
- With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.
- The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in the release activity by the ESF filter system is credited as this system serves those building areas where ESF equipment leakage can occur. The ESF filter system is evaluated based on the guidance of RG 1.52 (Ref. 15.6-22).

The analysis duration is 30 days for containment and ECCS leakage per Reference 15.6-4.

<u>Dose calculation</u>

The EAB dose is calculated for the 2-hour period over which the highest doses would be incurred by an individual located at the EAB. Because of the delays associated with the core damage for this accident, the first 2-hour of the accident are not the worst 2 hour interval for accumulating a dose.

The LPZ boundary dose is calculated for the 30-day duration of the accident.

For both the EAB and LPZ dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

The dose calculation models are provided in Section 15A.3 for the determination of doses resulting from activity which releases the environment.

15.6.5.5.1.2 Main Control Room Consequence Model

The release from the LOCA has the potential to expose personnel in the MCR. The TEDE analysis considered all sources of radiation that will cause exposure to MCR personnel. The sources include:

- Contamination of the MCR atmosphere by the intake of the radioactive material contained in the radioactive plume released from the facility,
- Contamination of the MCR atmosphere by the infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope (CRE),
- Direct radiation from the external radioactive plume released from the facility,
- Direct radiation from radioactive material in the containment,
- Direct radiation from radioactive material in the MCR emergency filtration unit.

The radioactive material releases and radiation levels used in the MCR dose analysis are based on the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.

Credit for engineered safety features that mitigate airborne radioactive material within the MCR are assumed according to the guidance given in the Reference 15.6-24. Such features included MCR isolation or pressurization, or intake or recirculation filtration.

When radioactivity enters the MCR, the MCR heating, ventilation, and air conditioning (HVAC) system switches over to the pressurization mode. The MCR HVAC system which works during normal operation is not a safety-class system but provides defense in depth.

The MCR HVAC system provides passive pressurization of the MCR from a filtered air intake to prevent in-leakage of contaminated air to the MCR during the accident. The MCR HVAC system automatically transfers to emergency operation mode (pressurization mode) on a ECCS actuation or a high radiation signal. The total amount of unfiltered inleakage into the control room is 120 cfm. This total value includes the inleakage through the control room envelope and unexpected inleakage through the control room envelope such as through ingress to and egress from doors without a vestibule.

Assumed filter efficiency of MCR emergency filtration units is based on RG 1.52 (Ref. 15.6-22) and Generic Letter 99-02 (Ref.15.6-23).

Reference 15.6-4 provides guidance on calculating the consequences to the MCR receptor. The dose receptor is a hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual is assumed to be 3.5×10^{-4} m^3/s.

MCR doses are calculated using dose conversion factors identified in Regulatory Position 4.1 of Reference 15.6-4. The deep dose equivalent (DDE) from photons are corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression is used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room.

$$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$

The MCR dose calculation models are provided in Section 15A.4 for the determination of doses resulting from activity which enters the CRE.

15.6.5.5.1.3 Technical Support Center Consequence Model

The radioactive material releases and radiation levels used in the technical support center (TSC) dose analysis used the same source term, transport, and release assumptions used for determining the MCR TEDE values. The TSC dose calculation models are the same as the MCR dose calculation model (See Table 15.6.5-5). That is, ratio of ventilation flow rate to TSC volume is the same value as that of the MCR. Also, the efficiency of HEPA filter and charcoal absorber of the TSC are the same as those of the MCR. The distances from release points to receptors are almost the same between the TSC and the MCR (See Table 15A-23). Therefore, the radiological consequences in the TSC are represented by those in the MCR.

15.6.5.5.2 Input Parameters and Initial Conditions

Major input parameters for the consequence analysis during the LOCA are summarized in Table 15.6.5-4, Table 15.0-10, and Tables 15.0-12 through 15.0-16. Also, the major input parameters for the MCR and TSC consequence analysis during the LOCA are summarized in Table 15.6.5-5, and Tables 15A-18 through 15A-24.

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix A of RG 1.183 (Ref. 15.6-4).

15.6.5.5.3 Results

The doses calculated for the EAB and the LPZ boundary are listed in Table 15.6.5-16. The TEDE doses for the limiting 2 hours are calculated to be 13 rem at the EAB and 13 rem at the LPZ outer boundary. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The doses calculated for the MCR personnel due to airborne activity entering the MCR are listed in Table 15.6.5-16. Also listed on Table 15.6.5-16 are the doses due to direct shine from the activity in the containment, from the radioactive plume and from the MCR emergency filtration unit. The total of the four dose pathways is within the dose criteria of 5 rem TEDE as defined in GDC 19. The dose for TSC is bounded by the MCR doses.

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 15.6.5-1

US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis

	Reference Case	ASTRUM run conditions
Plant physical configuration		
Fraction of SG tube plugged	10% (maximum)	10% (maximum)
Hot assembly location	Under the open hole	Under the open hole
Power-related Parameters		
Core power	4451MWt (100%)	98% $\leq P_{core} \leq$ 102% of 4451 MWt
Peaking factor (F_Q)	2.6	$F_Q \leq 2.6$
Axial power distribution	Top skewed	Figure 15.6.5-8
Peripheral assembly power	0.2 (lower bound)	0.2 (lower bound)
Hot assembly burnup	Beginning of life (BOL)	Beginning of life (BOL)
Fuel assembly type	17 X 17 ZIRLO™ cladding	17 X 17 ZIRLO™ cladding
Initial RCS Fluid Condition		
RCS average temperature	583.8°F	583.8-4.0°F $\leq T_{AVG} \leq$ 583.8+4.0°F
Pressurizer pressure	2250 psia	2250-30 psia $\leq P_{RCS} \leq$ 2250+30 psia
Primary coolant flow	112,000 gpm/loop (thermal design flow)	112,000 gpm/loop (thermal design flow)
Accumulator temperature	95°F	70°F $\leq T_{ACC} \leq$ 120°F
Accumulator pressure	655 psia	600 psia $\leq P_{ACC} \leq$ 710 psia
Accumulator water volume	2152 ft^3	2126 $ft^3 \leq V_{ACC} \leq$ 2179 ft^3
Accident Boundary Condition		
Break location	Cold leg (in the loop with pressurizer)	Cold leg (in the loop with pressurizer)
Break type	Double-ended guillotine break	Split break and double-ended guillotine breaks
Discharge coefficient	1.0	0.8 – 1.4
Offsite Power	Not available	Not available
Number of SI pumps available	2	2
Safety Injection flow rate	Minimum	Minimum
Safety Injection temperature	76°F	45°F $\leq T_{SI} \leq$ 120°F
Safety Injection delay	118 sec	118 sec
Containment pressure	Bounded (minimum)	Bounded (minimum)

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 15.6.5-3

US-APWR Major Plant Parameter Inputs Used in the Post-LOCA Long Term Cooling Analysis

Parameters	Values
System pressure	Atmospheric pressure (for large break LOCA)
	120 psia (*) (for small break LOCA)
Core Power	102% of rated power (4540 MWt)
Decay Heat	1971 ANS, infinite operation plus 20%
Boric Acid Source	
RWSP	
Boric Acid Concentration	Maximum (2.4 wt.%)
Volume	Maximum (89,000 ft^3)
Water Density	Maximum (Density at 39°F)
Accumulator	
Boric Acid Concentration	Maximum (2.4 wt.%)
Volume	Maximum
Density	Maximum (Density at 39°F)
RCS	
Boric Acid Concentration	Maximum (1.3 wt.%)
Volume	Minimum
Density	Minimum (Density at T_{hot} + 4°F)
ECC Water Temperature	Saturation temperature at atmospheric pressure (Large Break LOCA)
	RWSP maximum temperature reached during a LOCA (Small Break LOCA)
Operator Actions	Credited (**)

Notes:

(*) Corresponding to the boric acid congruent melting temperature of 339.8°F

(**) To perform the switchover from RV injection mode to the simultaneous RV and hot leg injection mode.

Table 15.6.5-4

US-APWR Major Input Parameters Used in the LOCA Consequence Analysis (Sheet 1 of 2)

Parameters	Value
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
reactor coolant radionuclide inventory	
Noble gas concentration	300 μCi/g DE Xe-133
Iodine concentration	1.0 μCi/g DE I-131
Particulate concentration	Based on 1% fuel defect (See Table 11.1-2.)
reactor coolant mass (lb)	646,000
Radionuclide release from damaged core	
Core activity at start of accident	See Table 15.0-14.
Release fractions to containment	See Table 15.0-15.
Release timing and durations	See Table 15.0-16.
Iodine species distribution	
• Cesium iodide (%)	95
• Elemental (%)	4.85
• Organic (%)	0.15
Containment purge release data	
Containment purge flow rate (cfm)	20,700
Duration of purge from accident initiation (s)	15
Release characteristics	100% of reactor coolant inventory is released to the containment at the initiation of the LOCA

Table 15.6.5-4

US-APWR Major Input Parameters Used in the LOCA Consequence Analysis (Sheet 2 of 2)

Parameters	Value
Containment leakage release data	
Containment volume (ft^3)	2,800,000
Containment leak rate (%/d), 0-24 hr	0.15
Containment leak rate (%/d), > 24 hr	0.075
Leakage fraction to penetration areas (%)	50
Leakage fraction to environment (%)	50
Filter efficiency for particulates in annulus emergency exhaust system (%)	99
Penetration areas negative pressure arrival time (min)	4
Containment spray system initiation time (min)	5
Containment spray flow rate (lb/h)	2,650,000
Sprayed Containment Volume (ft^3)	1,680,000
Mixing rate between the sprayed and unsprayed regions of containment (cfm)	37,300
Elemental iodine deposition removal coefficient in sprayed and unsprayed regions (h^{-1})	0.376
Powers model percentile for particulates deposition removal coefficient in unsprayed region only (%)	See Section 15A.1.2. 10
Particulates containment spray removal coefficient in sprayed region only (h^{-1})	7.32 (When DF for particulate reaches 50, this removal coefficient is decreased by a factor of 10.)
DF limit for elemental iodine removal	200
Elemental iodine removal end time	15.0 hr
The time when the DF for particulate equals 50	3.23 hr
ESF system leakage release data	
Recirculation water mass (lb)	3,540,000
Recirculation water leakage rate (lb/h)	17.6
Start time of recirculation water leakage (min)	0
Flash fraction (%)	10
Accident period (d)	30
χ/Q	See Tables 15.0-13 and 15A-23.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.6.5-5

US-APWR Major Input Parameters Used in the MCR and TSC Consequence Analysis for the LOCA

Parameters	Value	
	MCR	TSC
Envelope volume (including MCR) (ft3)	140,000	46,000
Occupancy frequency 0 to 24 hrs 24 hrs to 96 hrs 96 hrs to 720 hrs	 1.0 0.6 0.4	 1.0 0.6 0.4
Total amount of unfiltered inleakage (cfm)	120	40
HVAC system Time delay to switch from normal operation to emergency CRE air filtration mode (s) Unfiltered air intake flow during normal operation (cfm) Filtered air intake flow (cfm) Filtered air recirculation flow (cfm) Filter efficiency ● Elemental iodine (%) ● Organic iodine (%) ● Particulates (%)	 180 1800 1,200 2,400 95 95 99	 180 1000 400 1400 95 95 99

Table 15.6.5-6

Sequence of Events for Reference Case Large Break LOCA

Events	Time (sec)
Break occurs, coincident with LOOP	0.0
ECCS actuation signal	6
Accumulator high flow rate injection begins	13
End of blowdown	34
End of refill	37
Accumulator low flow rate injection begins	56
PCT occurs	64
High Head Injection System begins	124
PCT Elevation quenched	190
End of transient, core covered	220

Table 15.6.5-7

Sequence of Events for Limiting Case Large Break LOCA
(95th Percentile PCT with 95% Confidence)

Events	Time (sec)
Break occurs coincident with LOOP	0.0
ECCS actuation signal	6
Accumulator high flow rate injection begins	13
End of blowdown	32
End of refill	38
Accumulator low flow rate injection begins	56
PCT occurs	75
High Head Injection System begins	124
PCT Elevation quenched	170
End of transient, core covered	220

Table 15.6.5-8

Best Estimate Large Break LOCA Core Performance Results (95th Percentile with 95% Confidence)

Parameters	Values	Criteria
Peak Cladding Temperature (°F)	1758 (Run 48)	< 2200
Local maximum cladding oxidation (%)	2.8 (Run 103)	< 17.0
Core wide maximum cladding oxidation (%)	0.2 (Run 101)	< 1.0

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~~78
Emergency Power Source initiates	115
Core upper region uncovery	~~122~~124
High Head Injection System begins	130
Peak Cladding Temperature occurs	~~136~~137
Core upper region recovery	~~142~~141
Emergency feedwater flow begins	145
Accumulator injection begins	~~299~~317

Table 15.6.5-10

Core Performance Results for 7.5-inch Small Break LOCA

	Values
Peak Cladding Temperature (°F)	~~773~~761
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
Emergency Power Source initiates	111
High Head Injection System begins	126
Emergency feedwater flow begins	141
Peak Cladding Temperature occurs	~~169~~161
Core upper region recovery	~~339~~356

Table 15.6.5-12

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

Items	Values
Peak Cladding Temperature (°F)	~~1323~~1302
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~~1256
Peak Cladding Temperature	~~lower than the initial value~~1505
Core upper region recovery	~~N/A~~1856

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~~789
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~~1302°F
13-inch at cold leg (bottom)	~~1154~~1250°F
12-inch at cold leg (bottom)	~~938~~1220°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~~701°F
7.5-inch at cold leg (bottom)	761°F
7-inch at cold leg (bottom)	~~756~~715°F
6.5-inch at cold leg (bottom)	~~lower than the initial temperature~~691°F
6-inch at cold leg (bottom)	~~lower than the initial temperature~~708°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

Table 15.6.5-16

Radiological Consequences of the LOCA

Dose Location	TEDE Dose (rem)
EAB (0.5 to 2.5 hours)	13
LPZ outer boundary	13
MCR dose	
Airborne activity entering the MCR	4.5
Direct radiation from the containment	8.2×10^{-3}
Direct radiation from the radioactive plume	2.1×10^{-4}
Direct radiation from the recirculation filters	9.2×10^{-3}
Total	4.5
TSC dose	Less than MCR dose

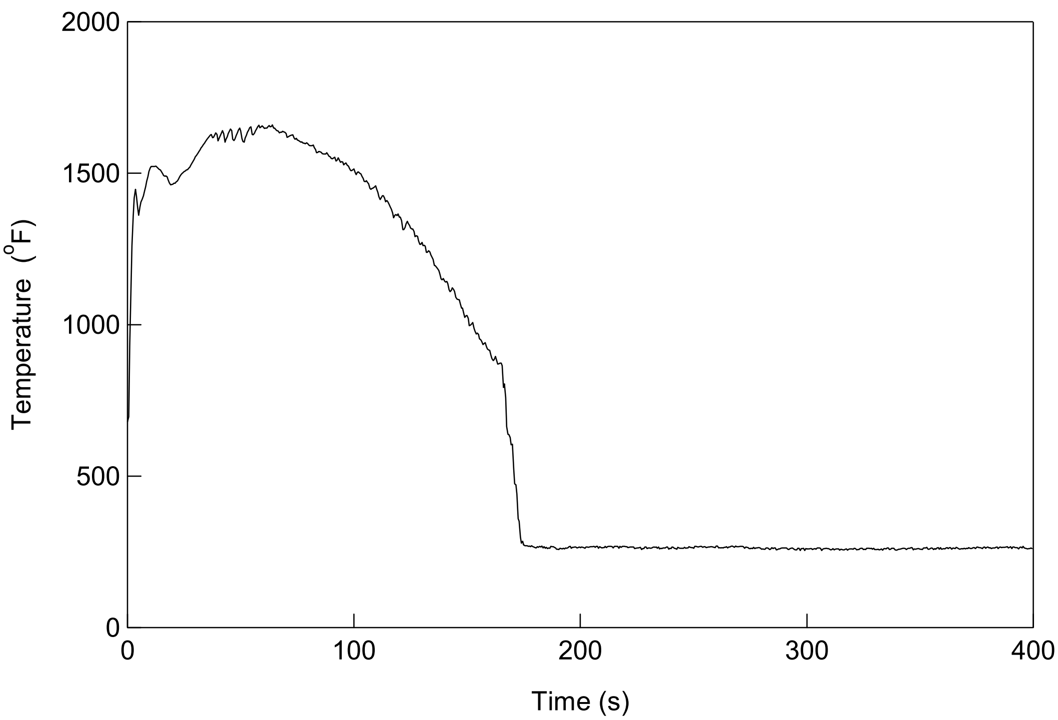


Figure 15.6.5-1 Hot Rod Cladding Temperature at the Limiting Elevation (10 ft) for Large Break LOCA (Reference Case)

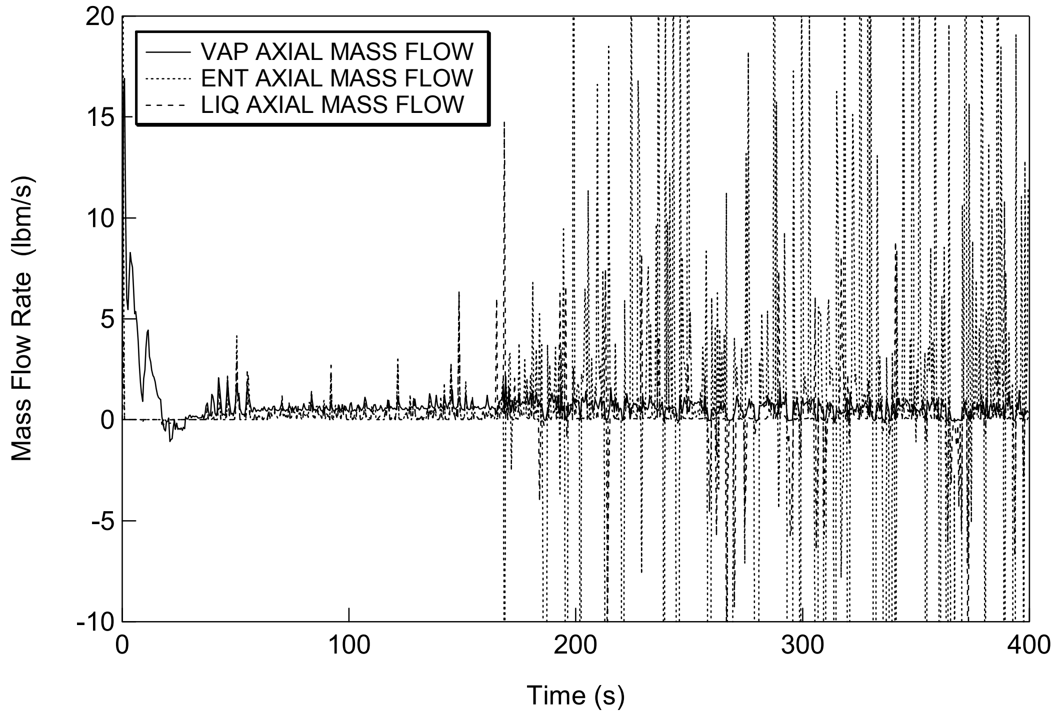


Figure 15.6.5-2 **Hot Assembly Exit Vapor, Entrainment, Liquid Flow Rates for Large Break LOCA (Reference Case)**

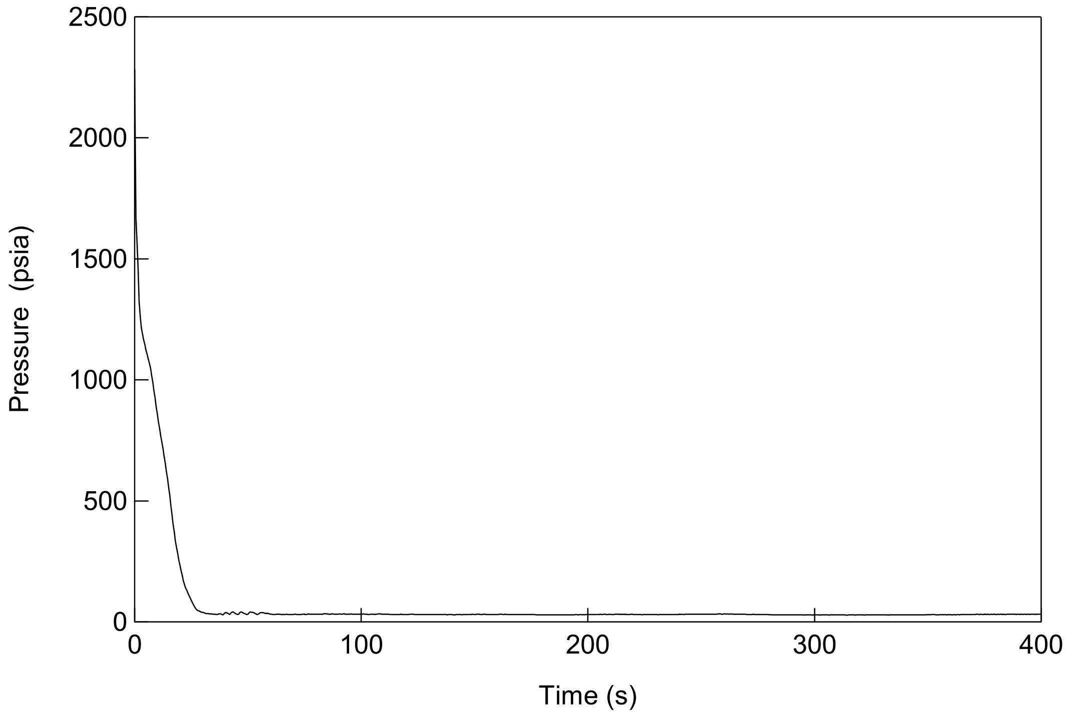


Figure 15.6.5-3 Core Pressure Transient for Large Break LOCA (Reference Case)

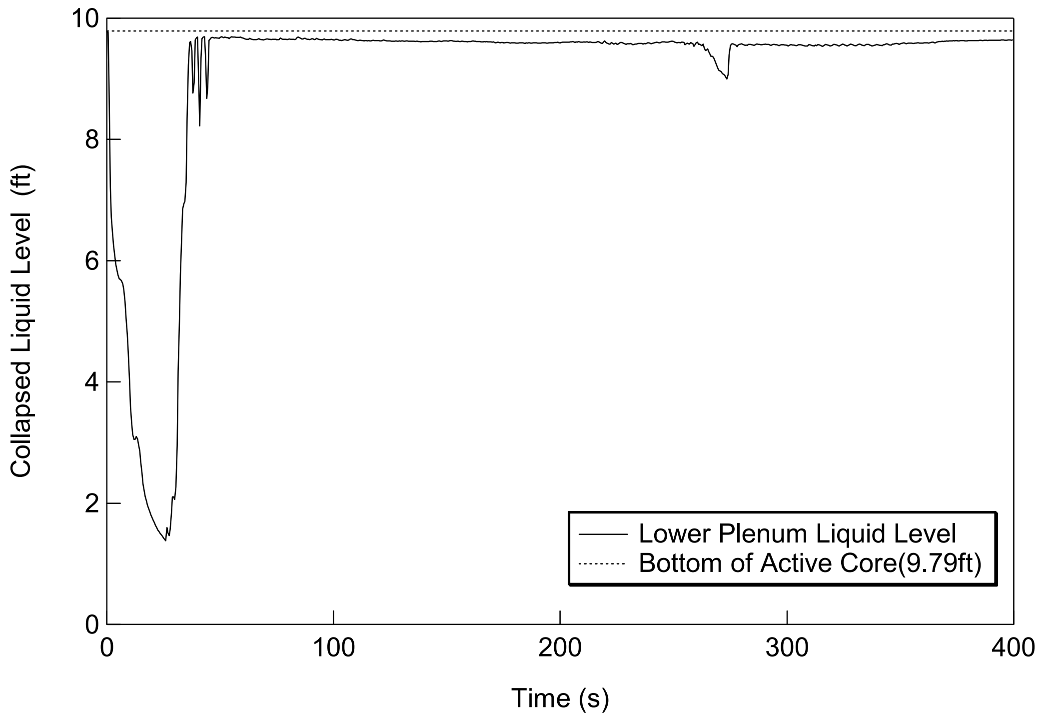


Figure 15.6.5-4 Lower Plenum Liquid Level for Large Break LOCA (Reference Case)

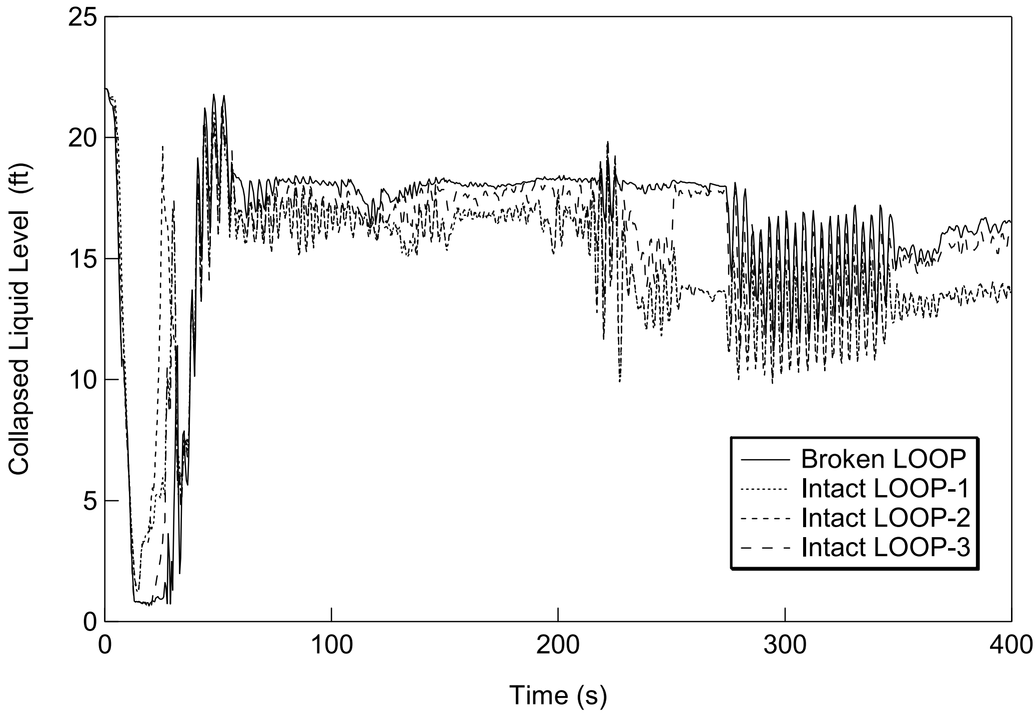


Figure 15.6.5-5 Downcomer Liquid Level for Large Break LOCA (Reference Case)

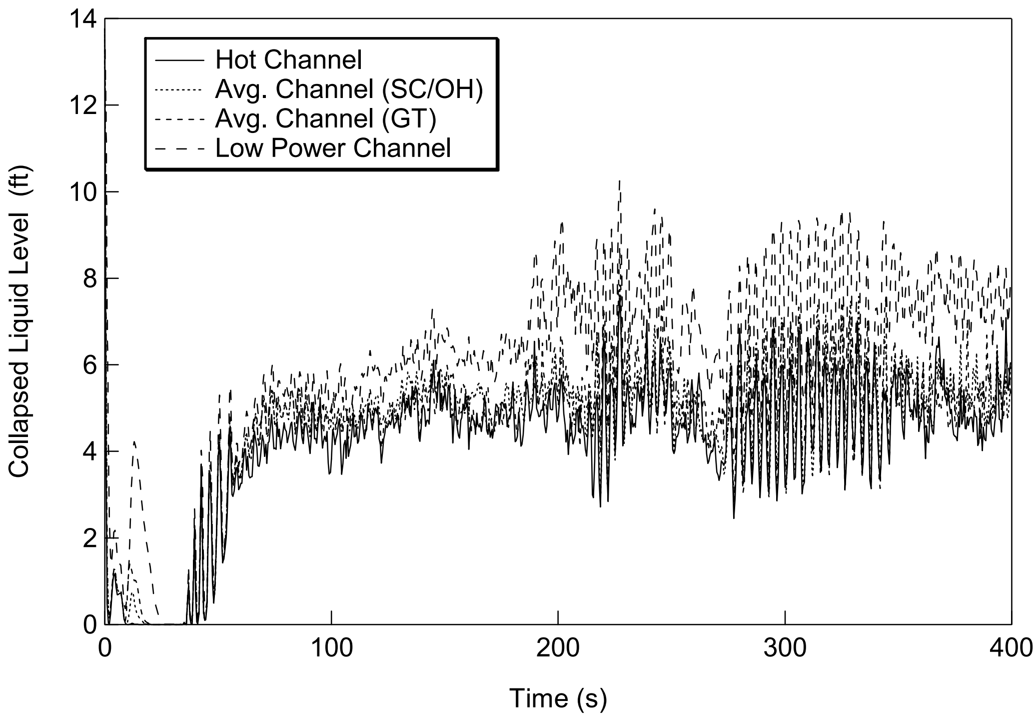


Figure 15.6.5-6 Core Collapsed Liquid Level for Large Break LOCA (Reference Case)

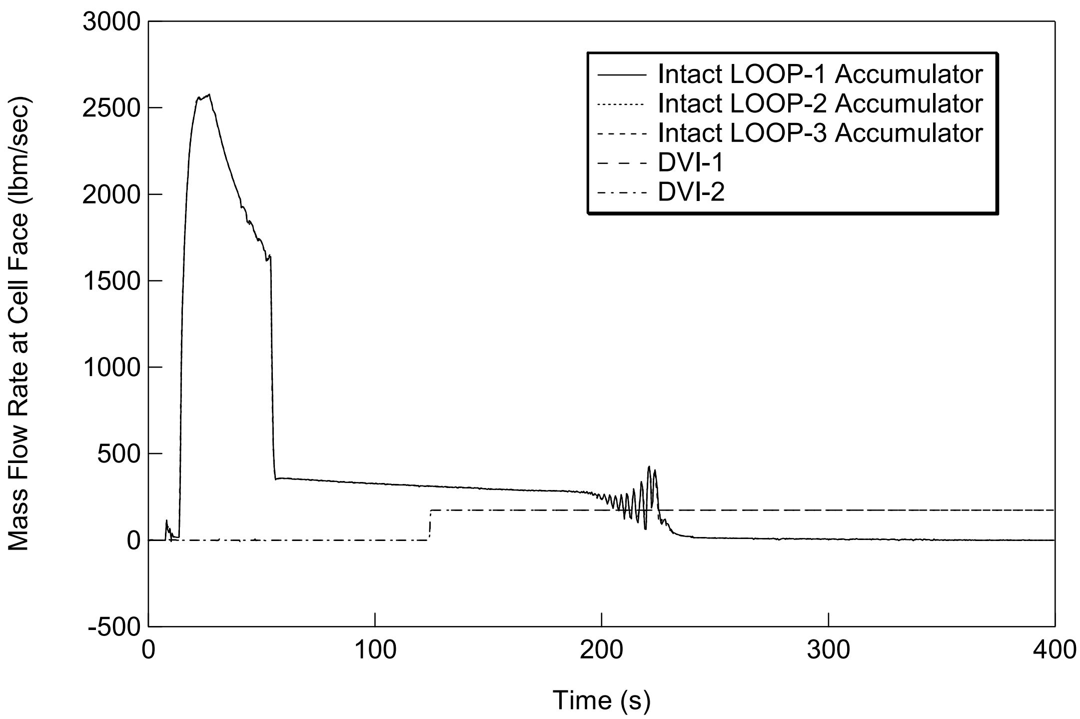
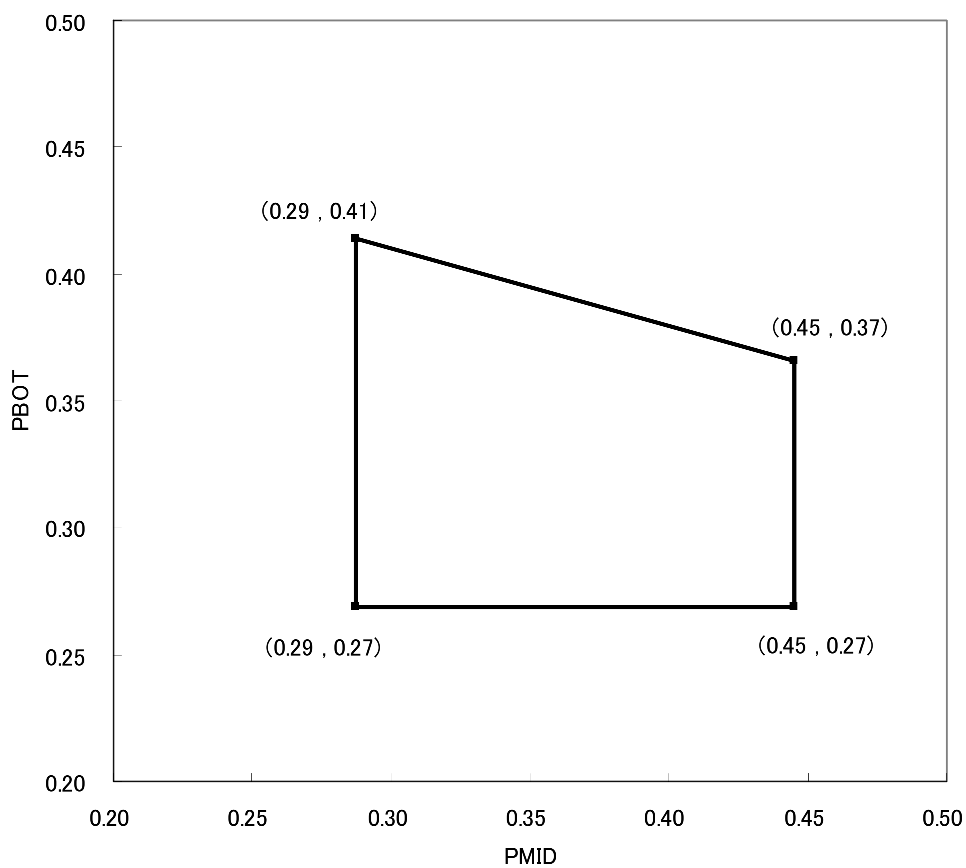


Figure 15.6.5-7 Accumulators and SI System Flowrates to DVI-1 and -2 for Large Break LOCA (Reference Case)



PBOT: integrated power fraction in the lower 3rd of the core

PMID: integrated power fraction in the middle 3rd of the core.

Figure 15.6.5-8 Axial Power Shape Operating Space Envelope for Large Break LOCA

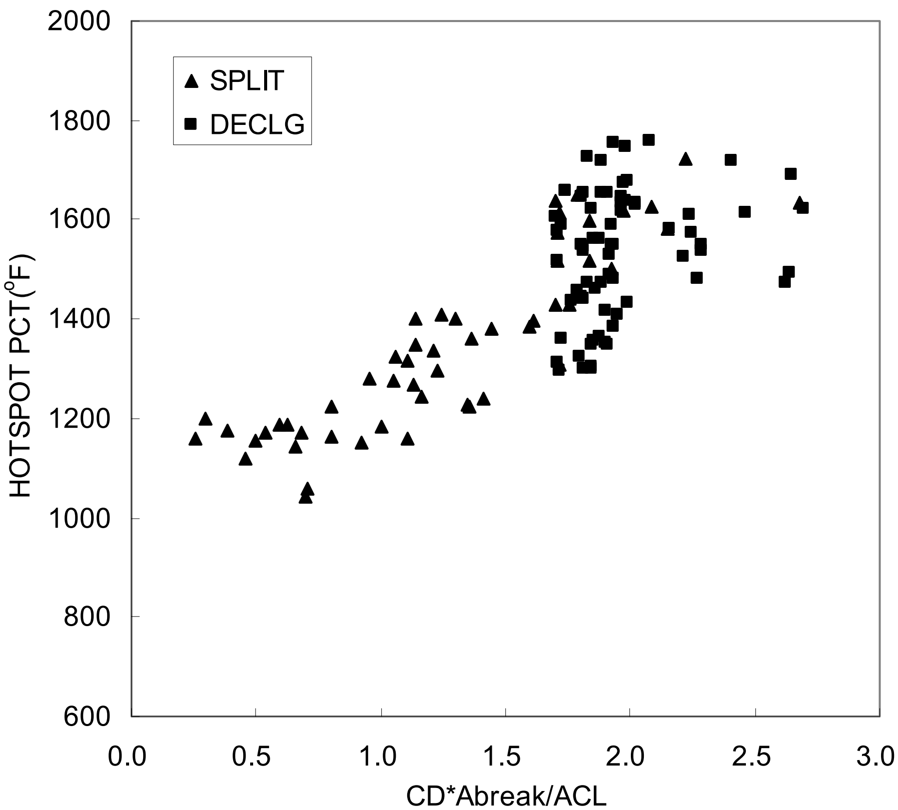


Figure 15.6.5-9 **HOTSPOT PCT versus Effective Break Area Scatter Plot for Large Break LOCA**

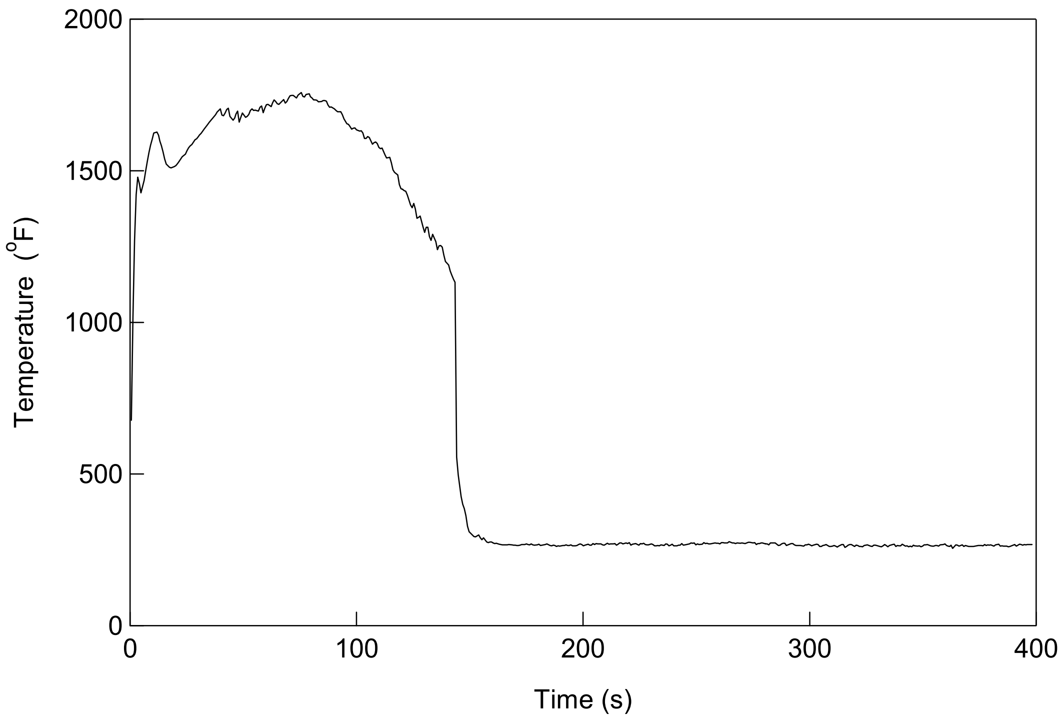


Figure 15.6.5-10 HOTSPOT Cladding Temperature Transient at the Limiting Elevation for the PCT Limiting Case for Large Break LOCA

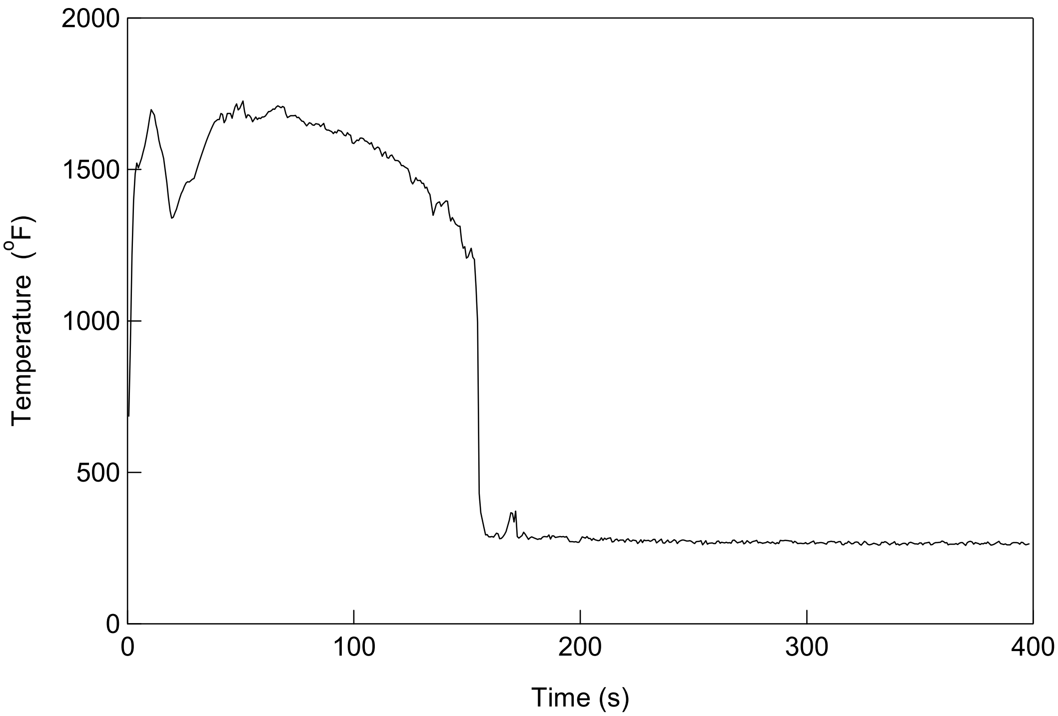


Figure 15.6.5-11 HOTSPOT Cladding Temperature Transient at the Limiting Elevation for the LMO Limiting Case for Large Break LOCA

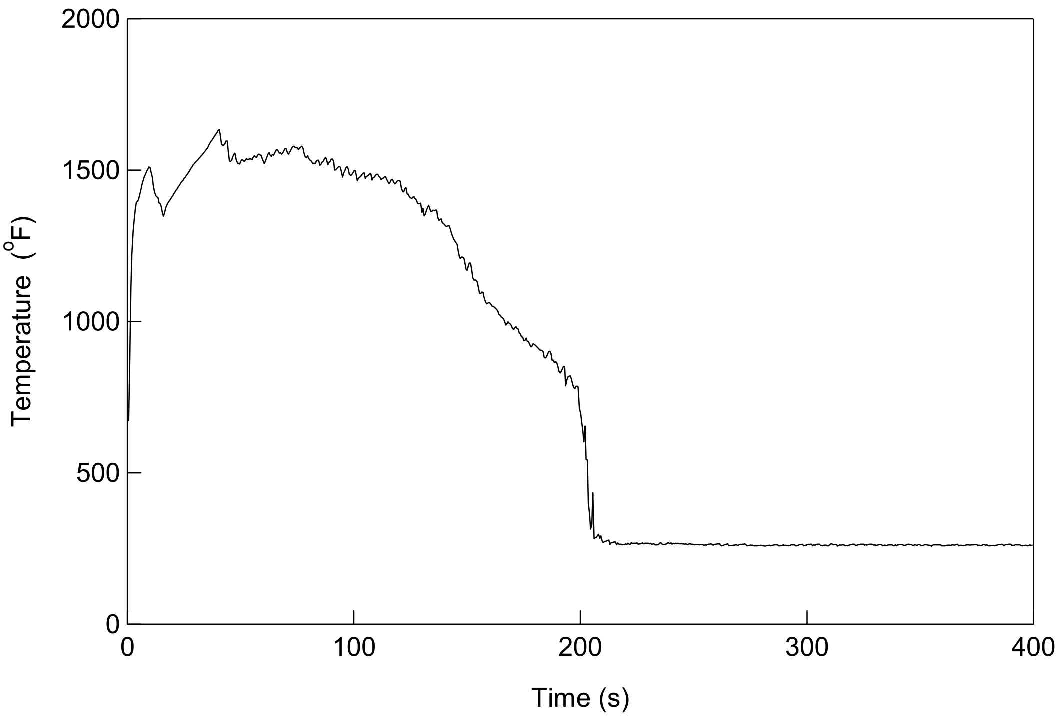


Figure 15.6.5-12 PCT Transient for the CWO Limiting Case for Large Break LOCA

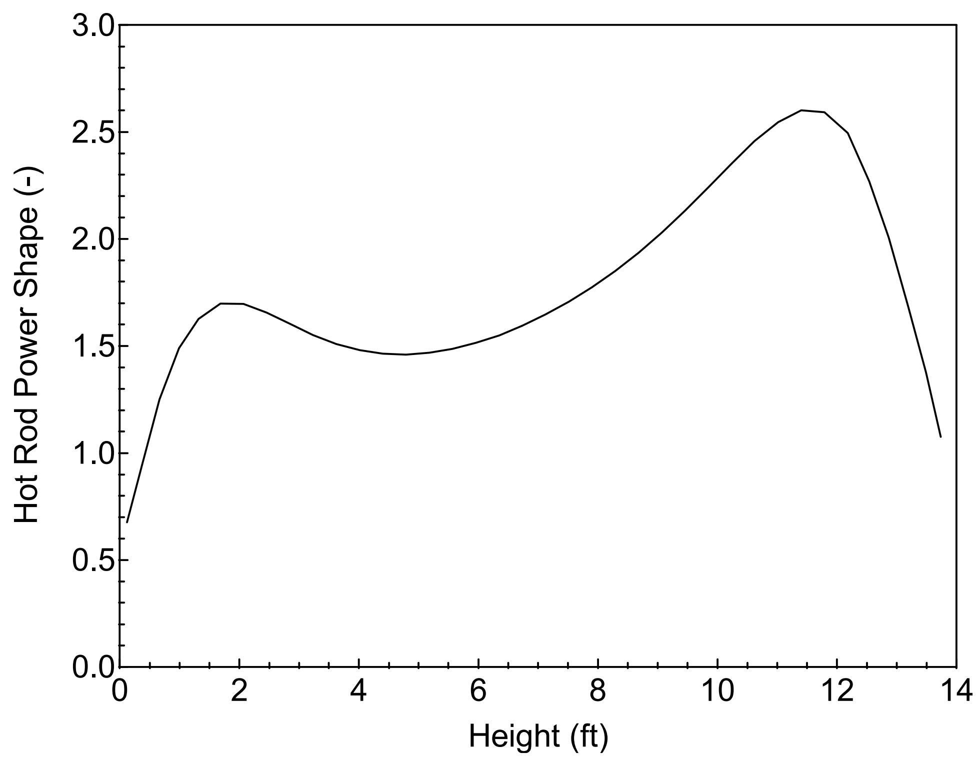
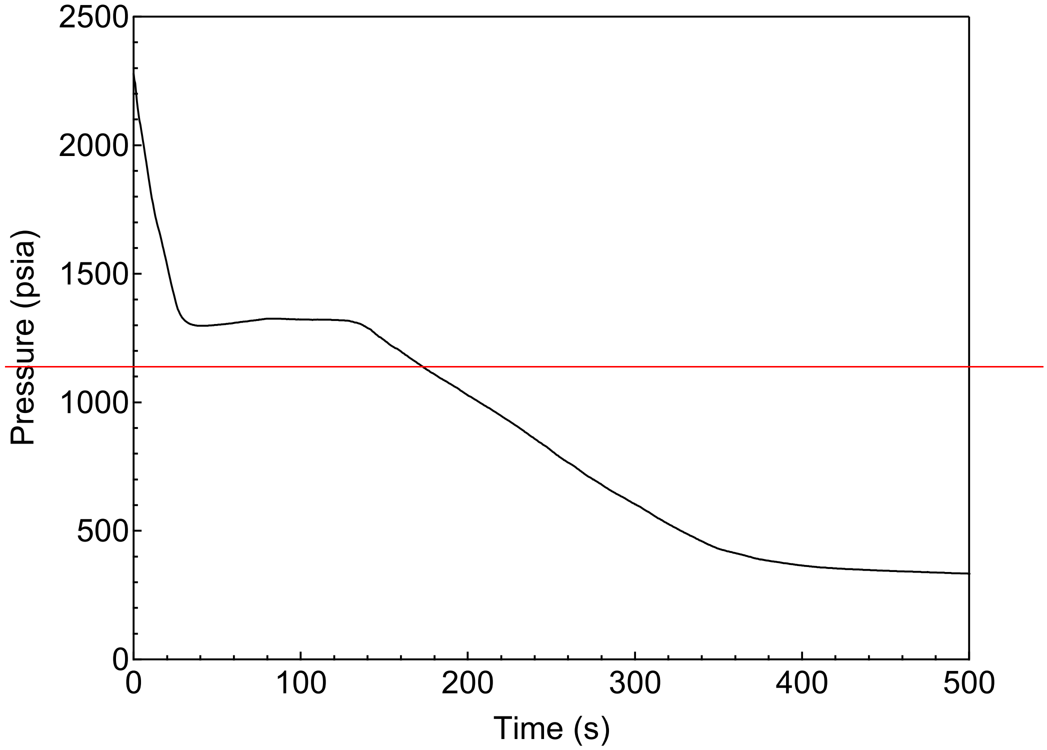


Figure 15.6.5-13 Hot Rod Power Shape Used for Small Break LOCA analysis



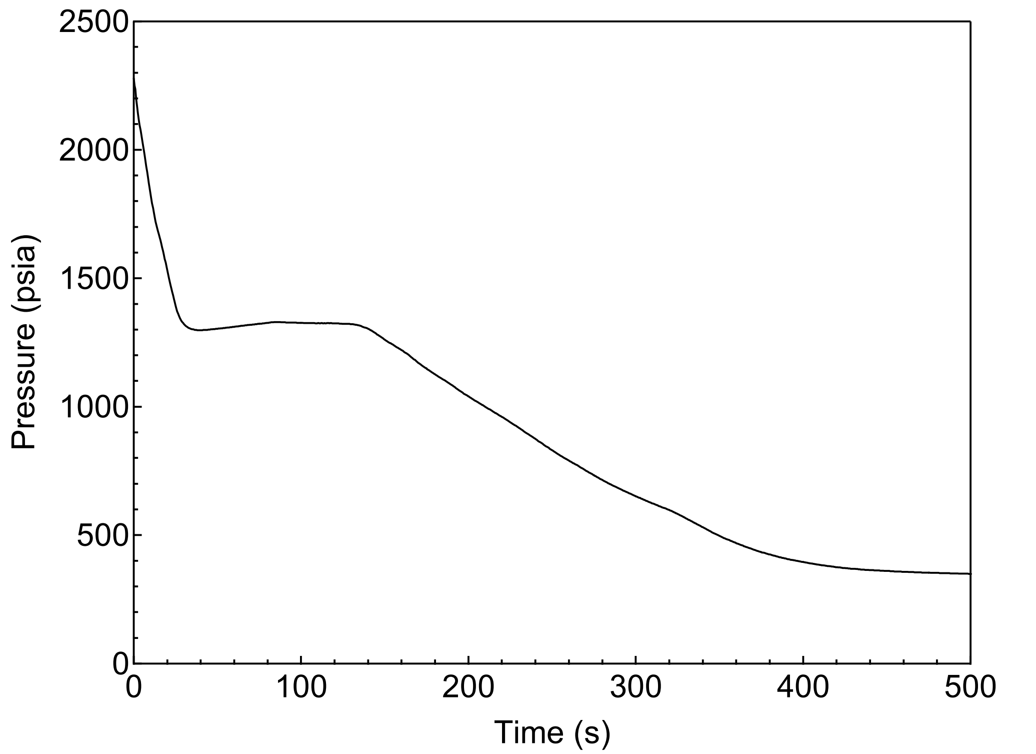


Figure 15.6.5-14 RCS (Pressurizer) Pressure Transient for 7.5-inch Small Break LOCA

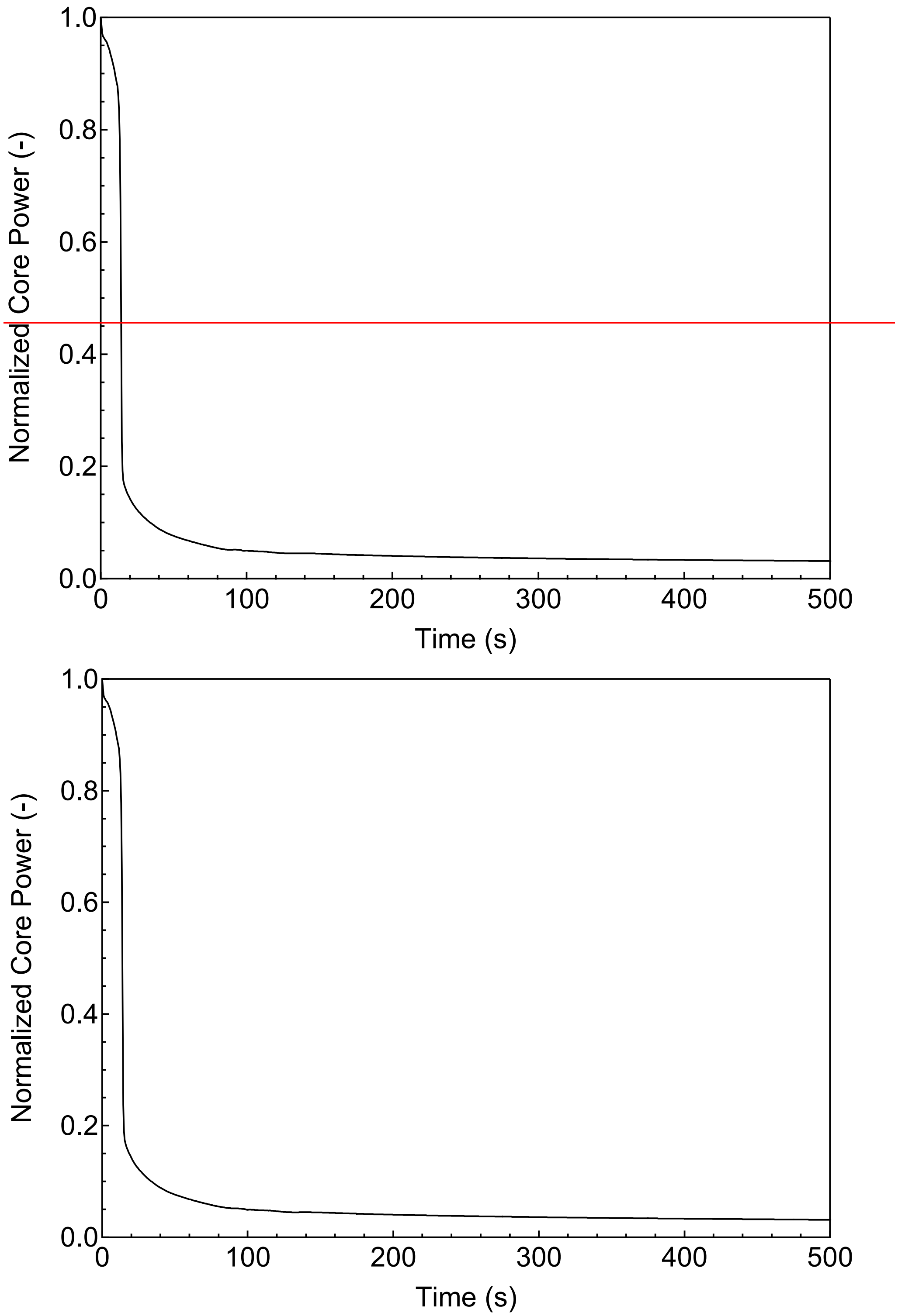


Figure 15.6.5-15 Normalized Core Power for 7.5-inch Small Break LOCA

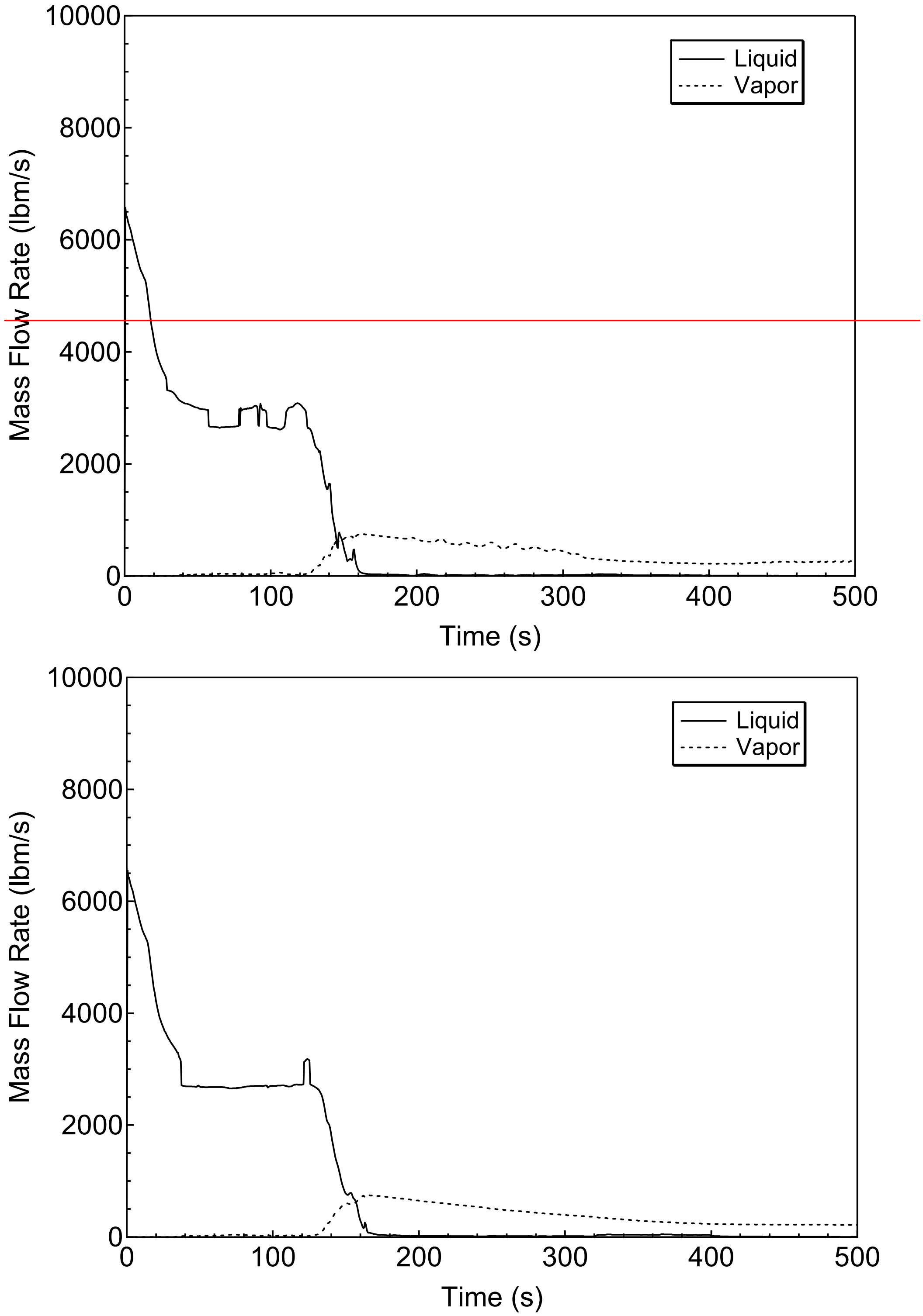


Figure 15.6.5-16 Liquid and Vapor Discharges through the Break for 7.5-inch Small Break LOCA

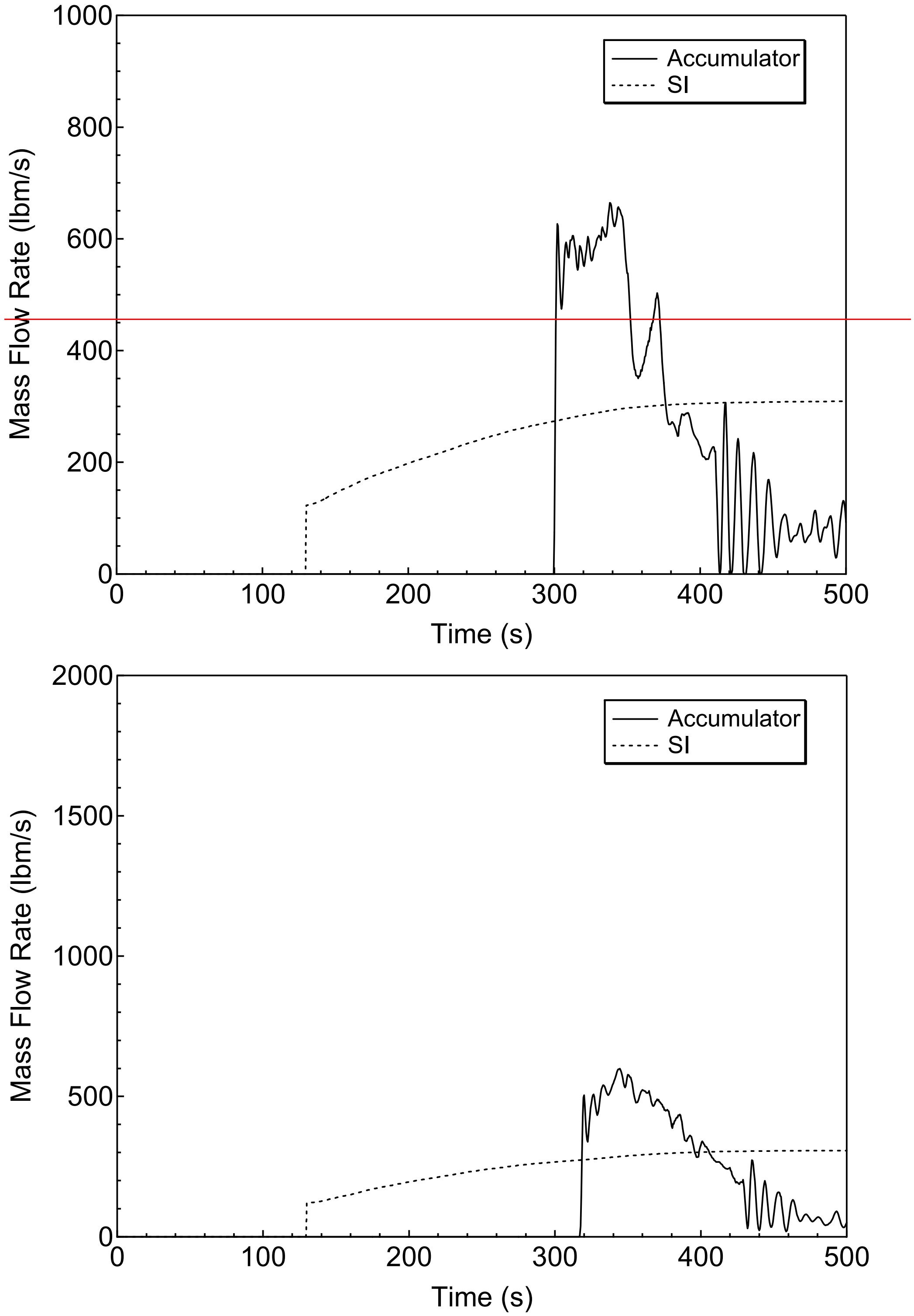


Figure 15.6.5-17 Accumulator and Safety Injection Mass Flowrates for 7.5-inch Small Break LOCA

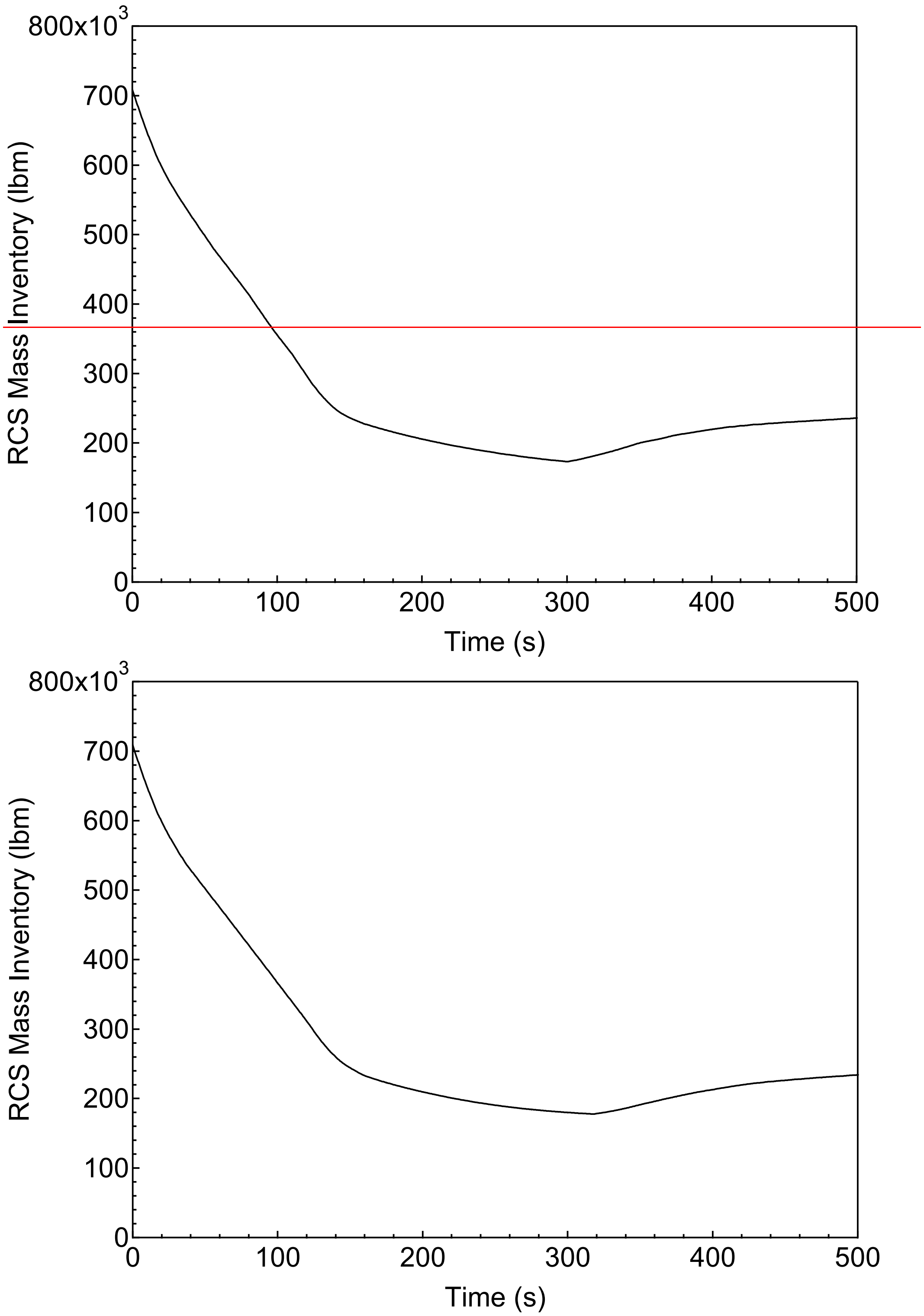


Figure 15.6.5-18 RCS Mass Inventory for 7.5-inch Small Break LOCA

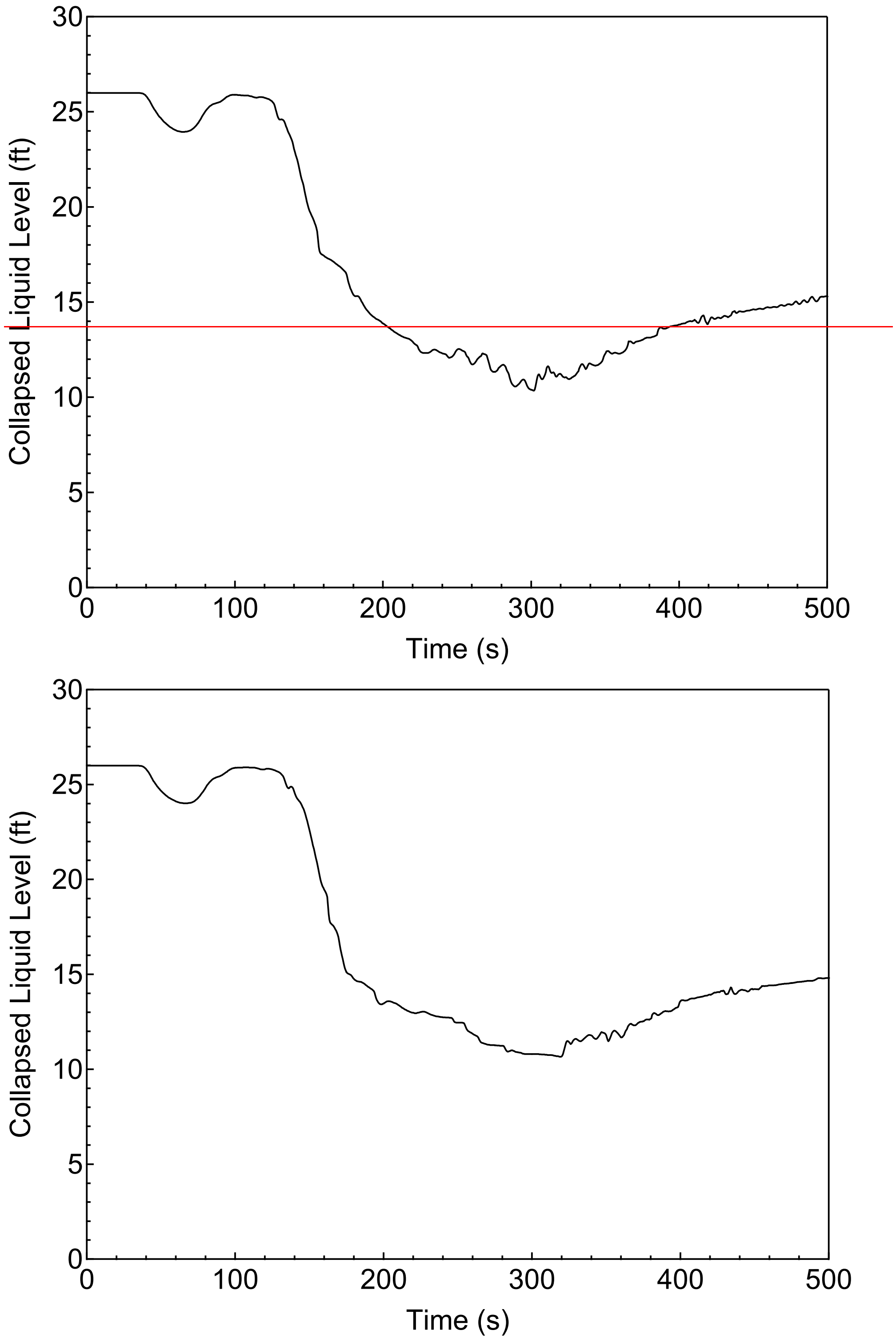


Figure 15.6.5-19 Downcomer Collapsed Level for 7.5-inch Small Break LOCA

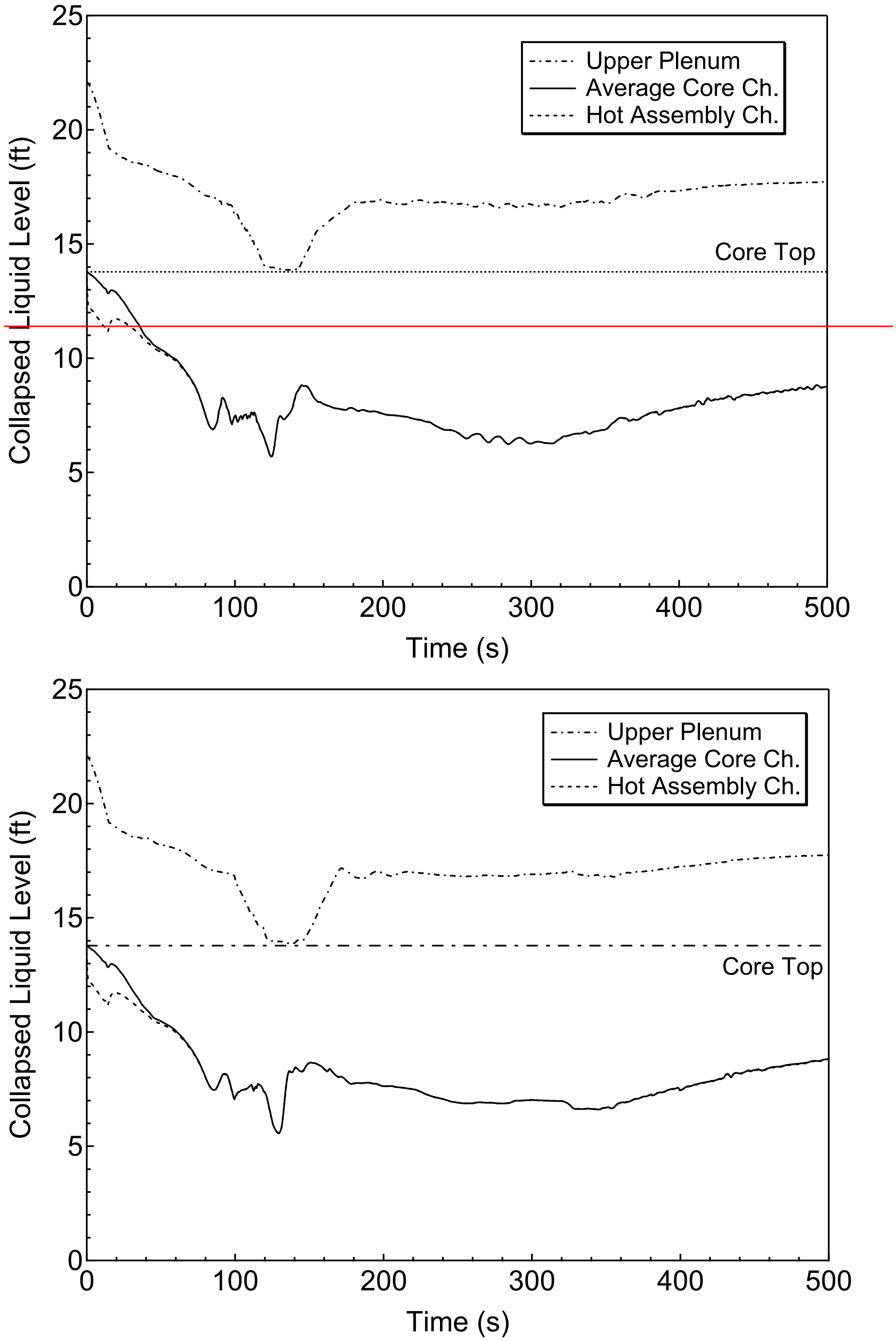
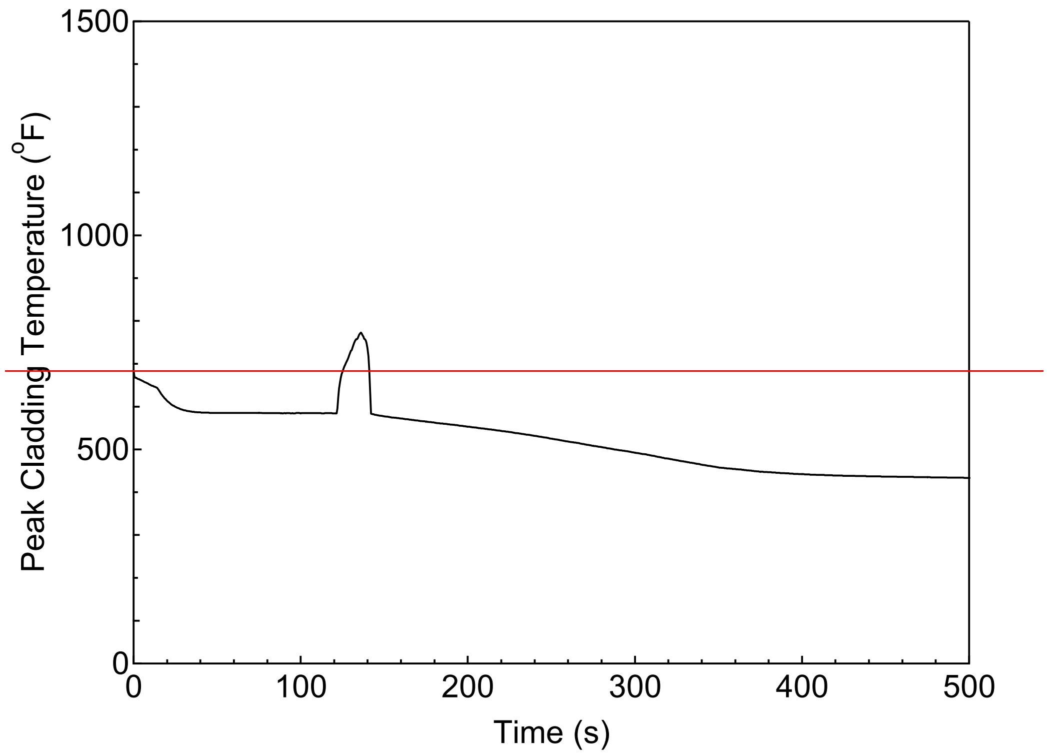


Figure 15.6.5-20 Core/Upper Plenum Collapsed Level for 7.5-inch Small Break LOCA



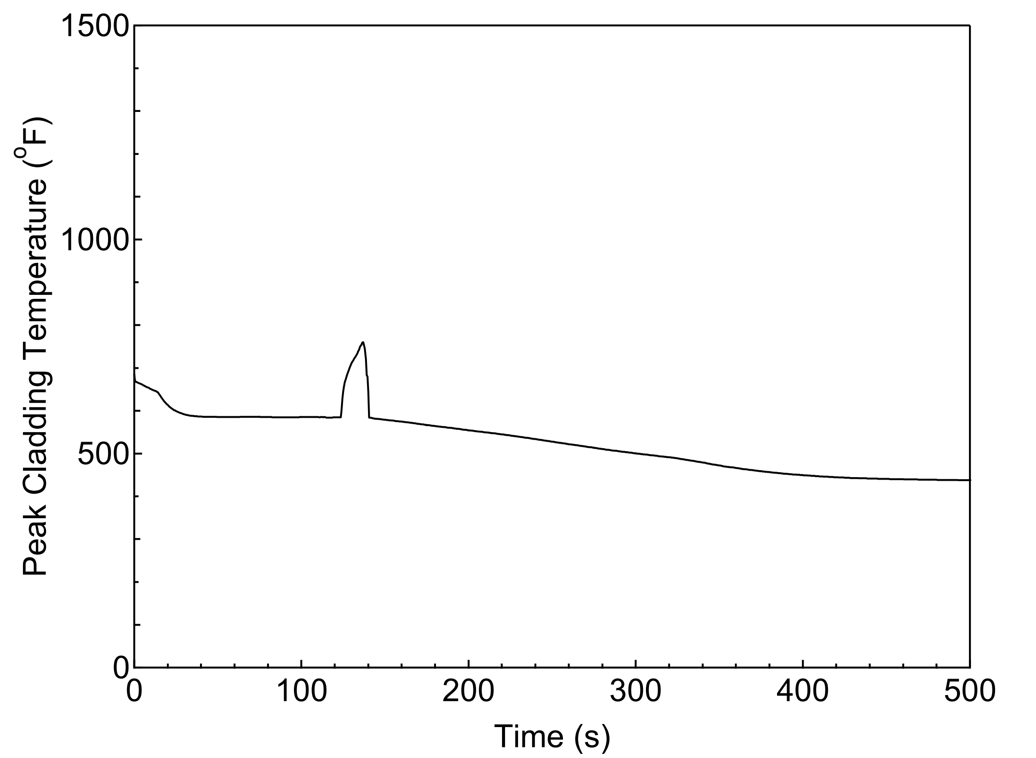


Figure 15.6.5-21 PCT at All Elevations for Hot Rod in Hot Assembly for 7.5-inch Small Break LOCA

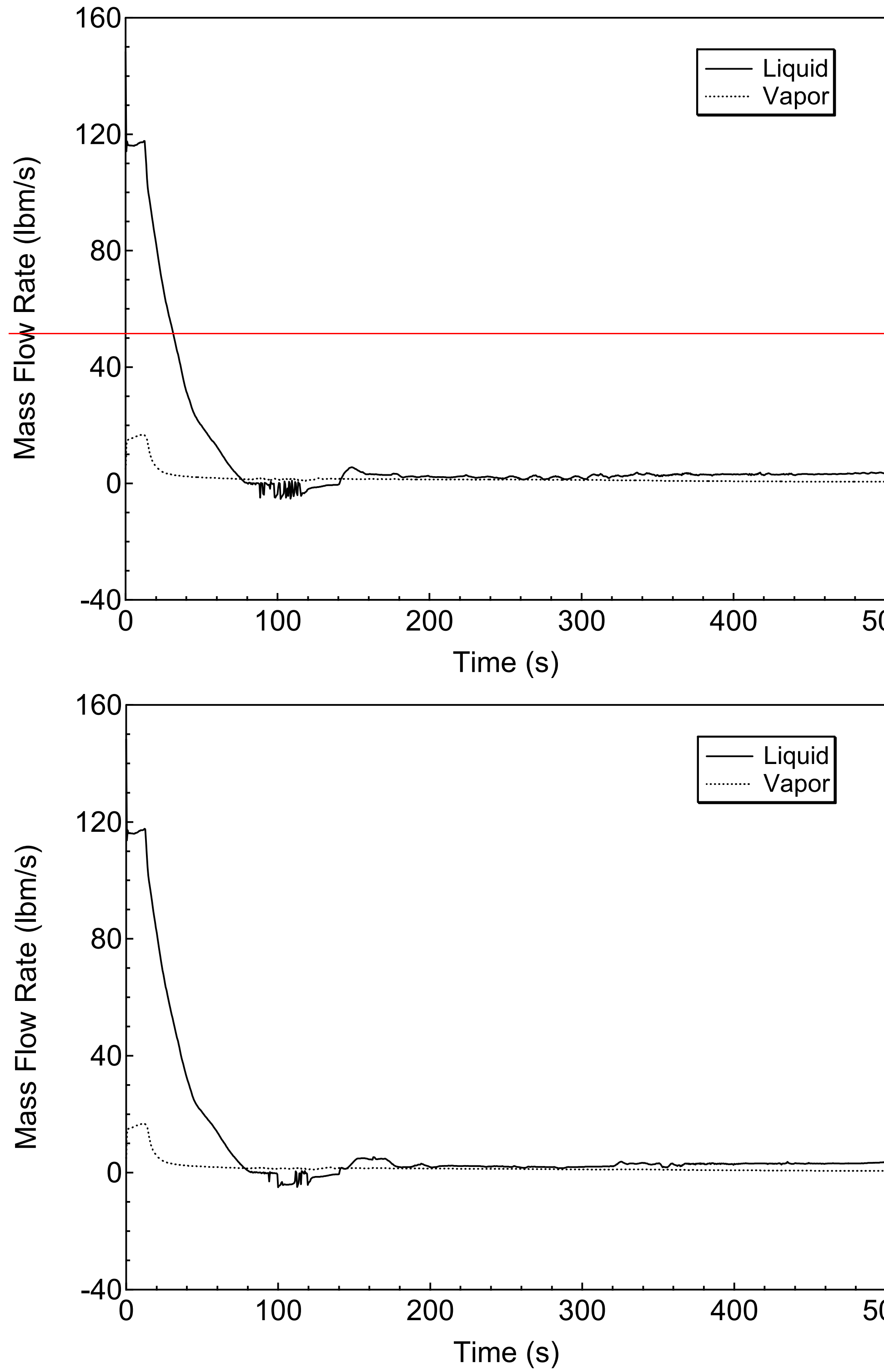


Figure 15.6.5-22 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 7.5-inch Small Break LOCA

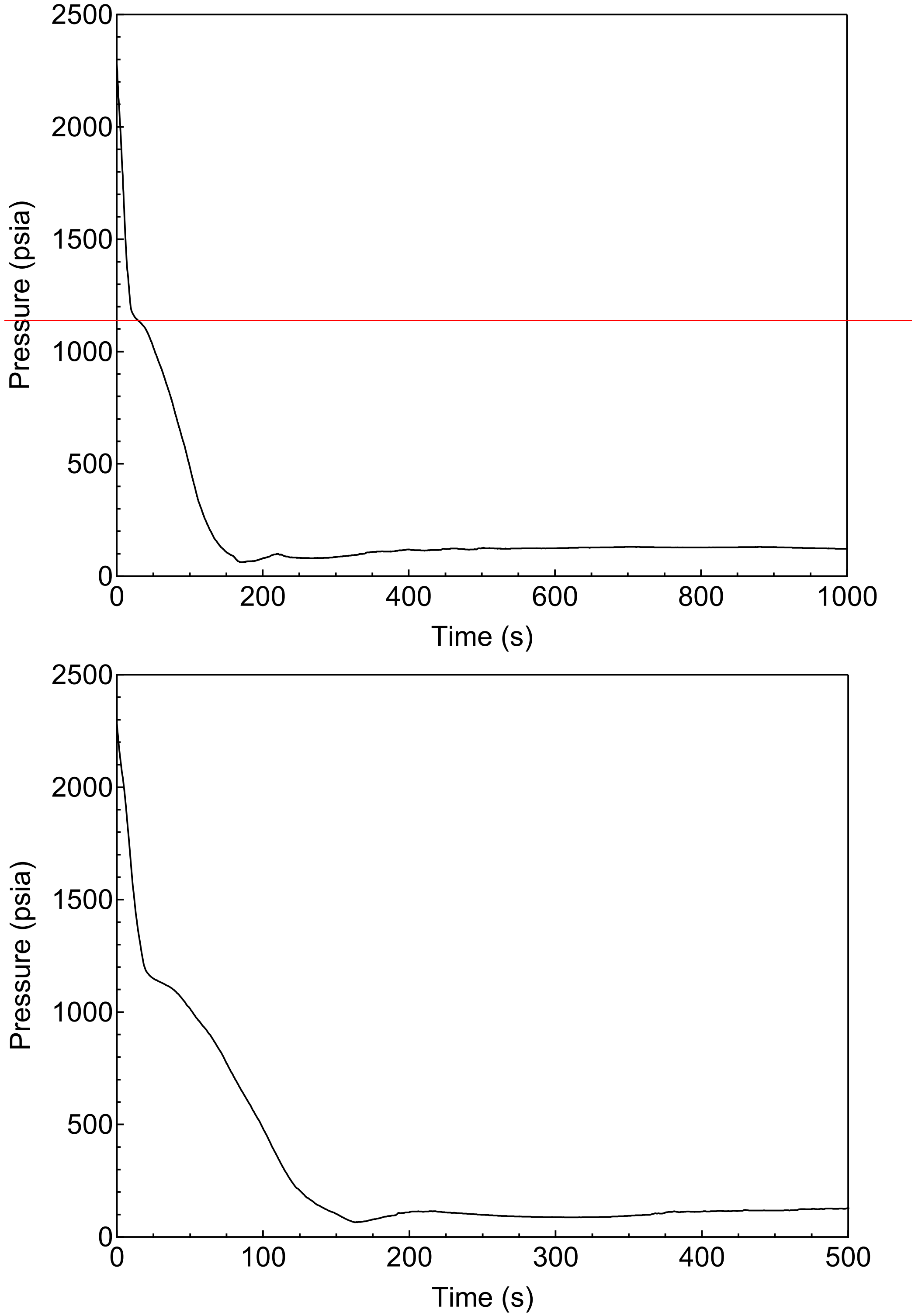


Figure 15.6.5-23 RCS (Pressurizer) Pressure Transient for 1-ft^2 Small Break LOCA

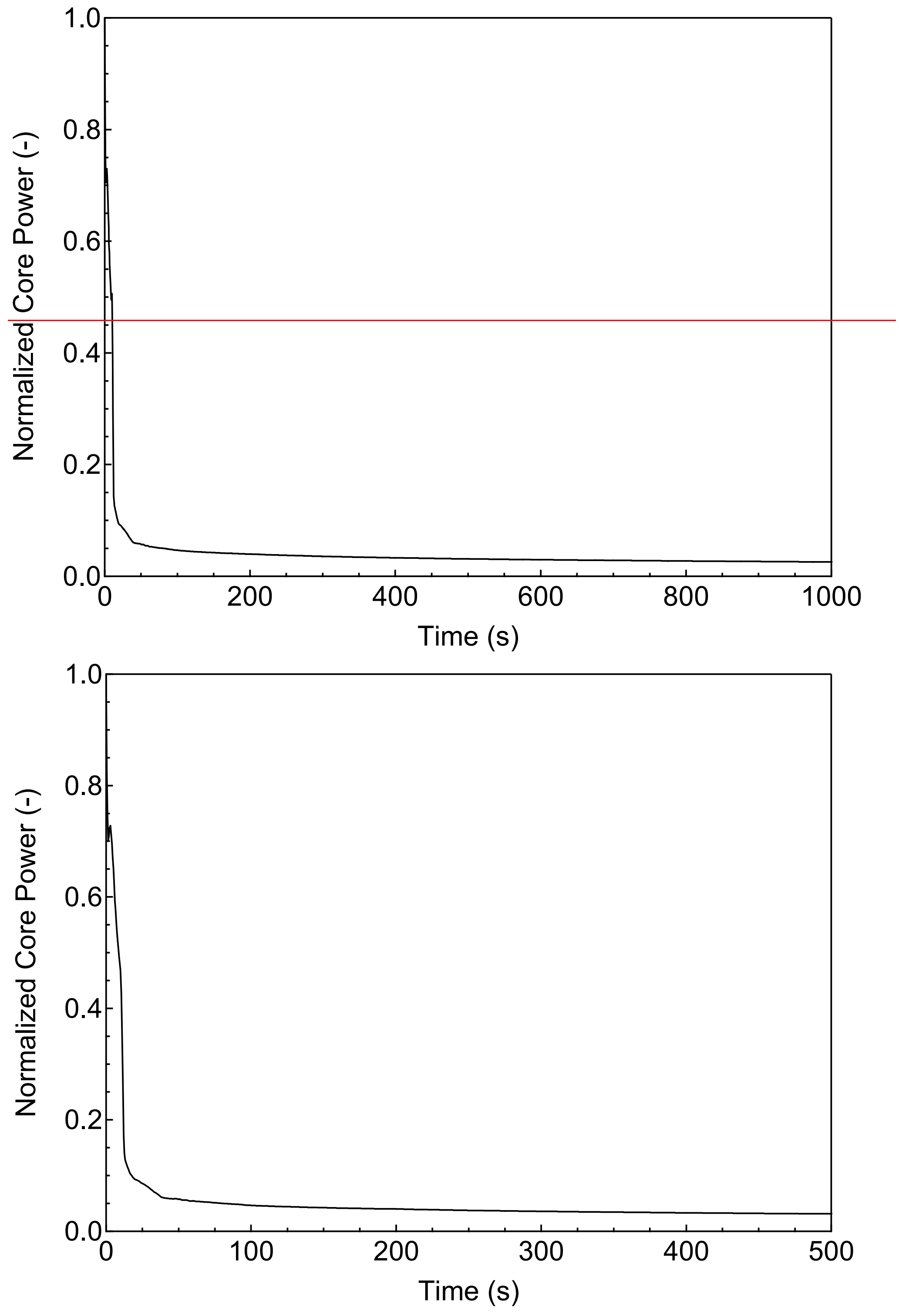


Figure 15.6.5-24 Normalized Core Power for 1-ft^2 Small Break LOCA

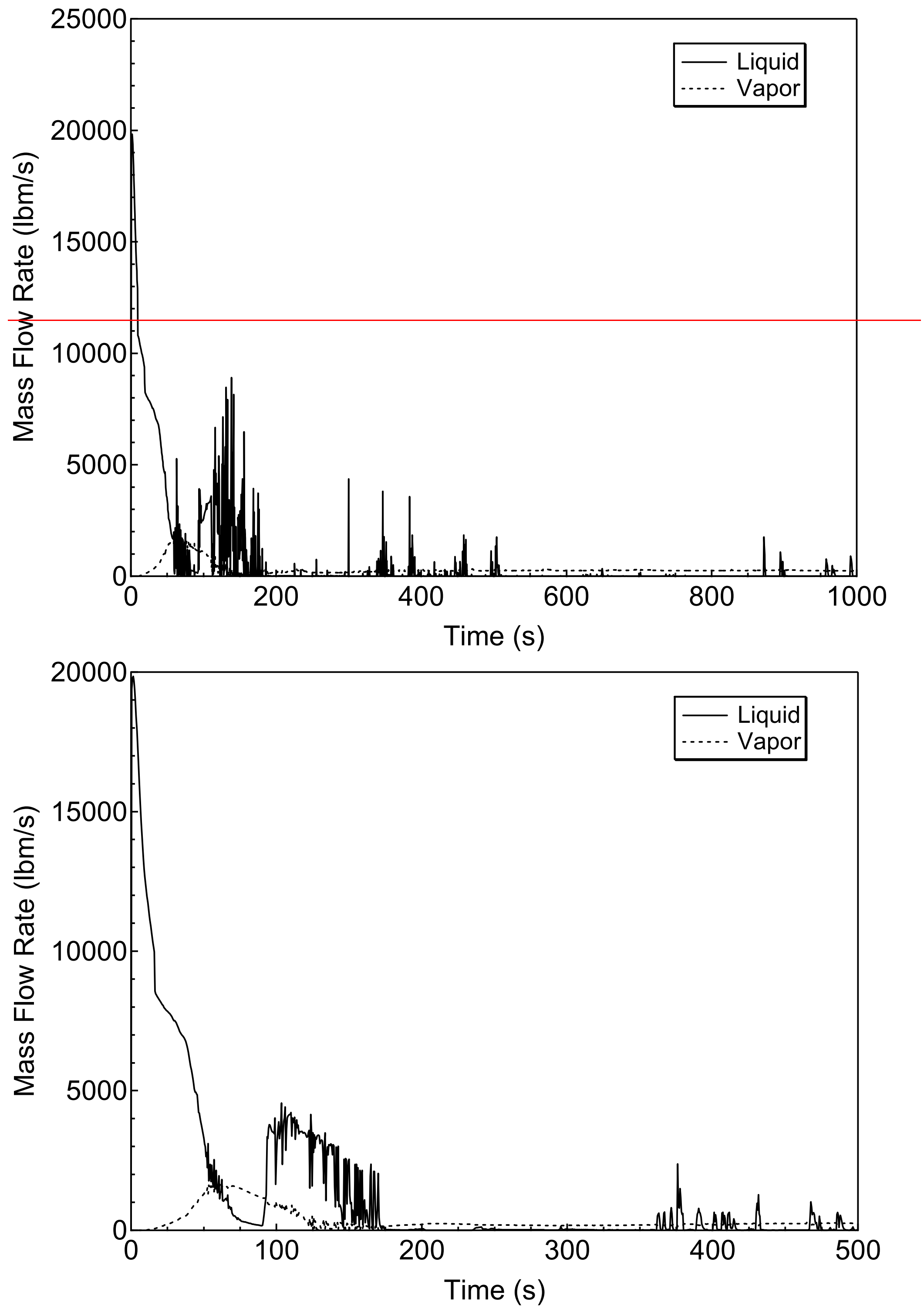


Figure 15.6.5-25 Liquid and Vapor Discharges through the Break for 1-ft^2 Small Break LOCA

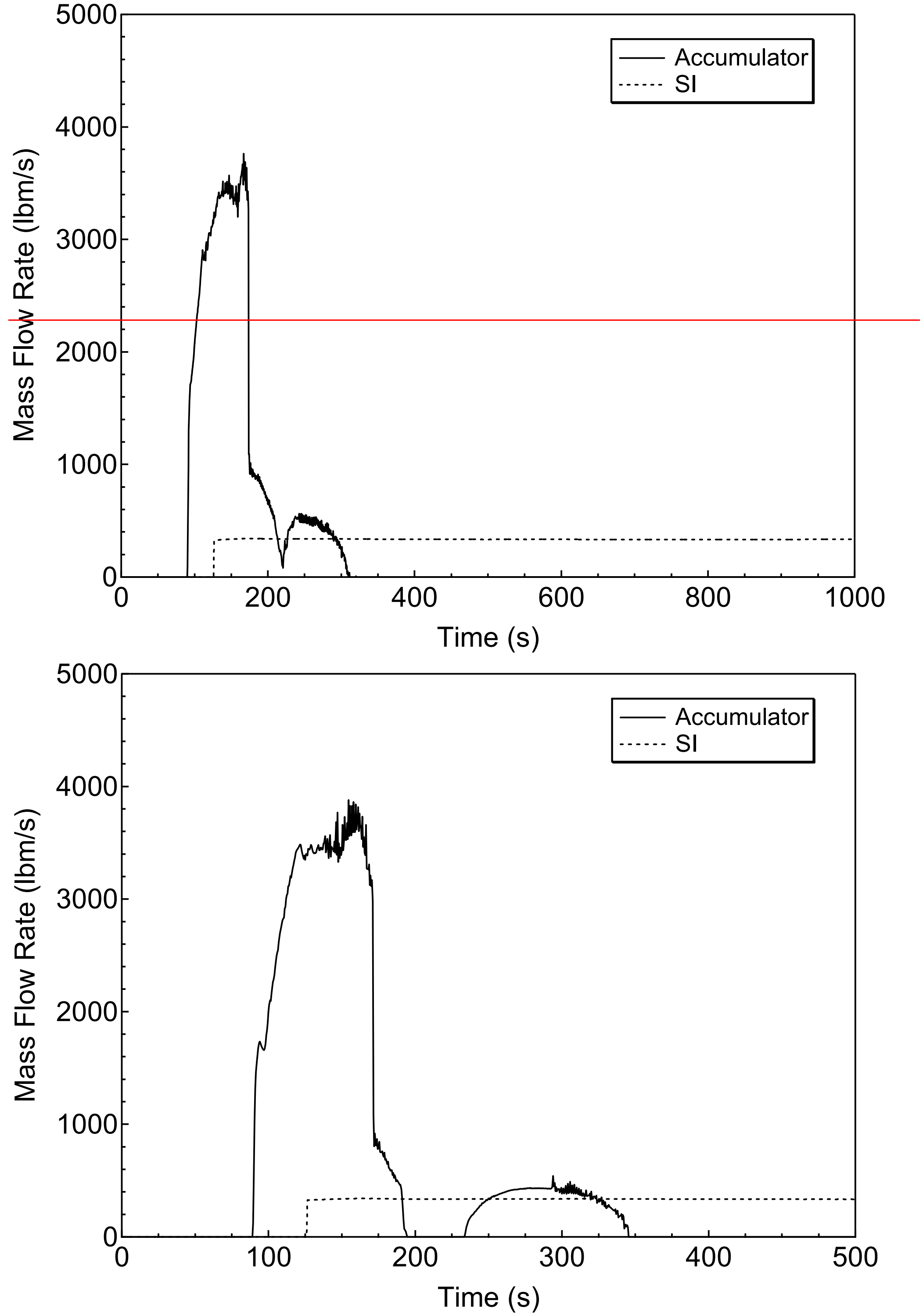


Figure 15.6.5-26 Accumulator and Safety Injection Mass Flowrates for 1-ft^2 Small Break LOCA

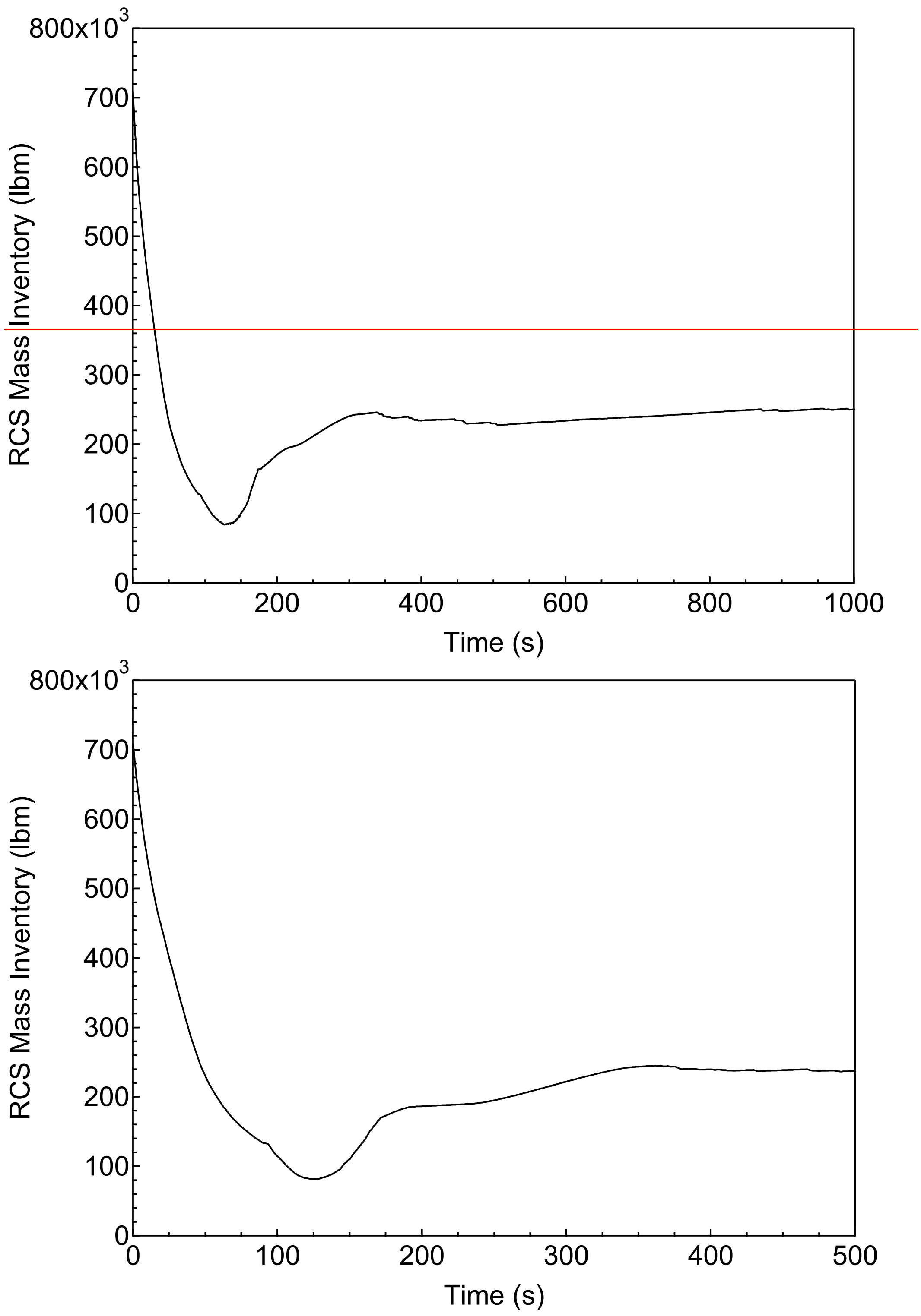
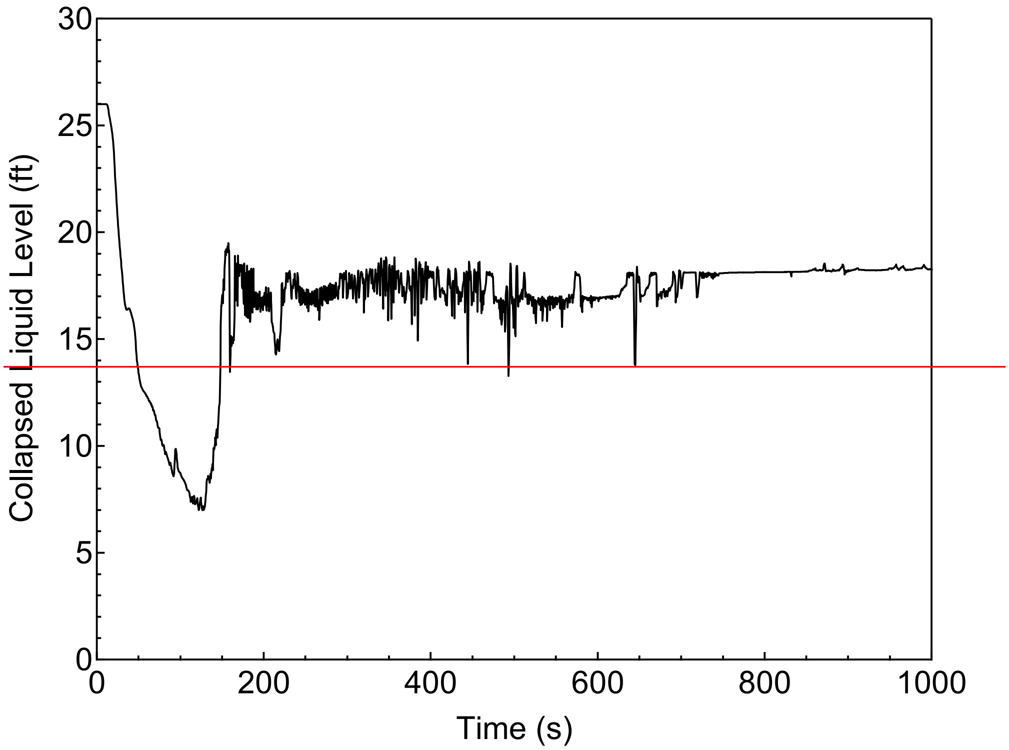


Figure 15.6.5-27 RCS Mass Inventory for 1-ft^2 Small Break LOCA



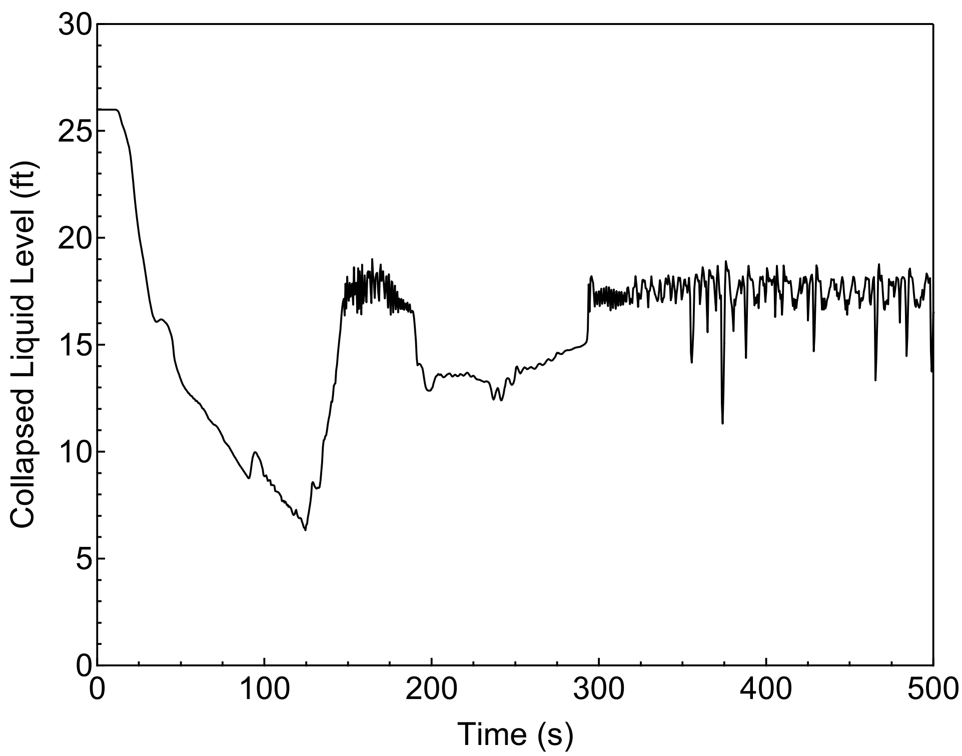


Figure 15.6.5-28 Downcomer Collapsed Level for 1-ft^2 Small Break LOCA

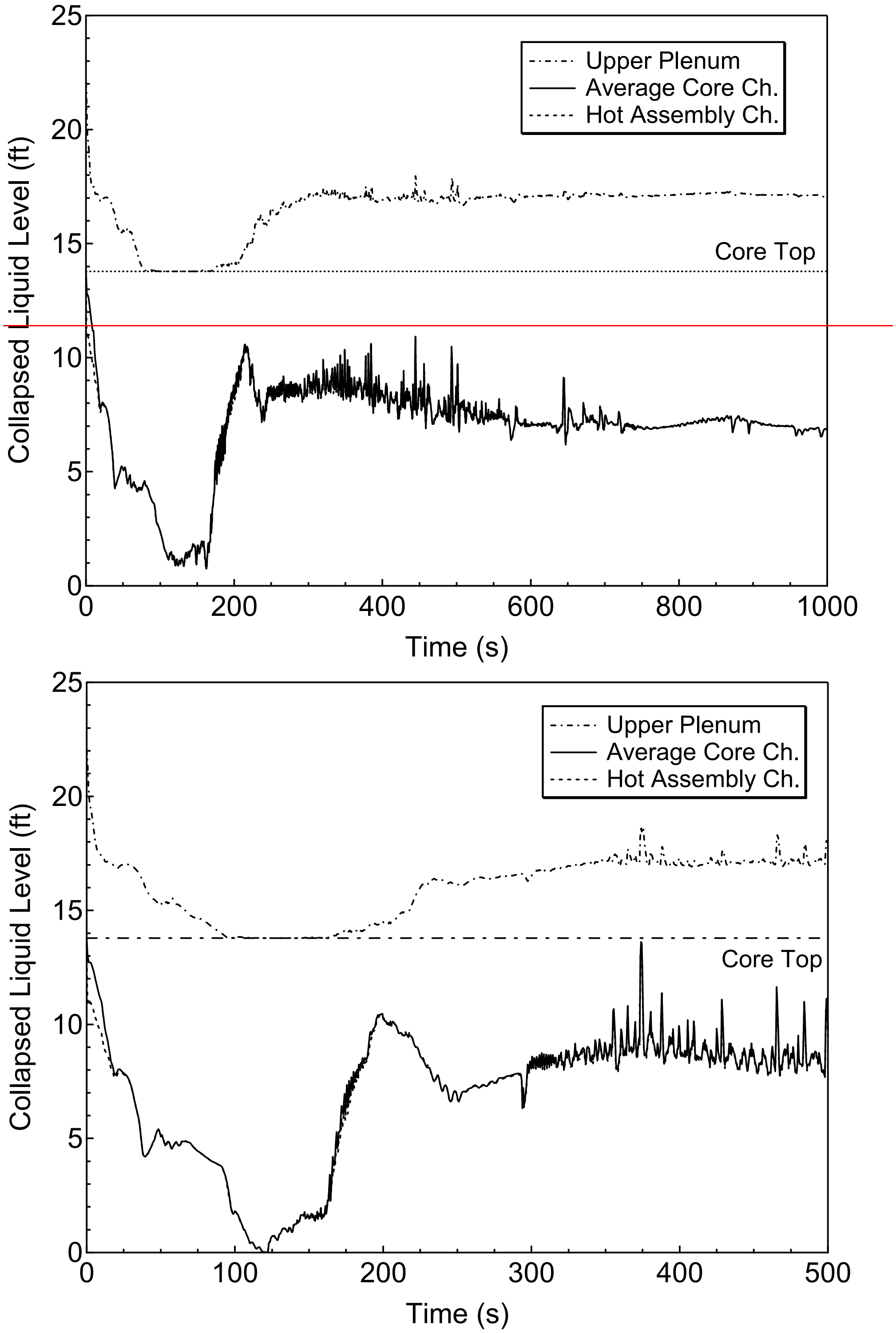


Figure 15.6.5-29 Core/Upper Plenum Collapsed Level for 1-ft^2 Small Break LOCA

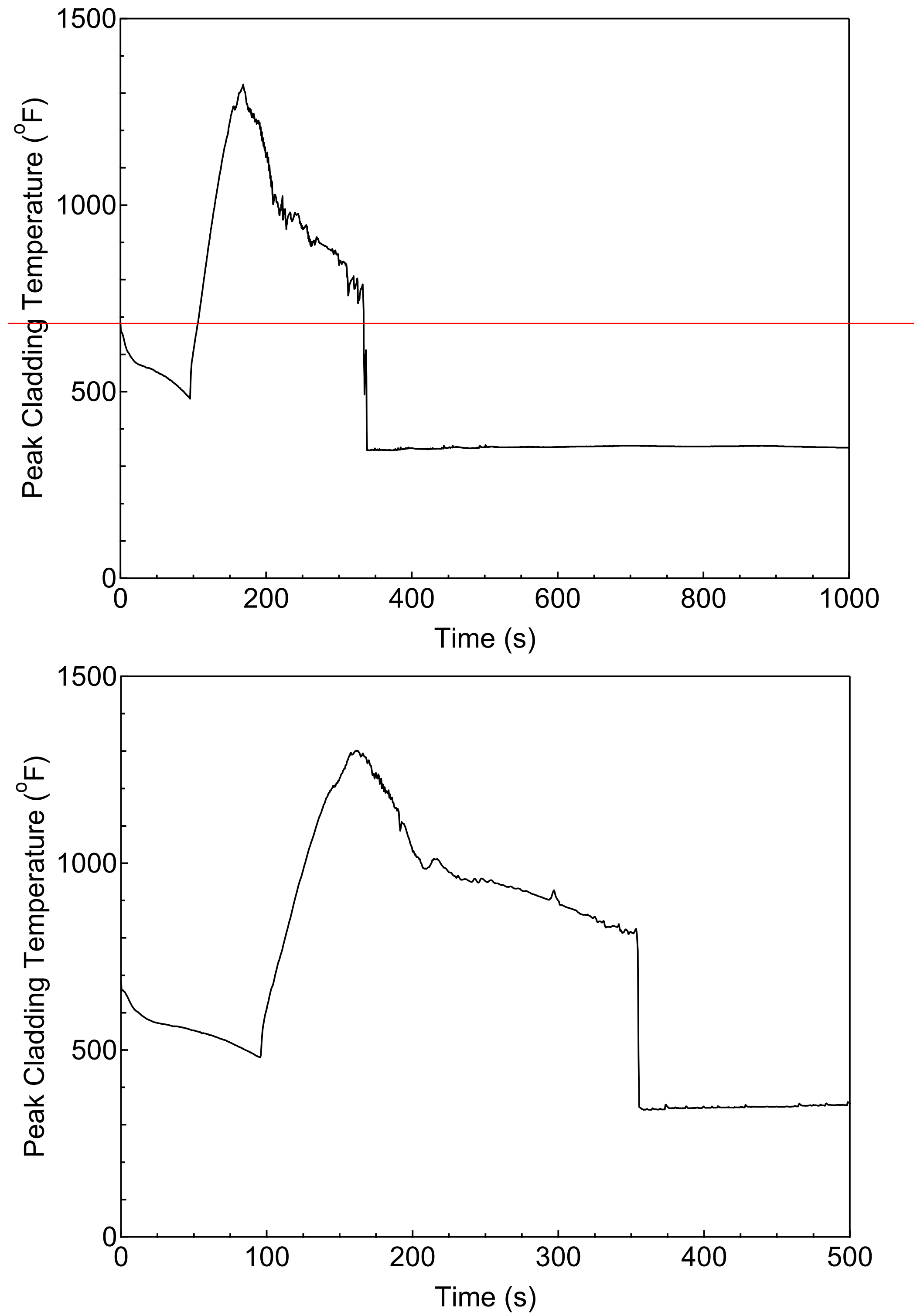


Figure 15.6.5-30 PCT at All Elevations for Hot Rod in Hot Assembly for 1-ft^2 Small Break LOCA

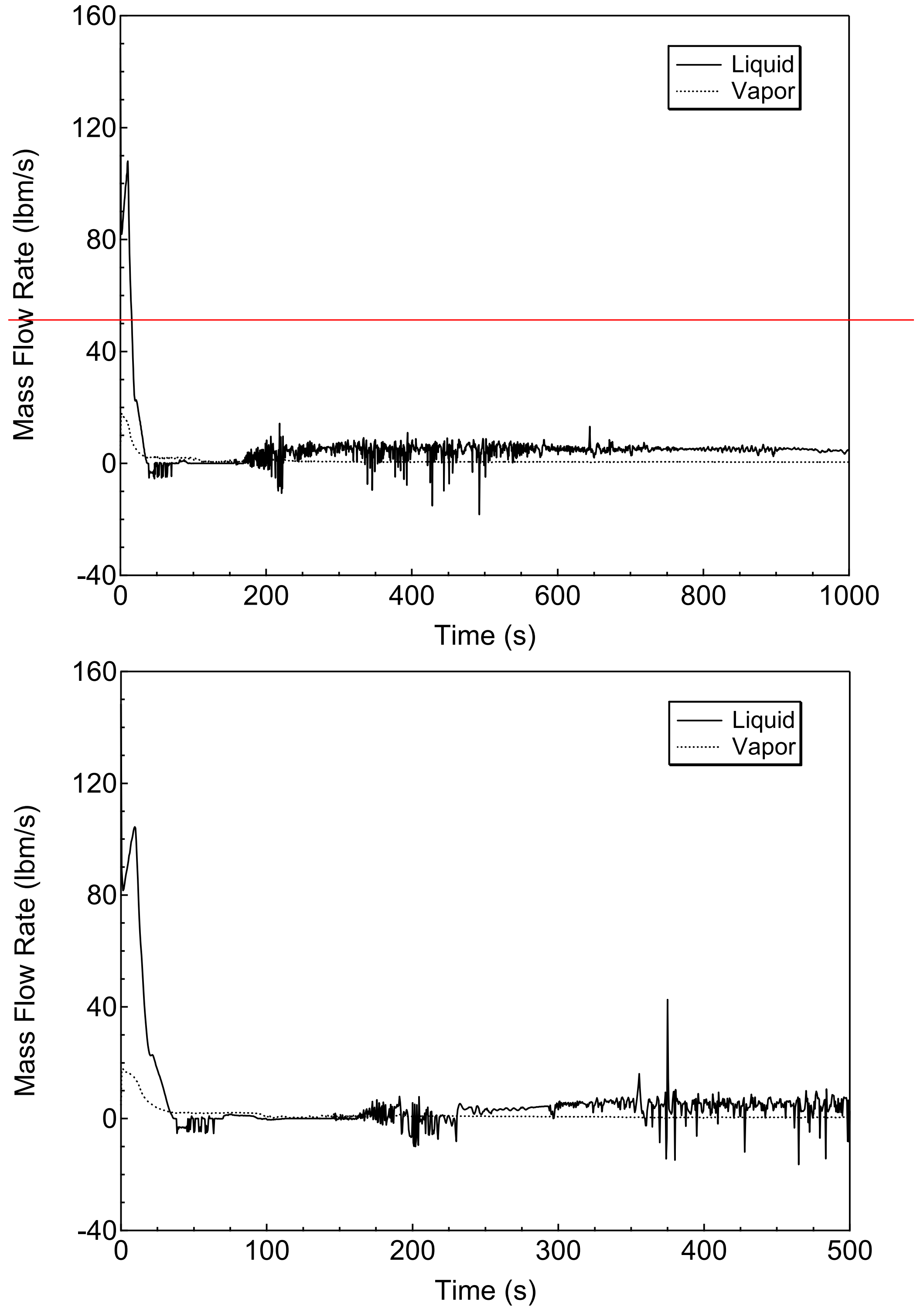
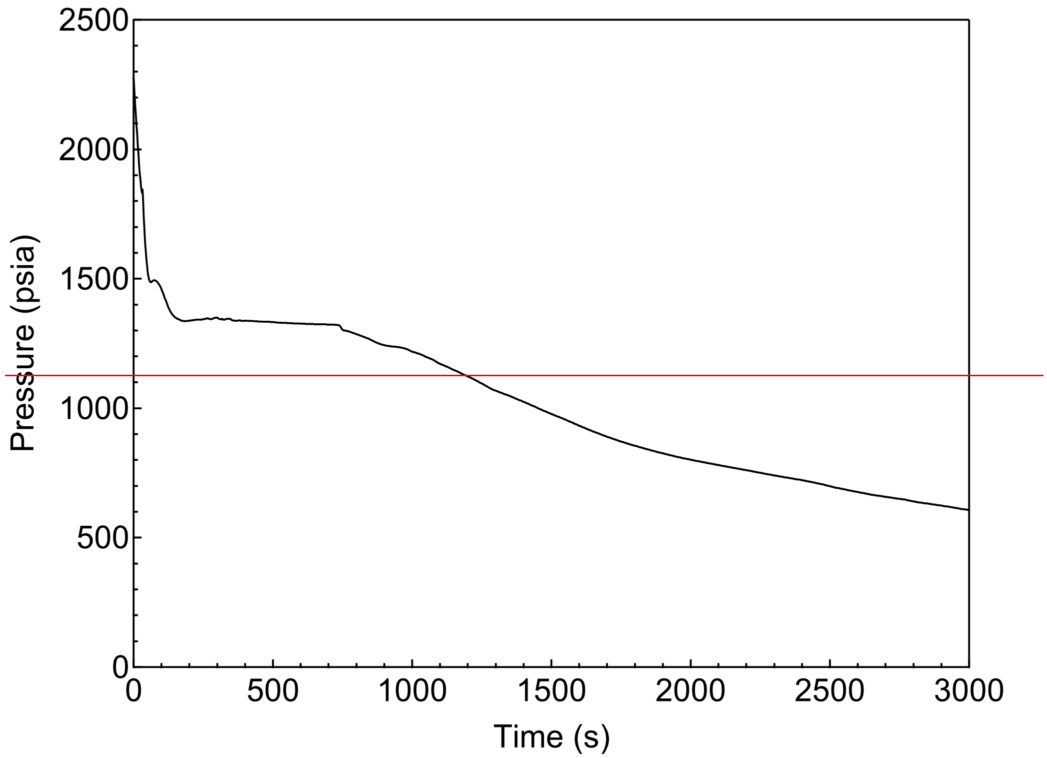


Figure 15.6.5-31 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 1-ft^2 Small Break LOCA



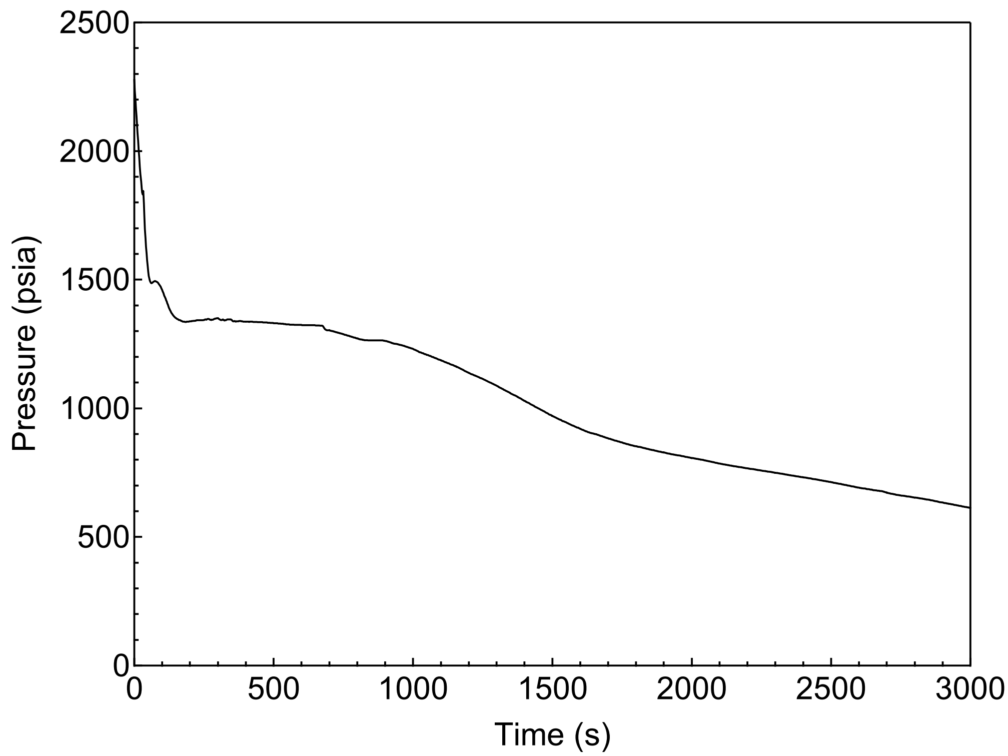


Figure 15.6.5-32 RCS (Pressurizer) Pressure Transient for DVI-line Small Break LOCA

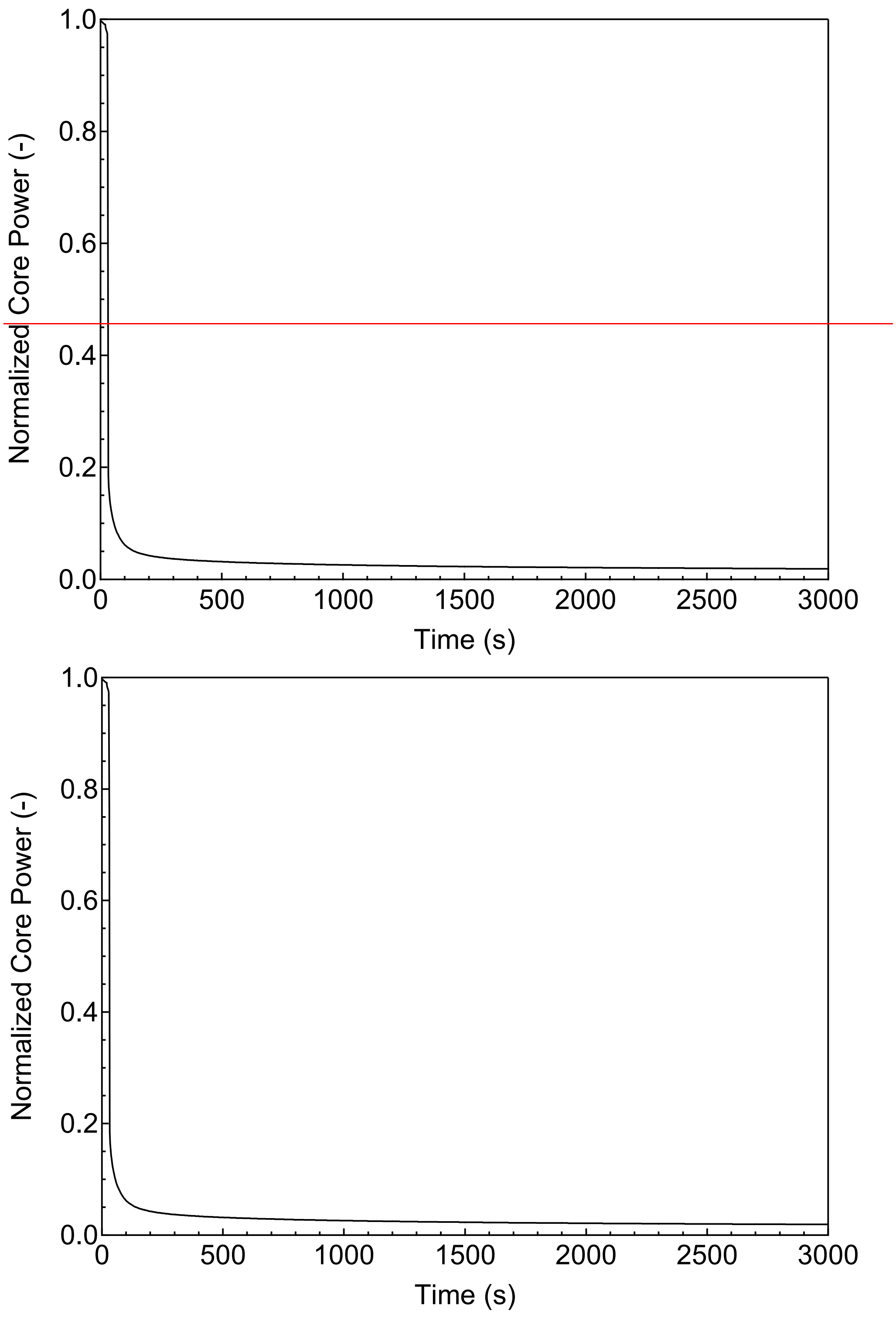


Figure 15.6.5-33 Normalized Core Power for DVI-line Small Break LOCA

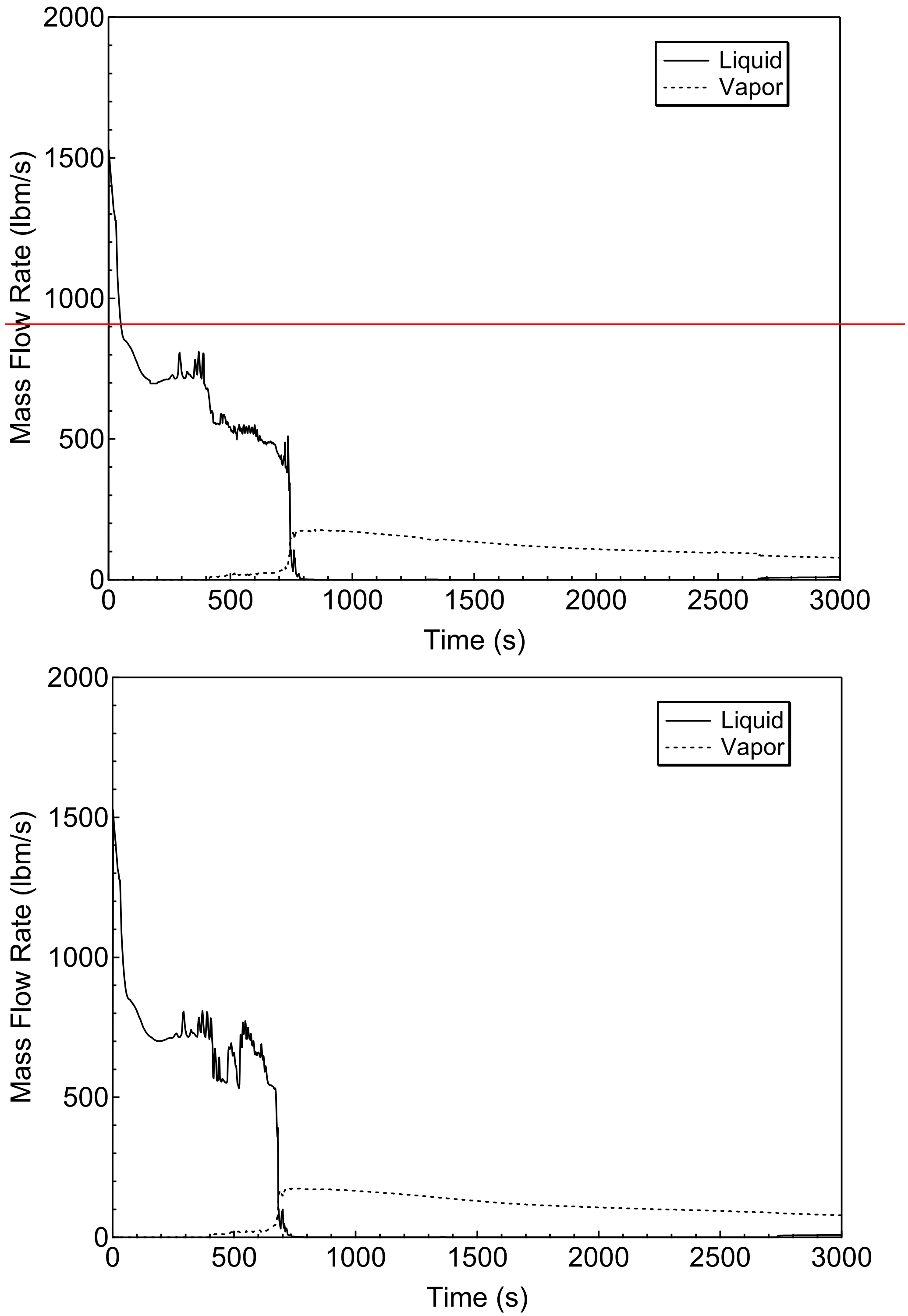
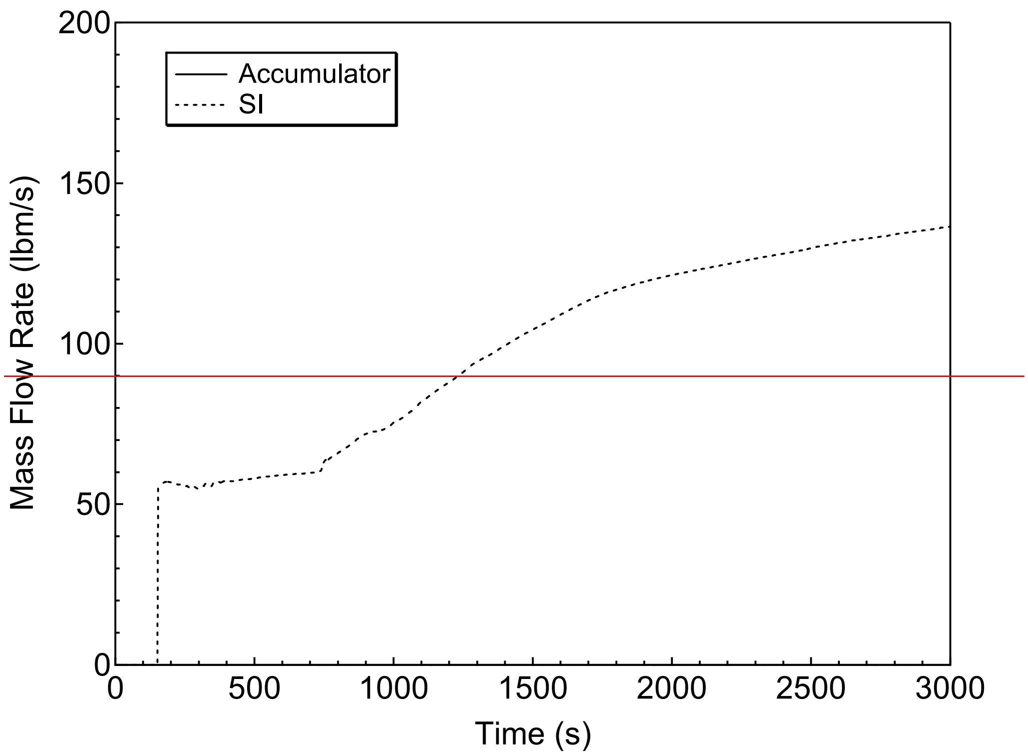


Figure 15.6.5-34 Liquid and Vapor Discharges through the Break for DVI-line Small Break LOCA



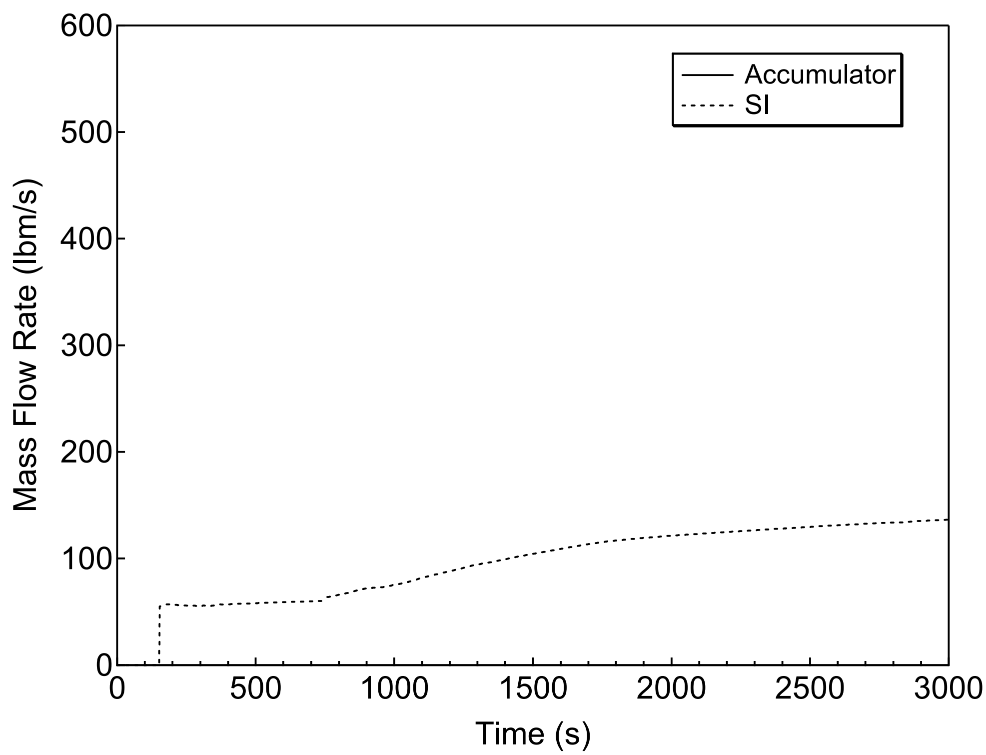
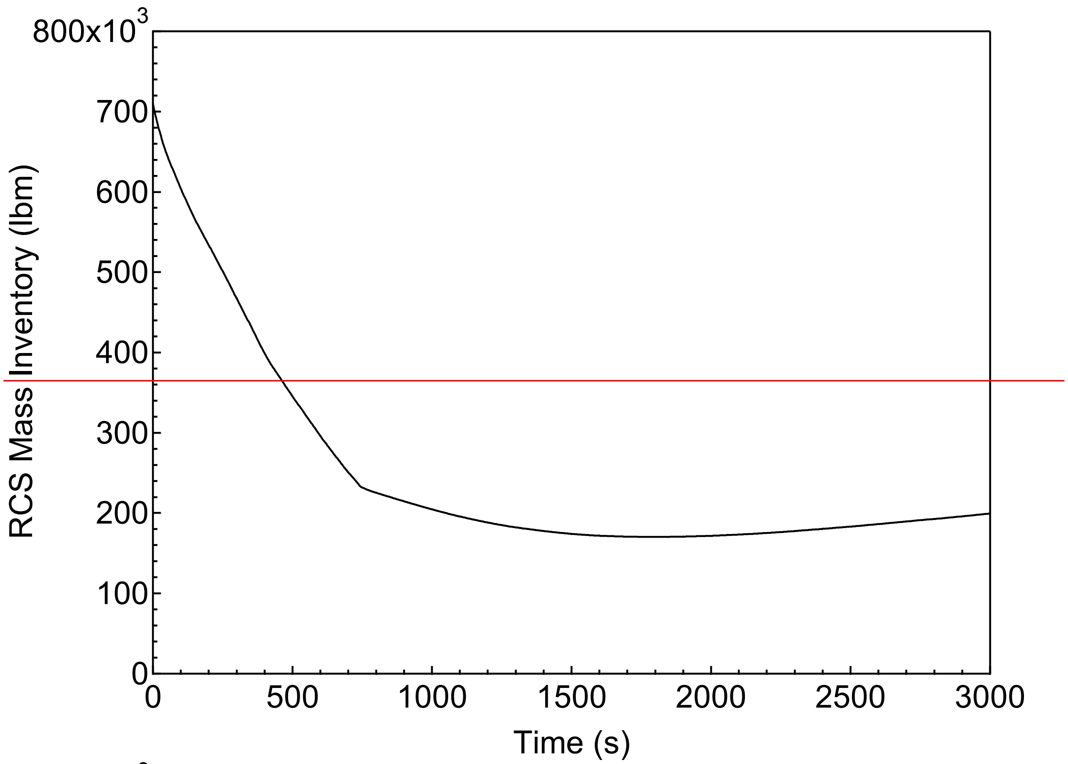


Figure 15.6.5-35 Accumulator and Safety Injection Mass Flowrates for DVI-line Small Break LOCA



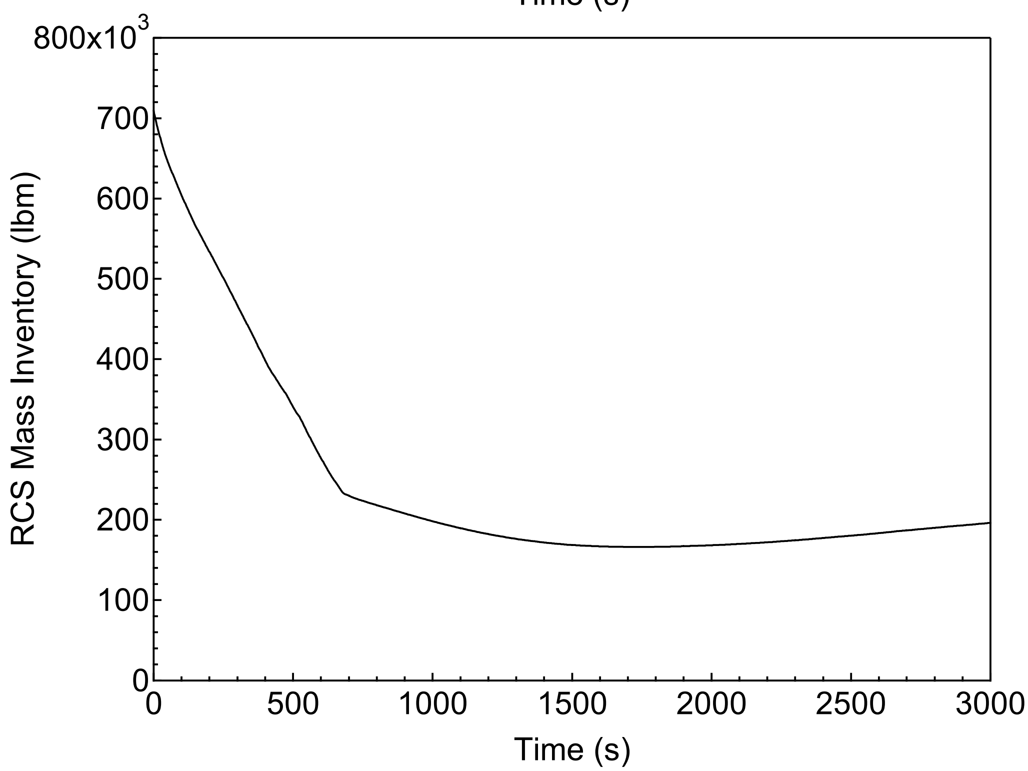


Figure 15.6.5-36 RCS Mass Inventory for DVI-line Small Break LOCA

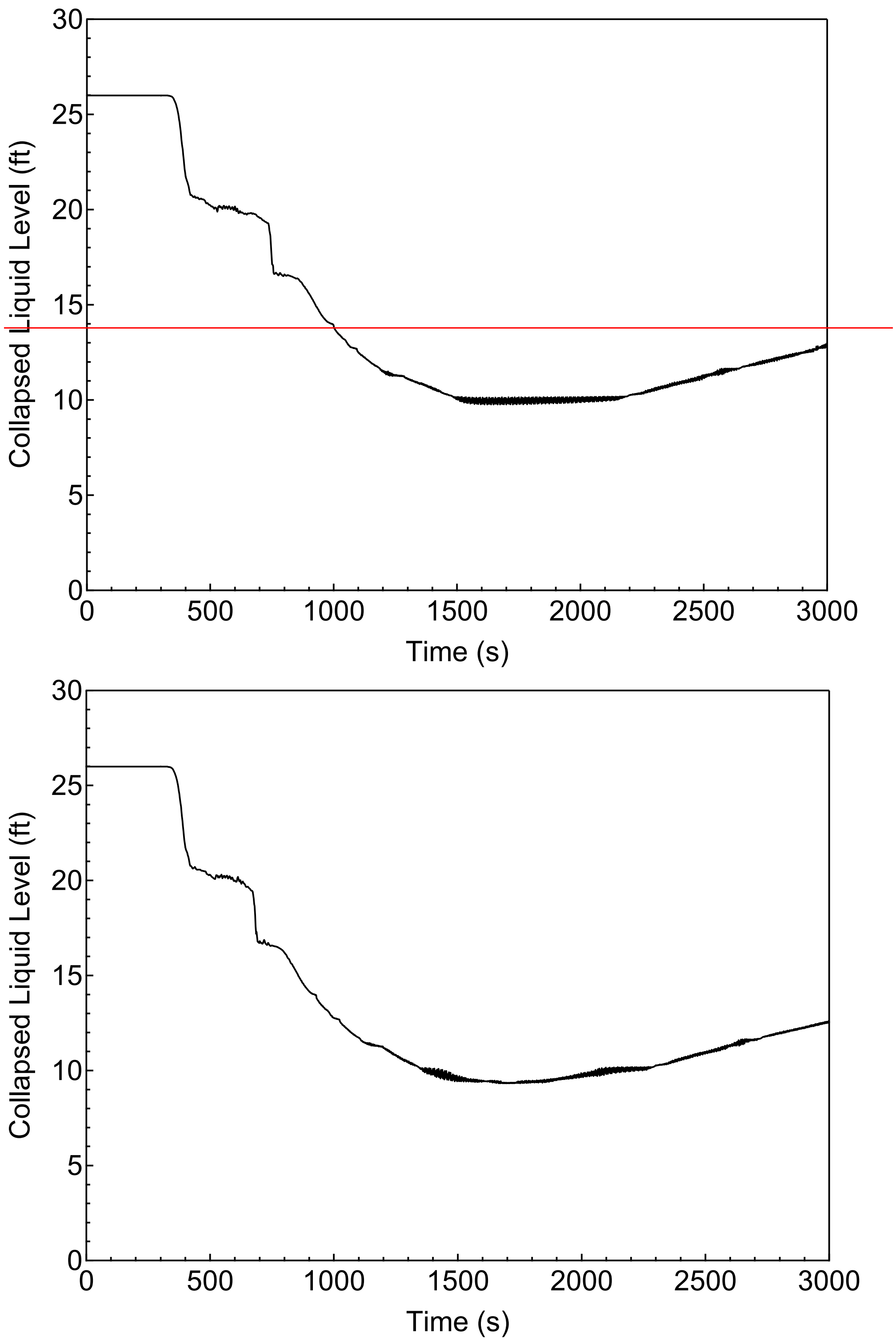
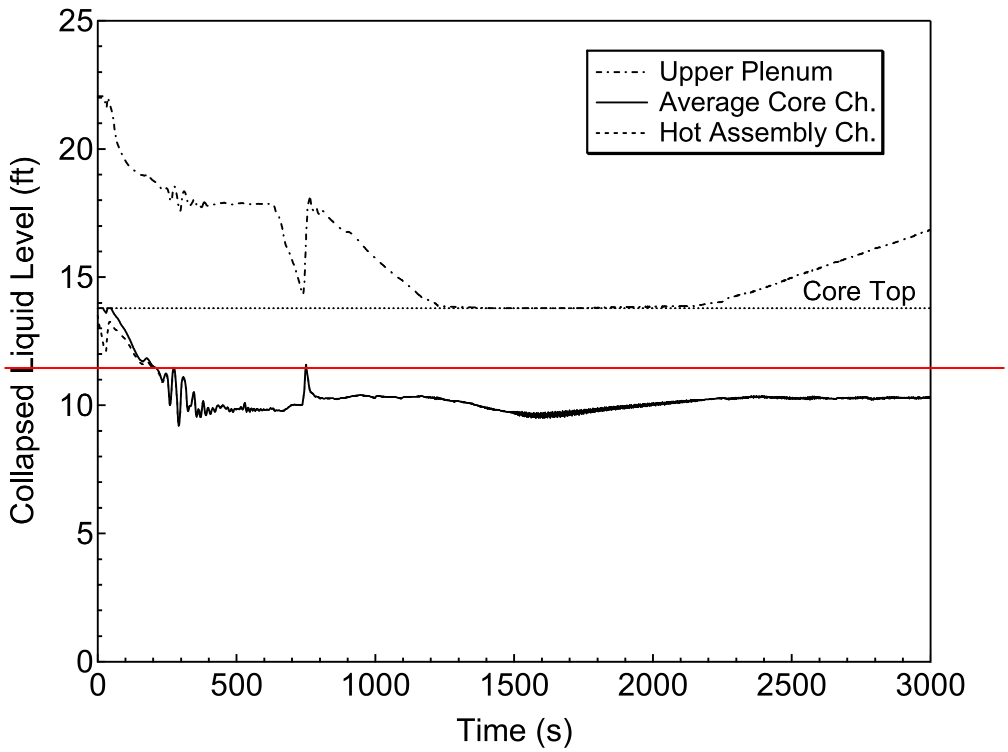


Figure 15.6.5-37 Downcomer Collapsed Level for DVI-line Small Break LOCA



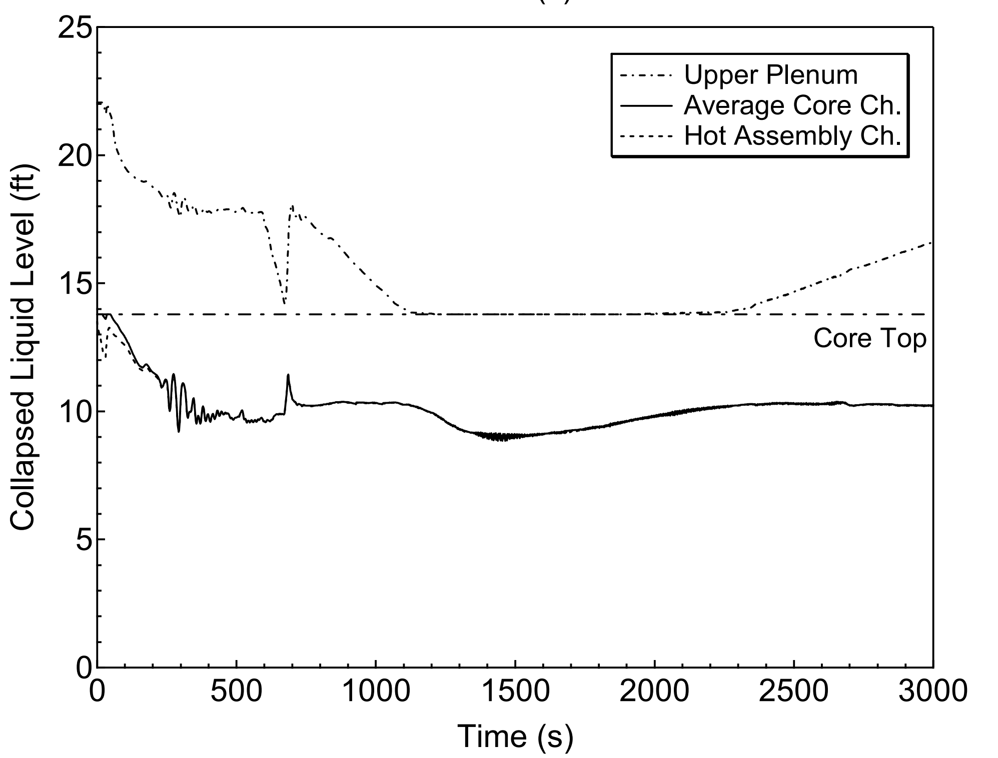


Figure 15.6.5-38 Core/Upper Plenum Collapsed Level for DVI-line Small Break LOCA

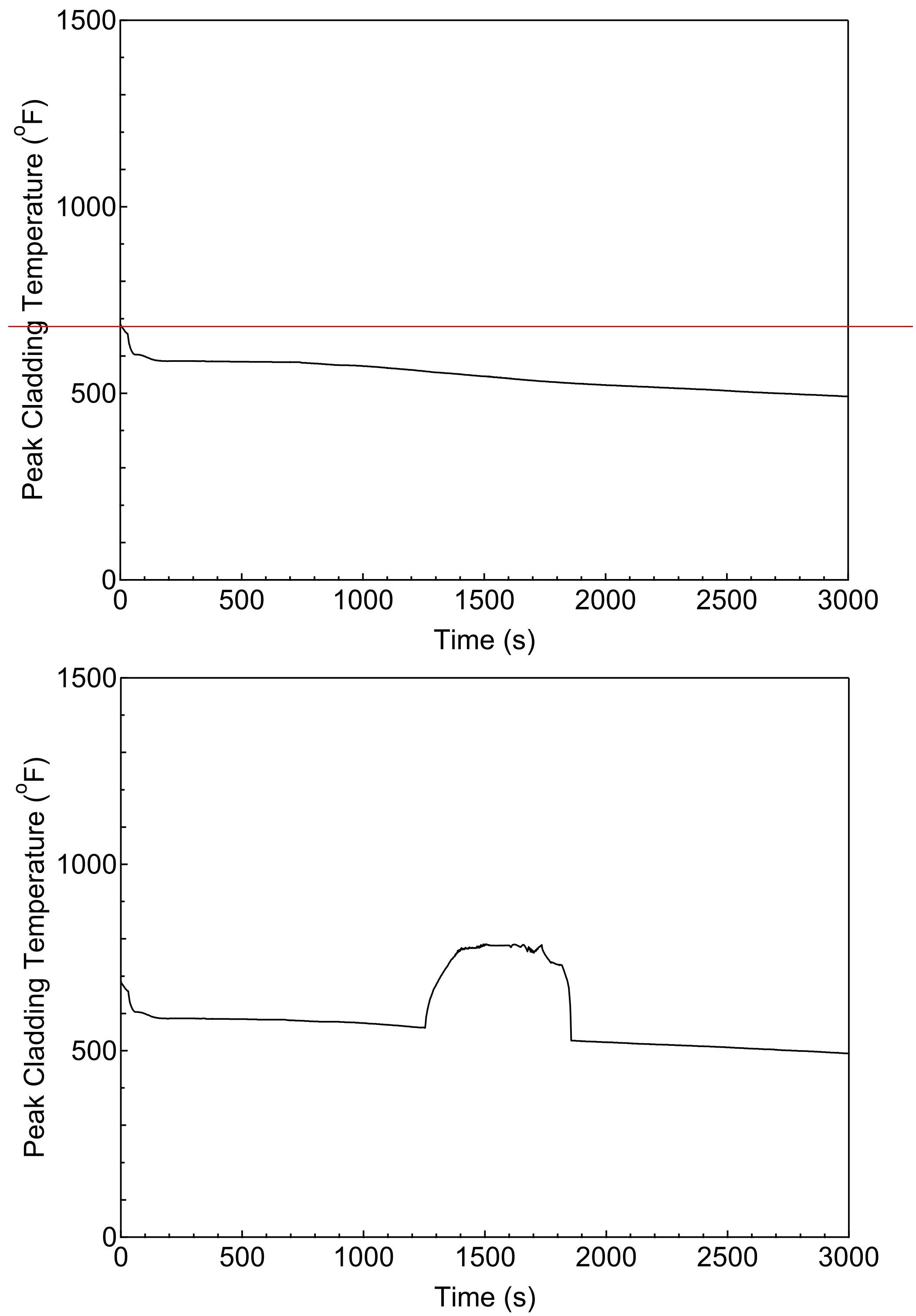


Figure 15.6.5-39 PCT at All Elevations for Hot Rod in Hot Assembly for DVI-line Small Break LOCA

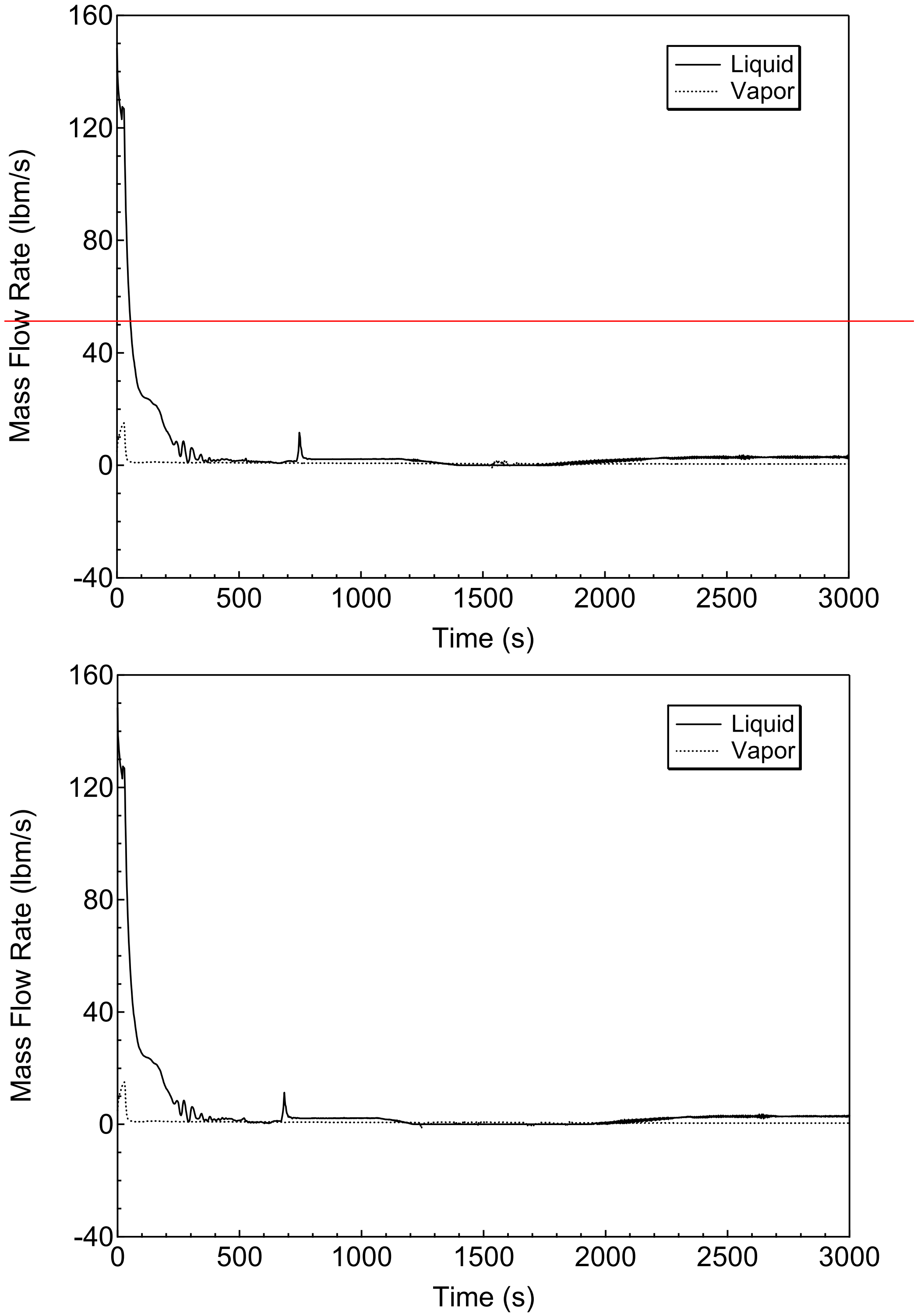
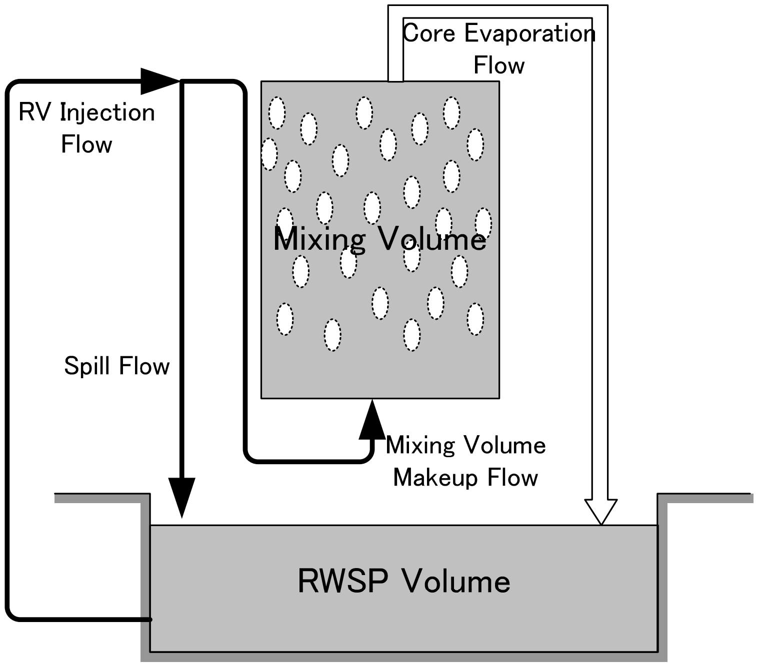
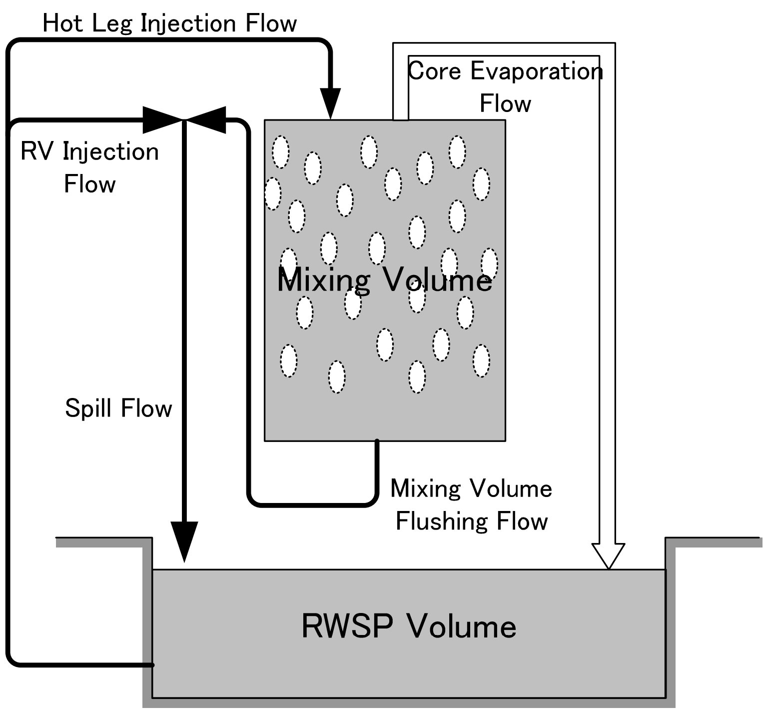


Figure 15.6.5-40 **Hot Assembly Exit Vapor and Liquid Mass Flowrates for DVI-line Small Break LOCA**



(a) RV injection mode



(b) Simultaneous RV and hot leg injection mode

Figure 15.6.5-41 Post-LOCA Long Term Cooling Evaluation Model

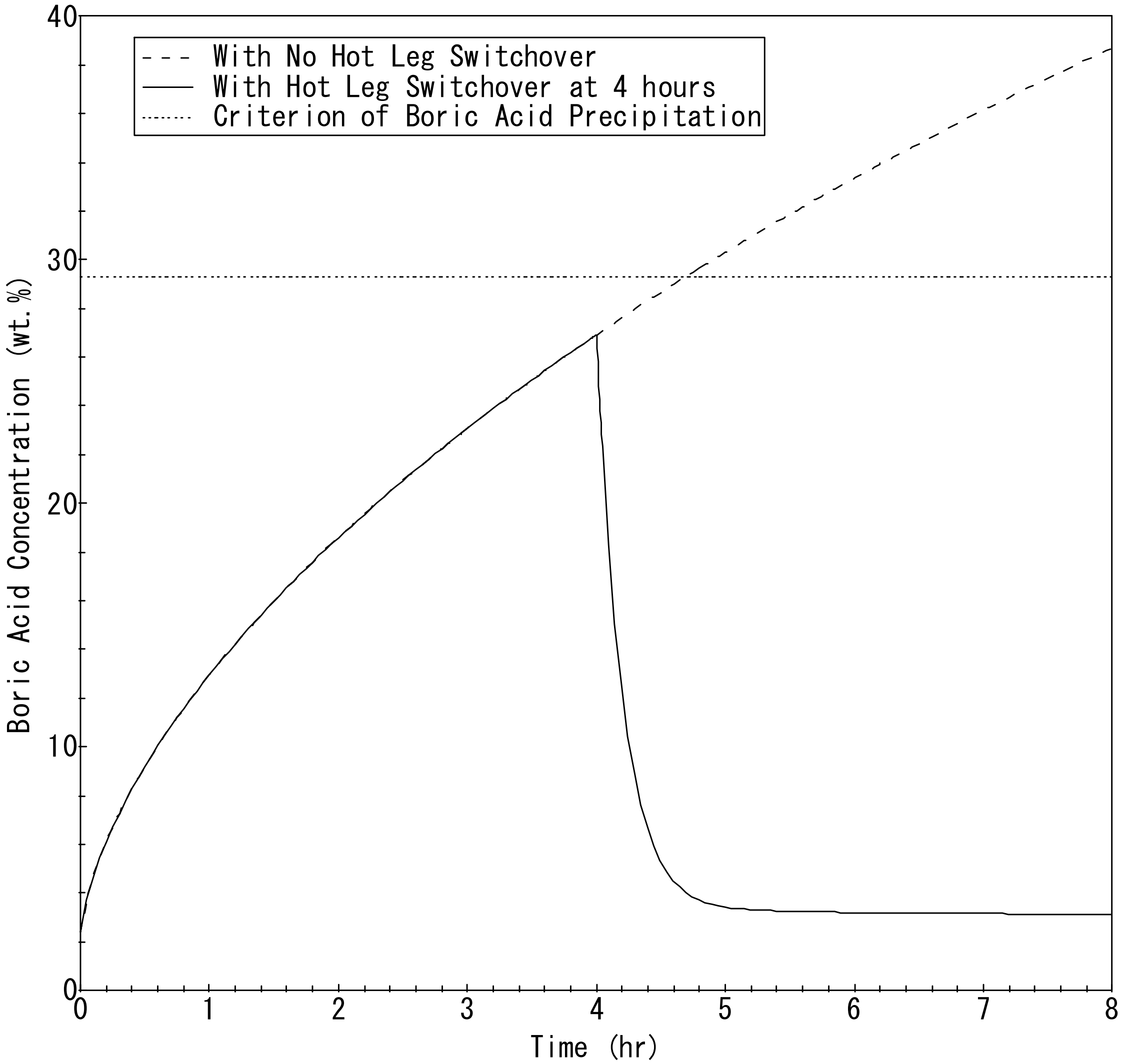


Figure 15.6.5-42 US-APWR Post LOCA Long Term Cooling Evaluation for 14.7 psia

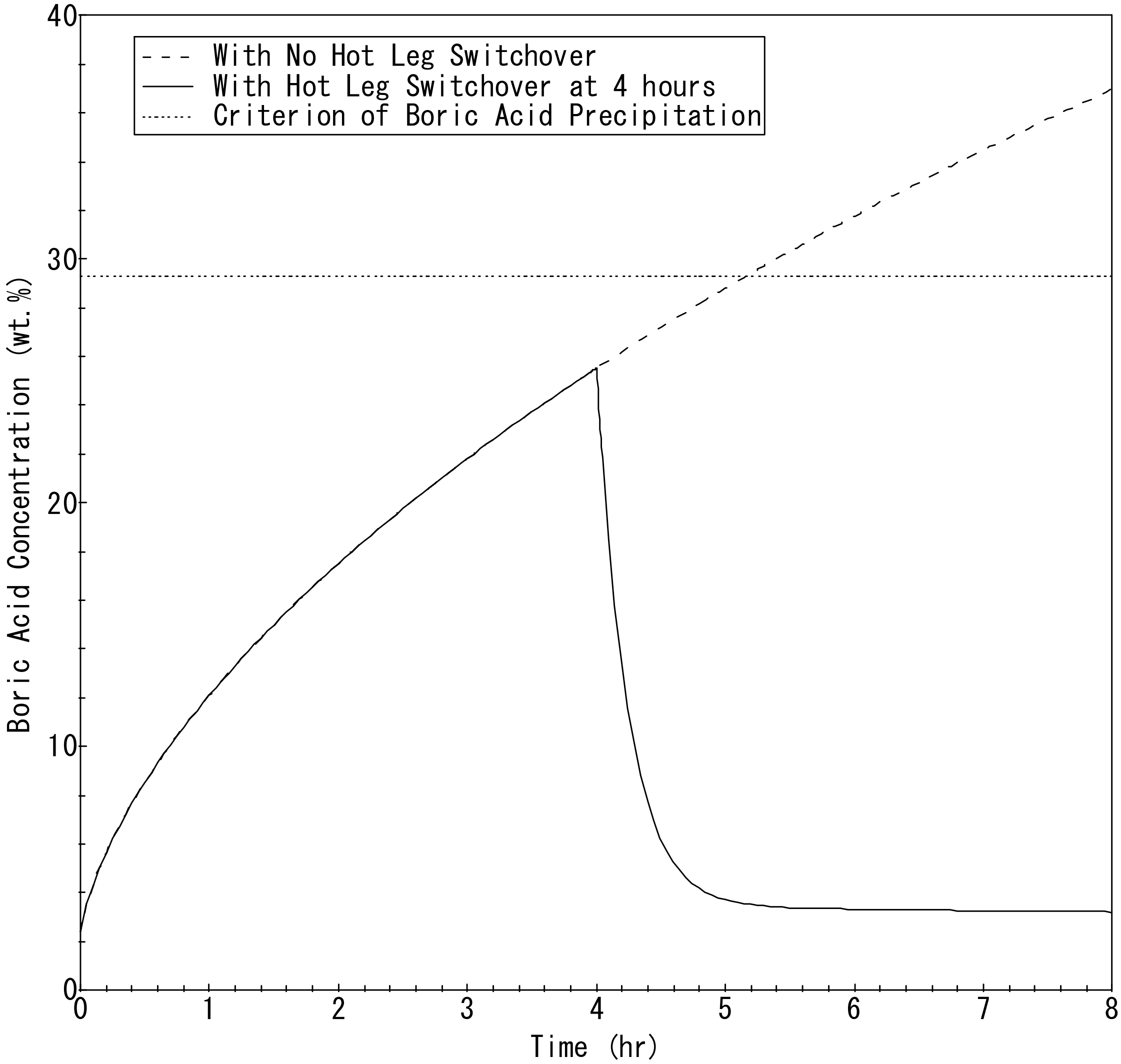


Figure 15.6.5-43 US-APWR Post LOCA Long Term Cooling Evaluation for 120 psia

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