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Your ref: Docket No. 52-006
Our ref: DCP/NRC2128

April 29, 2008

Subject: AP1000 COL Response to Request for Additional Information (TR 29)

Westinghouse is submitting a revised response to the NRC request for additional information (RAI) on AP1000 Standard Combined License Technical Report (TR) 29, APP-GW-GLR-044, Nuclear Island Basemat and Foundation. This RAI response is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in the response is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

A revised response is provided for RAI-TR29-SRSB-01 as sent in an email from Billy Gleaves to Don Lindgren, dated November 9, 2007. This response completes all requests received to date for TR 29. Revision 0 of the response to RAI-TR29-SRSB-01 was submitted under letter DCP/NRC2065 dated January 3, 2008.

Pursuant to 10 CFR 50.30(b), the response to the request for additional information on TR 29 is submitted as Enclosure 1 under the attached Oath of Affirmation.

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

D063
NRO

/Attachment

1. "Oath of Affirmation," dated April 29, 2008

/Enclosure

1. Response to Request for Additional Information on Technical Report 29

cc:	B. Gleaves	-	U.S. NRC	1E	1A
	E. McKenna	-	U.S. NRC	1E	1A
	P. Ray	-	TVA	1E	1A
	P. Hastings	-	Duke Power	1E	1A
	R. Kitchen	-	Progress Energy	1E	1A
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	C. Pierce	-	Southern Company	1E	1A
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	E. Schmiech	-	Westinghouse	1E	1A
	J. Monahan	-	Westinghouse	1E	1A

ATTACHMENT 1

"Oath of Affirmation"

ATTACHMENT 1

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:)

AP1000 Design Certification Amendment Application)

NRC Docket Number 52-006)

APPLICATION FOR REVIEW OF
"AP1000 GENERAL INFORMATION"
FOR DESIGN CERTIFICATION AMENDMENT APPLICATION REVIEW

B. W. Bevilacqua, being duly sworn, states that he is Vice President, New Plants Engineering, for Westinghouse Electric Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this document; that all statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

Bruce W. Bevilacqua

B. W. Bevilacqua
Vice President
New Plants Engineering

Subscribed and sworn to
before me this 29th day
of April 2008.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia S. Aston, Notary Public
Murrysville Boro, Westmoreland County
My Commission Expires July 11, 2011
Member, Pennsylvania Association of Notaries

Patricia S. Aston
Notary Public

ENCLOSURE 1

Response to Request for Additional Information on Technical Report 29

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-TR29-SRSB-01
Revision: 1

Question:

In technical report TR29, WCAP-16716-NP, Westinghouse proposed changes to the AP1000 Internals design. These proposed changes include (1) the addition of a flow skirt in the lower reactor vessel head, and (2) the addition of four neutron panels attached to the outside surface of the core barrel. These changes would affect the core inlet flow distribution and the flow area and flow resistance in the reactor downcomer and lower plenum. The applicant should demonstrate that these changes would not have significant adverse impact on the thermal-hydraulic behavior of the design basis transients and accidents.

Please provide an evaluation of the impacts of these proposed changes on the analysis results of each of the transients and accidents in DCD Chapter 15. The evaluation should demonstrate either the analysis results of DCD Chapter 15, Revision 15, remain bounding and valid, or the effects of these changes are insignificantly small that the applicable acceptance criteria for these transients and accidents remain complied with.

Westinghouse Response:

This document presents the results of evaluations and safety analyses of reactor vessel internals design changes (Reference 1) as described in Standard Combined License Technical Report Number 29. The updated analyses were performed using methods and computer codes previously reviewed and approved by the NRC in NUREG 1793 (Reference 8). The updated analyses presented here also include the pressurizer changes described in Technical Report 36 (Reference 9) and analyzed/evaluated in the response to RAI-TR36-012 (Reference 10).

As described in References 1 and 2, the following reactor vessel internals changes are made:

1) Relocation of radial support keys and tapered periphery on lower core support plate (LCSP)
2) Addition of flow skirt to the reactor vessel lower head
3) Addition of the neutron panels
4) Increase in radius of reactor vessel

These changes could have impact on the following input used in safety analyses:

- reactor vessel fluid volumes
- reactor vessel metal masses and metal stored energy
- reactor vessel downcomer and lower plenum flow areas
- distribution of pressure drops through the vessel and overall reactor vessel pressure drop

These reactor internals changes have minimal impact on the fluid volumes, metal masses, and pressure drop input used in the safety analyses; they also do not change the overall pressure drop through the reactor vessel enough to impact the design RCS flow rates used in the safety analyses. The design RCS flows given in Table 15.0-3 in Revision 16 of the DCD (Reference 3) remain unchanged. For example, at 0% and 10% tube plugging for non-DNBR calculations, the reactor coolant loop flows of 149,900 gpm and 148,000 gpm remain unchanged. The other design operating conditions given in the aforementioned Table 15.0-3, which are used as the initial full power operating conditions of safety analyses, also remain unaffected by the reactor internals changes.

The reactor core design sets core inlet flow distribution criteria for mechanical and power capability purposes. The flow skirt is added to the vessel to enable the core inlet flow distribution to meet the fuel design requirements. The core inlet flow distribution assumptions which form the basis of the original fuel thermal hydraulic calculations remain valid.

The evaluation or analysis results are presented in the following sequence of DCD Section 15 event categories:

(1) Increase in heat removal from the primary system
(2) Decrease in heat removal by the secondary system
(3) Decrease in reactor coolant system flow
(4) Reactivity and power distribution anomalies
(5) Increase in reactor coolant inventory
(6) Decrease in reactor coolant inventory
(7) Radioactive release from a subsystem or component
(8) Anticipated transients without scram (ATWS).

The analysis results for these events show that the applicable acceptance criteria for each event are met or that the existing analysis is bounding.

1. Increase in Heat Removal From the Primary System

The following events involving increased heat removal from the primary system are analyzed in Section 15.1 of the DCD:

- Feedwater system malfunctions that result in a decrease in feedwater temperature
- Feedwater system malfunctions that result in an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal (PRHR) heat exchanger

These events are insensitive to minor changes in reactor vessel fluid volumes due to the reactor internals changes. To conservatively maximize plant cooldown, these events do not model the energy stored as metal heat. Therefore, the changes in vessel metal masses due to the internals changes will have no impact on the results of these analyses.

As previously discussed, the overall vessel pressure drop has not changed enough to impact the design RCS flow rates used in safety analyses. The pressure drop distribution through the vessel does differ somewhat. Of these cooldown events, the response to inadvertent opening of a SG relief valve and the steam system piping failure use the Core Makeup Tanks (CMTs) to borate the RCS and mitigate the transients. The CMTs inject into the reactor vessel downcomer. CMT flow is a function of the pressure differences between the CMT injection and the CMT balance line connection points to the RCS. The reactor vessel downcomer is between these two connection points. Therefore the steam line break event was analyzed to illustrate possible effects of changes in the vessel downcomer pressure drop on CMT flow and the steam line break transient.

A full double ended steam line rupture analysis from hot zero power conditions was performed for the updated AP1000 configuration. A comparison of the results from the updated analysis to those of the analysis for the original AP1000 configuration is shown in Figures 1-1 through 1-10. Table 1-1 provides the sequence of events for the original and the updated AP1000 configurations.

The transient results of the updated plant configuration during a steam line break are very similar to those of the original plant configuration. Core state parameters (inlet temperature, pressure, flow, and boron concentration) remain essentially the same. The return to criticality occurs at essentially the same time in the updated AP1000 configuration. Following the return to criticality, the core power is slightly less for the updated AP1000 (see Figures 1-1 and 1- 2). Based on these results, the DNB ratio will continue to meet the acceptance criteria with the revised AP1000 configuration.

Additionally, Section 15.1.5 of the DCD contains the radiological consequences for the steam system piping failure. The steam releases for the steam line break event are not sensitive to local conditions within the core or the changes in volumes, metal mass, and surface area that result from the design changes. The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes. The radiological consequences of a steam line break remain unchanged.

2. Decrease in Heat Removal by the Secondary System

The following events involving decreased heat removal by the secondary system are analyzed in Section 15.2 of the DCD:

- Loss of steam load transients (loss of electrical load, turbine trip, inadvertent closure of main steam isolation valves, loss of condenser vacuum, and other events resulting in turbine trip)
- Loss of ac power to station auxiliaries
- Loss of normal feedwater
- Feedwater system pipe break

The loss of ac power, loss of normal feedwater, and feed line break events are longer term transients analyzed to show that core decay heat can be safely removed by the PRHR. Small changes in vessel fluid volumes, metal mass, or vessel pressure drop distribution due to the reactor internals change will have minimal impact on these longer term transients. This is confirmed by an analysis of pressurizer overfill caused by a malfunction of the chemical and volume control system. This analysis was performed with the reactor internals changes incorporated and is discussed in Section 5 of this RAI response.

A loss of steam load (turbine trip) analysis for the updated AP1000 configuration was performed to investigate the impact of the reactor internals change on the "loss of heat sink" type of transients. Cases were considered for minimum and maximum reactivity feedback, with and without the pressurizer pressure control system operable, and with and without offsite power available. The most limiting cases with respect to maximizing reactor coolant pressure and minimizing core DNB ratio were analyzed.

The most limiting case with respect to maximizing reactor coolant pressure occurs without the pressurizer pressure control system available, with minimum reactivity feedback, and without offsite power available. Results for this case are shown in Figures 2-1 through 2-6. A sequence of events for this case is provided in Table 2-1 and is identified as Case C.2. The maximum reactor coolant pressure for the updated AP1000 configuration is 2642.77 psia [182.2 bar] as compared to 2693.65 psia [185.7 bar] in the original AP1000 plant configuration. The overpressurization results are less severe for the updated AP1000 plant.

The most limiting case with respect to minimizing the core DNB ratio is the case with the pressurizer pressure control system available, with minimum reactivity feedback, and without offsite power available. Results for the limiting DNB ratio case are shown in Figures 2-7 through 2-12. A sequence of events is provided in Table 2-1 and is identified as Case A.2. The minimum DNB ratio for the updated AP1000 is 1.58 as compared to a value of 1.57 for the original AP1000 plant configuration. The analyses show that the predicted DNBR is greater than the safety analysis limit of 1.5 at all times during the transient. Thus, the departure from nucleate boiling design basis continues to be met, and the existing turbine trip analysis remains applicable for the new AP1000 configuration.

3. Decrease in Reactor Coolant System Flow Rate

The following events involving a decrease in forced reactor coolant flow are analyzed in Section 15.3 of the DCD:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break

These events are mitigated by reactor trip within a few seconds of fault initiation. The duration of the analyses is approximately 10 seconds. Due to the rapid nature of these transients, energy stored in the reactor coolant system metal has an insignificant impact on the results and is not modeled. Therefore, the changes in vessel metal masses due to the internals changes will have no impact on the results of these analyses. Similarly these events are insensitive to minor changes in reactor vessel fluid volumes due to the reactor internals changes. The dominant system phenomena affecting these transients are the rate of reactor coolant flow decrease and the speed of detecting the fault and tripping the reactor.

As previously discussed, the overall vessel pressure drop has not changed enough to impact the design reactor coolant flow rates used in safety analyses. In a complete loss of flow, the rate of flow decrease is dominated by the inertia of the reactor coolant pumps. A change in the pressure drop distribution within the vessel will have little impact on the results of a complete loss of RCS flow.

In partial loss of flow transients, the active RCPs will ultimately force reverse flow through the faulted cold legs and reactor coolant pumps. A variation in the distribution of the vessel pressure drops could result in different flows in the faulted loops. Therefore, an analysis of partial loss of reactor coolant flow caused by the loss of electrical power to two reactor coolant pumps was performed to investigate this effect. Following the loss of power to the reactor coolant pumps, a reactor trip is actuated by the low hot leg flow trip function.

Results from the analysis for the updated AP1000 configuration are shown in Figures 3-1 through 3-6. A sequence of events for the analysis is summarized in Table 3-1. These updated results are compared with analysis results for the original AP1000 configuration.

With the updated AP1000 configuration, the results have improved. This is principally because there is a slight increase in the reactor coolant pump inertia from that used in the original AP1000 analyses (Reference 21) as a result of a more refined RCP design. The increased inertia results in a slightly slower reactor coolant pump coastdown (see Figure 3-1). With a slower reactor coolant pump coastdown, the time of reactor trip is delayed slightly. However, at the time of minimum DNBR, reactor coolant flow is slightly higher and minimum DNBR is higher. The DNBR safety analysis limit is 1.5. The original AP1000 configuration had a minimum DNBR of 1.64 for the partial loss of flow. The updated AP1000 configuration results in a minimum DNBR of 1.78.

The existing partial loss of reactor coolant flow analysis is bounding for the current AP1000 configuration and the existing analysis results for locked rotor, shaft break, and complete loss of flow events remain applicable.

Additionally, Section 15.3.3 of the DCD contains the radiological consequences for the reactor coolant pump shaft seizure (locked rotor) event. The steam releases for the locked rotor event are not sensitive to local conditions within the core or the changes in volumes, metal mass, and surface area that result from the design changes. The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes. The radiological consequences of a locked rotor are unchanged.

4. Reactivity and Power Distribution

The following reactivity and power distribution anomalies are analyzed in Section 15.4 of the DCD:

- Uncontrolled rod cluster control assembly bank (RCCA) withdrawal from a subcritical or low-power startup condition
- Uncontrolled rod cluster control assembly bank withdrawal at power
- Rod cluster control assembly misalignment (system malfunction or operator error)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant
- Inadvertent loading and operation of a fuel assembly in an improper position
- Spectrum of rod cluster control assembly ejection accidents

Rod ejection and uncontrolled RCCA bank withdrawal from subcritical or low power condition transients are analyzed using the TWINKLE code (Reference 6) and the FACTRAN code (Reference 7). The TWINKLE code is a multidimensional spatial neutron kinetics code which models only the reactor core. The FACTRAN code simulates the transient fuel rod conditions in the average and hot channels or at the hot spot. The hydraulic conditions are input as boundary conditions to these codes. Reactor vessel internals changes have no impact on the input used in the codes and the original AP1000 DCD analyses remain applicable.

The analysis for the inadvertent loading and operation of a fuel assembly in an improper position is completed using nuclear design models which simulate only the core. Reactor vessel internals changes have no impact on the input used in the methods and the original AP1000 DCD analyses remain applicable.

Uncontrolled RCCA bank withdrawal at power and RCCA misalignment transients such as dropped rods are analyzed using the LOFTRAN system analysis code. The impact of the reactor vessel internals changes on the input used in these analyses is small, and there will be minimal effect on the calculated DNB results of these transients.

The magnitude of boron dilution transients is directly proportional to the volume of active reactor coolant system fluid being diluted. The changes to the reactor vessel internals impact the fluid volumes within the reactor vessel. Therefore a boron dilution analysis of the updated AP1000 plant configuration was performed.

The impact that the flow skirt, neutron panels, and the change in vessel diameter have on the boron dilution event can be summarized by their effect on the active mixing volume. The time from initiation of the dilution to a loss of shutdown margin (that is, a return to critical) is directly proportional to the active mixing volume. An increase in active mixing volume equates to an increase in the time to a loss of shutdown margin. Conversely, a decrease in active mixing volume equates to a decrease in the time to a loss of shutdown margin.

Table 4-1 gives the volumes used in the original AP1000 boron dilution analysis and those corresponding to the updated AP1000 configuration. The changes to the reactor vessel internals result in an overall increase in the fluid volume of the reactor vessel. For example, in Mode 3 the active mixing volume increases by 4.6% and will result in a 4.6% increase in the time to a loss of shutdown margin. The results of boron dilution analyses with the updated AP1000 configuration using the reactor vessel internals modifications are less severe than those given in the original DCD analyses. It should be noted that Table 4-1 identifies increases in the active system mixing volumes greater than +100% for Modes 4 and 5. This change is due to a Technical Specification change and is not because of the reactor internals change.

Revision 16 of the AP1000 DCD was submitted to the NRC in Reference 11. It included a change to LCO and Applicability requirements of Technical Specification Section 3.4.8 on minimum RCS flow:

- The technical specifications in Revision 15 of the DCD required at least one reactor coolant pump (RCP) be operating whenever the reactor trip breakers are open in Modes 3, 4, and 5.

- The updated technical specifications in Revision 16 of the DCD now require at least one reactor coolant pump in operation in Modes 3, 4 and 5 whenever the reactor trip breakers are open and with unborated water sources not isolated from the RCS.

This Technical Specification change precludes boron dilutions from occurring in Modes 3, 4 and 5 when no reactor coolant pumps are running. A boron dilution can only occur in Modes 3, 4 and 5 when at least one RCP is running. When a reactor coolant pump is running, the active system mixing volume increases substantially because the active loop and steam generator volumes are now included.

Additionally, Section 15.4.8 of the DCD discusses the radiological consequences of the rod ejection event. The steam releases for the rod ejection event are not sensitive to local conditions within the core or the changes in volumes, metal mass, and surface area that result from the design changes. The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes. The radiological consequences of a rod ejection are unchanged.

5. Increase in Reactor Coolant Inventory

The following transients that increase reactor coolant inventory are analyzed in Section 15.5 of the DCD:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunction that increases reactor coolant inventory (CVS)

These events are similar in observed phenomena and similarly mitigated. Both are long term events that demonstrate that the PRHR can control the swell of the RCS fluid due to decay heat and the addition of fluid to the RCS. To examine the impact of the fluid volume and metal mass input differences due to reactor internals changes, an updated AP1000 analysis for the CVS malfunction that increases reactor coolant inventory was performed.

A comparison of the results from the updated analysis to those from the analysis for the original AP1000 configuration is shown in Figures 5-1 through 5-6. Table 5-1 provides the sequence of events for both the original and updated AP1000 configuration.

The transient results of the updated plant configuration during a CVS malfunction are similar to those for the original plant configuration. The "S" signal is obtained on a low T_{cold} signal about 15 seconds earlier for the updated AP1000 configuration. As a result, the subsequent reactor trip and the time that the core makeup tanks and passive residual heat removal heat exchanger are aligned occur about 15 seconds earlier. Thirty minutes after the reactor trip, the pressurizer water volume is 14 cubic feet greater for the updated AP1000 configuration than for the original AP1000 plant configuration. The peak pressurizer water volume occurs about 42 seconds later for the updated AP1000 plant configuration. From Figure 5-3 it can be seen that the trend in peak pressurizer volume for the updated AP1000 analysis closely resembles that seen for the original AP1000 analysis. Based on these results, 1) the CVS malfunction analysis is not significantly impacted by the flow skirt and neutron panels design changes (and the pressurizer design change, which was analyzed for a prior NRC Request for Additional Information), and 2) the case presented in the DCD continues to bound all cases that model explicit operator action 30 minutes after reactor trip.

6. Decrease in Reactor Coolant Inventory

The response to Item 6 is divided into six separate sections:

- 6.1 Inadvertent opening of a pressurizer safety valve or operation of the ADS
- 6.2 Failure of small lines carrying primary coolant outside containment
- 6.3 Steam generator tube ruputure
- 6.4 LOCA radiological consequences
- 6.5 Large-break loss-of-coolant accident (LOCA) analysis
- 6.6 Small-break LOCA analyses
- 6.7 Post-LOCA long-term cooling

6.1 Inadvertent Opening of a Pressurizer Safety Valve or Operation of the ADS

The analyses for the inadvertent opening of a pressurizer safety valve or operation of the ADS examine the short term (less than approximately one minute) depressurization effects from full power conditions to demonstrate that the protection system will trip the reactor prior to exceeding any core thermal design limits. The slight increase in reactor vessel fluid volumes will retard the reactor coolant depressurization rate. As previously discussed, the overall pressure drop has not changed enough to impact the design reactor coolant flow rates used in safety analyses. The change in the pressure drop distribution through the vessel will have minimal impact on the reactor coolant depressurization rate. To conservatively maximize the plant depressurization rate, these events do not model the energy stored in the reactor vessel metal. Therefore, changes in vessel metal masses due to the internals changes have no impact on the results of these analyses.

6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

Section 15.6.2 of the DCD discusses the radiological consequences of the failure of a small line break outside containment. The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes.

6.3 Steam Generator Tube Rupture

Section 15.6.3 of the DCD contains the steam generator tube rupture analysis for the AP1000 plant design. Neither the addition of the flow skirt or the addition of the four neutron panels will impact the results of the steam generator tube rupture margin to overfill or thermal hydraulic input to dose analyses. The steam generator tube rupture is not sensitive to local conditions within the core or the changes in volumes, metal mass, and surface area that result from the design changes. The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes. The steam generator tube rupture analyses remain unchanged by the addition of the flow skirt and neutron panels.

6.4 LOCA Radiological Consequences

Section 15.6.5.3 of the DCD discusses the radiological consequences of the Loss-of-Coolant Accidents. The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes.

6.5 Large-Break LOCA Analysis

Figures 6-1 through 6-3 show the changes as they apply in Rev. 16 of the DCD (Reference 3). Figure 6-1 shows the flow skirt installed in the vessel. A weld attaches the skirt along its lower edge to the inside vessel wall. The flow area between the downcomer and the lower plenum is reduced to the area through the holes of the flow skirt and in the 2.0 in. gap (1.5 in. when at full power operating conditions) between the lower core support plate and the flow skirt upper edge.

Figure 6-2 provides a full-vessel view of the changes, while Figure 6-3 shows the circumferential arrangement. The two Direct Vessel Injection (DVI) ports, two hot leg (outlet) nozzles, four neutron panels, and four radial keys are all placed at cardinal angles of 0°, 90°, 180°, and 270°. The four cold leg (inlet) nozzles are located at 45°, 135°, 225°, and 315°.

WCAP-15644 (Reference 14) provides documentation of the WCOBRA/TRAC code applicability to AP1000. It draws heavily from the AP600 code applicability report (WCAP-14171, Rev.2 (Reference 15)) for validation of modeling features such as the Direct Vessel Injection (DVI). Validation of flooding phenomena in the downcomers was supported by the Upper Plenum Test Facility (UPTF) tests (Reference 17), and WCOBRA/TRAC was shown to be conservative in its prediction of ECCS bypass and refill timing (Reference 15). In this section, an investigation is shown to determine the impact on the large break LOCA analysis (DCD, Rev. 15 (Reference 13), Section 15.6.5.4A) from the changes of Reference 2.

The peak cladding temperature (PCT) for the AP1000 large break LOCA reference transient is reproduced from the DCD, Rev. 15 Reference 13 in Figure 6-4. Table 6-1 shows the sequence of events following the break. The event timing is supported by Figures 6-5 through 6-10.

Immediately following the break, rapid vessel depressurization occurs during blowdown (Figure 6-10) and flow out of the broken cold decreases with time, approaching zero flow at 47 seconds (Figure 6-6). Vapor in the core, a result of flashing, flows downward into the lower plenum and toward the break through the downcomers (Figure 6-7). Liquid at the lower vessel head evaporates and exits into the lower plenum, but the flowrate is negligible compared to that from the core (Figure 6-8). After 15 seconds, the accumulator flow begins, each increasing to approximately 800 lbm/s before 30 seconds (Figure 6-5). The liquid level reaches a minimum in the lower plenum at 25 seconds, and evidence of liquid penetration from the downcomer to the lower plenum exists between 25 and 47 seconds as the inventory in the lower plenum begins to increase (Figure 6-9). At 47 seconds, the ECCS bypass ends with the end of core vapor flow through the lower plenum to the break, and the lower plenum then rapidly refills. By approximately 55 seconds after the break, the liquid level reaches the bottom of the core and reflood begins.

The Phenomena Identification and Ranking Table (PIRT) hierarchy from Reference 18 identifies the following phenomena for blowdown and refill:

Downcomer	Lower Plenum
Entrain / De-entrain	Sweep out
3-D	Hot Wall
CCF, Slug, nonequil. flow	Multi-D flow
2-Φ convection	
Saturated nucleate boiling	
Flashing (blowdown only)	

The neutron panels are investigated for their potential impact on counter-current flow (CCF) limitation due to changes in superficial vapor velocity (area reduction) and on saturated nucleate boiling on the downcomer wall (downcomer boiling). The change in superficial velocity is expected to have negligible local impact on entrainment / de-entrainment, and all other phenomena are not significantly affected by the neutron panels.

The flow skirt has little potential impact on downcomer entrainment phenomena (bulk superficial velocities in the downcomer will not change, and only local changes in superficial vapor velocities near the flow skirt are expected), slug flow, 2-phase convection, or flashing. The following phenomena are therefore identified and investigated in the following sections for both new components:

- Downcomer / Lower Plenum Boiling
- Local Pressure Losses during Sweep out

- Blowdown De-entrainment (due to impacting flow skirt)
- 3-D Flow and CCF in the downcomer and lower plenum

6.5.1 Assessment of Downcomer Boiling

The increased metal mass from the neutron panels presents a risk for increasing the severity of downcomer boiling. However, this is typically not of concern to the large break analysis of plants without containment ice condensers due to the high pressures and consequent high saturation temperatures. Figures 6-11 and 6-12 show that the only period of time that liquid temperatures approach saturation in the downcomer are between 60 and 70 seconds after the break, during the initial stages of reflood. The mid-core elevation is still uncovered (Figure 6-14), and the low-core elevation is only beginning to cover (Figure 6-15) during this time. As such, the liquid temperature approaching the saturation temperature is not an indication of downcomer boiling that may cause level-swell and inhibit the rate of reflood. It instead is due to liquid backflow following the initial surge of fluid into the core and subsequent pressurization from vapor production. This is also evident from the lower plenum liquid level (Figure 6-9); vapor generation from the initial reflooding liquid results in slight pressurization in the core (a slight increase occurs at 60 seconds in Figure 6-10), introducing a series of manometer oscillations as the core refloods. Later in reflood, no evidence of downcomer boiling is seen, and significant margin remains between the liquid and saturation temperatures.

Total additional mass as a result of the neutron panels is 30,400 lbs (13791 kg), or 3.5% of the total reactor vessel and internals weight. This additional mass is not sufficient to introduce any significant downcomer boiling.

Figure 6-13 shows that liquid temperature in the lower plenum remains well below the saturation temperature after the beginning of reflood, similar to conditions in the downcomer. Additionally, drawing APP-MI01-V2-340 R0 lists a mass of 2083 lbm (945 kg) for the flow skirt, in contrast to the lower core support plate, APP-MI01-V6-352 R0, which is 46330 lbm (21015 kg). Further, neglecting any blowdown cooling (assuming an initial temperature of 580°F), the flow skirt will have approximately 83000 Btu of stored heat available for liquid evaporation. The potential mass of liquid evaporated is then approximately 85 lbm assuming no liquid subcooling, no vapor superheat, and a heat of vaporization near atmospheric pressure. Since this mass is negligible relative to the lower plenum liquid inventory (see Figure 6-26), the rate of reflood as calculated in the DCD, Rev. 15 (Reference 13) will be maintained as the flow skirt does not introduce the potential for any significant boiling of liquid injected from the DVI.

6.5.2 Assessment of Local Pressure Losses

From the DCD, Rev. 15 (Reference 13), a double-ended break resulting in two 22-inch diameter pipe ends, or 760 in^2 total (380 in^2 each), is most limiting. The flow area in the downcomer is approximately 4180 in^2 at the radial keys and 4950 in^2 at the neutron panels, and the area through the flow skirt and the gap between it and the lower core plate is approximately 2950 in^2 during hot operation. None of these areas is sufficiently small to create local choke points between the core and the vessel-side break, so the behavior of (a prolonging of) blowdown will not change.

Further, the steady state pressure drop increase due to the flow skirt and neutron panel addition (evident in the Lower Plenum and Barrel-vessel annulus pressure drop) is mostly offset by the reduction in pressure drop through the inlet nozzle. The Former pressure drop between the inlet nozzle and the upper plenum at 315,000 gpm was 56.31 psi (10.72 inlet nozzle, 0.21 barrel-vessel annulus, 5.46 lower plenum, 39.92 core), compared to 57.05 psi after the changes (5.00 inlet nozzle, 0.41 barrel-vessel annulus, 11.81 lower plenum, 39.83 core). This change in steady-state pressure drop is negligible to the large break LOCA behavior, so changes in blowdown timing will not occur as a result of the flow skirt addition since the pressure immediately upstream of the break, which determines the critical flow rate, will not significantly change.

6.5.3 Assessment of Blowdown De-Entrainment

During blowdown, the radial outflow of vapor and entrained liquid will encounter increased opportunity for de-entrainment due to the local flow area reduction and droplet impact with the frontal area of the flow skirt. The total flow area through the flow skirt holes (approximately 35% porous) and the space between the skirt and lower core plate (see Figure 6-1) is approximately 2950 in^2. Without the skirt, the flow area was approximately 7250 in^2.

Methodology for calculating the de-entrainment due to an area contraction in vertical flow is described in the WCOBRA/TRAC Code Qualification Document (CQD) (Reference 16), Section 4-6-8. The situation is typically encountered during reflood, where vertically-flowing droplets de-entrain after striking solid portions of the upper tie plate. Experimental validation includes tests from the CCTF, SCTF, UPTF, and LOFT facilities (Reference 16). Section 4-6-7 of Reference 16 documents the crossflow de-entrainment model, validated to experiments relevant to the upper plenum structures. Otherwise, significant de-entrainment is not observed in horizontal flow other than within the core, which is addressed in Section 4-6-5 of Reference 16 concerning the spacer grid droplet breakup model.

For the case of the flow skirt, de-entrainment during blowdown may result in an increase in liquid inventory in the lower plenum at the beginning of refill. Approximately equal massflow is lost in the form of entrained liquid and vapor (see Figures 6-20, 6-21 and 6-22). Based on the flow skirt porosity and the phenomena observed for vertical flow through an area contraction, an upper bound of 65% of the entrained liquid may be retained in the lower plenum, although some may be re-entrained and eventually swept out. The net result is a shorter refill time and earlier onset of reflood. Although no credit for this phenomenon is taken, the effects are beneficial, and so the analysis of the DCD, Rev. 15 (Reference 13) is bounding.

6.5.4 Assessment of 3-D and Countercurrent Flow

The addition of the neutron panels decreases slightly the flow area in the downcomer below the DVI injection points and above the radial keys, resulting in a small increase in local vapor velocity. Due to the size of the neutron panels (each occupying approximately 30° of the downcomer circumference and reducing the downcomer flow width by a net of 1 in. after a 2 in. vessel ID increase), this will be localized both vertically and circumferentially. The effect will not cause any significant change to the ECCS bypass behavior observed during refill since the flow limiting is a result of local velocities at the injection plane (Reference 19) and not intermediate velocities at lower downcomer elevations.

As described by Okabe (Reference 20), the counter-current flow limiting (CCFL) that occurs in the downcomer has direct impact on the transient timing following a large break LOCA. Flow of vapor from the core causes the ECCS bypass that effectively determines the time to begin refill of the lower plenum. Glaeser (Reference 19) showed that "the predominant flow regime observed in both small scale and large scale countercurrent flow experiments is a heterogeneous distribution of gas upflow regions and liquid downflow regions." This behavior demonstrated in large scale tests results in earlier refill than what would result from a less heterogeneous, more uniform counter-current flowfield.

Figure 6-16 shows that circumferential flow rates of liquid and entrained liquid are predicted to persist throughout the ECCS bypass phase on the order of the injection flow rates (Figure 6-5) at the nozzle elevation. Fig. 7 of Glaeser (Reference 19) implies that this strong circumferential flowfield becomes weaker at lower elevations. This is clear as well from Figures 6-17, 6-18, and 6-19. The circumferential flow in the downcomer channel is one order of magnitude less at the mid-core elevation than at the nozzles throughout the transient, and two orders of magnitude less near the lower plenum.

The radial orientation of the flow skirt holes may slightly impede circumferential flow in the lower plenum during periods in which the flow has both a radial and circumferential component. A potential effect is the creation of more uniform radial outflow of vapor in the time when vapor from the core is exiting through the lower plenum (until 47 seconds, Figure 6-7). However, since Figure 6-19 shows that all phases have no significant circumferential component during blowdown, ECCS bypass, and refill, the presence of the flow skirt in the lower plenum will have no effect on the azimuthal flow at the upper elevations. The presence of the flow skirt therefore may increase the local superficial velocities in regions of radial outflow, but this does not affect the rate of liquid penetration in the downcomer that is limited by counter-current flow at the injection elevation.

With the total rate of liquid from DVI injections unaffected by the presence of the neutron panels and flow skirt, increased vapor velocity exiting through the skirt as a result of flow area reduction is investigated for the potential of localized CCFL that could aggravate liquid holdup in the downcomer. Three lower plenum regions are investigated to span the possible counter-current flow conditions; (1) the area below the DVI opposite the broken cold leg, (2) below the DVI adjacent to the broken cold leg, and (3) below the cold leg break.

6.5.4.1 Region 1: Below the DVI Opposite the Break

The WCOBRA/TRAC model documented in the DCD, Rev. 15 (Reference 13) utilizes 6 downcomer channels. Gap 7 is a radially-inward oriented gap flowing into the lower plenum from the region in the downcomer directly below the DVI opposite the postulated break. Flow across this gap represents flow that will be through the flow skirt. Figure 6-20 shows the vapor, entrained vapor, and liquid flowrates.

Co-current vapor and liquid outflow begins immediately following the break. The radial outward flow of vapor ceases at approximately 28 to 30 seconds after the break, with a simultaneous cessation of entrained liquid flow. Liquid penetration into the lower plenum becomes positive between 25 and 30 seconds, and the lower plenum liquid level begins to rise above its minimum (Figure 6-9). Note that the majority of liquid entering the lower plenum in this region flows out (see *Region 3* below), limiting the rate at which the lower plenum fills.

The co-current flowfield is expected during blowdown due to the high pressure remaining in the vessel driving flow to the break (Figure 6-10). Shortly after the start of accumulator flow (15 seconds) only liquid flow exists in the region below the DVI opposite the break, consistent with the observations of Glaeser (Reference 19) in the UPTF tests. As a result of the blowdown timing and the nature of ECCS bypass at the nozzle elevations, no significant counter-current flow persists in this region.

6.5.4.2 Region 2: Below the DVI Adjacent to the Break

Gap 10 is a radially-inward oriented gap flowing into the lower plenum from the region in the downcomer directly below the DVI that is adjacent to the broken cold leg. Flow across this gap represents flow that will be through the flow skirt. Figure 6-21 shows the vapor, entrained vapor, and liquid flowrates.

Co-current outward flow persists below the break until approximately 40 seconds after the break occurs when single phase liquid flows into the lower plenum. Figure 6-23 shows the vertical flow in the downcomer channel immediately below the DVI injection. Counter-current flow limiting in the upper downcomer region is clear between 15 (immediately after the start of accumulator injection) and 25 seconds, with upward (positive) vapor and entrained liquid flow and downward (negative) liquid flow. Beginning at 25 seconds, downward liquid flow increases to approximately the accumulator flow rate, and co-current flow persists for the remaining time due to the precipitous reduction in flow to the break (Figure 6-6) and the weakening in the circumferential bypass flowfield (Figure 6-16).

From Figures 6-21 and 6-23, the counter-current flow expected in the region below the DVI near the broken cold leg occurs at the upper downcomer elevations, limiting the rate of liquid penetration to the lower downcomer regions. Only co-current flow is observed in the lower plenum, since the end of complete bypass (approximately 40 seconds) results from significant reduction in core vapor flow (Figure 6-7), allowing single phase liquid flow into the lower plenum (Figure 6-21).

6.5.4.3 Region 3: Below the Broken Cold Leg

Gap 11 is a radially-inward oriented gap flowing into the lower plenum from the region in the downcomer directly below the broken cold leg. Flow across this gap represents flow that will be through the flow skirt. Figure 6-22 shows the vapor, entrained vapor, and liquid flowrates.

Co-current outward flow begins immediately following the break and continues until approximately 50 seconds (also corresponding with the end of break flow at 47 seconds). The mass flow rate is approximately 50% higher than the flow out of the lower plenum below either DVI during blowdown. Single phase liquid flow out of the plenum persists after the end of blowdown until approximately 65-70 seconds; this is attributable to manometer oscillations as the liquid level reaches the core (Figure 6-9), since no vertical flow of liquid exists at the upper elevations during this time (Figure 6-24). Once liquid reaches into the core, steam is produced, the core pressurizes, and the liquid level within the core barrel drops while liquid is pushed back up the downcomer.

The highest vertical flowrates observed in the downcomer at the nozzle elevations occur below the broken cold leg (Figure 6-24). In contrast to the vertical column of liquid in the absence of vapor flow below the DVI opposite the break (Figure 6-25), the lower plenum in this region is essentially void of liquid until the very end of ECCS bypass when vapor flow from the core ceases.

The investigation of the three regions shows that CCFL is predominant in the upper region of the downcomer, while the lower plenum region is not limiting as the vapor exiting the bottom of the core flows preferentially toward the lower pressure at the break as opposed to uniformly outward.

In addition to localized CCFL, there exists a risk for liquid holdup circumferentially along the flow skirt below the first row of holes (see Figure 6-1) as well as increased liquid holdup in the downcomer as a result of injected liquid spreading around the radial keys and flow skirt. The total rate of liquid entering the vessel is dominated by the downcomer CCFL at the injection location as discussed above; however the flow skirt may delay the liquid penetration into the lower plenum.

The onset of liquid penetration below the DVI opposite the break is shown to occur between 25 and 30 seconds after the break. The rate of lower plenum filling increases around 40 seconds after the break (Figure 6-9) when vapor flow from the core is two orders of magnitude below its peak and liquid from the DVI adjacent to the break begins to overcome the ECCS bypass. At 47 seconds, vapor flow exiting the core ceases and the remaining time to refill is approximately 8 seconds (Table 6-1).

A bounding assumption to predict the effect of flow skirt impedance and/or CCFL on liquid penetration from the downcomer into the lower plenum is to disallow penetration until the end of vapor flow out of the bottom of the core. The liquid penetrating into the downcomer between 25 and 47 seconds would then not contribute to the lower plenum liquid level. The liquid stored in the downcomer during this period will then flow by gravity into the lower plenum once the vapor flow is negligible, similar to a monometric oscillation.

The DCD (Reference 13) case shows the start of liquid penetration from the DVI opposite the break as an increase in liquid mass at 25 seconds (Figure 6-26). This is followed by the start of penetration from the DVI adjacent to the break, apparent as an increase in slope starting around 40 seconds. Around 50 seconds, after vapor flow exiting the core stops, a manometer effect forces liquid rapidly into the lower plenum until the levels equilibrate (around 52-53 seconds). This is evidence that liquid holdup occurs in the downcomer even without the flow skirt addition. Total accumulator flow, the integral of which is shown in Figure 6-26, then dictates the fill rate. Throughout the bypass phase, very little accumulator flow contributes to the lower plenum liquid inventory as most is either bypassed or held up in the downcomer.

The bounding case of complete liquid holdup is shown in Figure 6-26. The liquid inventory is maintained at its minimum from 25 seconds until the core vapor exit flow terminates at 50 seconds. A rate of fill due to manometer oscillation is assumed equal to the case of incomplete holdup (this is a bounding estimate since the increased liquid mass in the downcomer may contribute to a higher oscillation frequency, although more circumferential pooling is expected as opposed to a tall vertical column) until the total mass reaches the same level as at the end of manometer filling for the base case. This results in a 1.4 second delay in the time at which the refill rate is dictated solely by the accumulator flow, and therefore indicates a 1.4 second delay in the end of refill / beginning of reflood.

A more severe estimate accounts for the possibility of liquid sweep out as a result of holdup in the downcomer. Were the held up liquid lost from the vessel inventory completely, the lower plenum would only fill with the mass observed during the 50 to 52 second manometer filling from the base case. (It is assumed that this liquid already held up in the downcomer would remain so in the presence of the flow skirt.) This case is shown in Figure 6-27, where the 2 second manometer filling is followed by filling at the approximate linear flowrate observed after manometer behavior. To completely fill the lower plenum, an additional 9.4 seconds is required over the DCD (Reference 13) case.

6.5.5 Conclusions

The changes as outlined in Reference 2 have been investigated for potential effects concerning the large break LOCA analysis of the DCD, Rev. 15 (Reference 13), Chapter 15. All phenomena relevant to the regions of interest (lower plenum and downcomer) were shown have insignificant changes as a result of the flow skirt and neutron panels except for potential delay of liquid penetration into the lower plenum during the refill phase.

An assumption that all liquid penetrating into the lower plenum during the bypass phase is instead held up in the downcomer region (in addition to that already held up in the analysis of Reference 13 due to downcomer CCFL) results in a delay in the onset of reflood of 1.4 seconds. A highly conservative and bounding assumption that the held up liquid is entirely swept out of the vessel results in a delay of 9.4 seconds.

From Section 3.2 of Reference 15, WCOBRA/TRAC is conservative in predicting ECCS bypass in that delays in timing relative to validation experiments (UPTF Test 6) are observed. The bounding delay calculated herein is on the order of magnitude of this conservative bias, and so the acceptance criteria concerning large break LOCA remain complied with.

6.6 Small-break LOCA Analyses

The Double-Ended Direct Vessel Injection (DEDVI) small-break loss of coolant accident (SBLOCA) analysis presented in Section 15.6.5.4B.3.5 of both Revisions of the DCD (References 13 and 3) and the 2-Inch Cold Leg Break analysis presented in Section 15.6.5.4B.3.4 of both documents were re-performed, taking into account the revised reactor internals and the previously assessed Pressurizer design changes. The DEDVI line break represents the limiting break in terms of available injection capacity because, one DVI line having failed, only one DVI line is available for RCS makeup. The 2-Inch Cold Leg Break demonstrates the response to a smaller break class.

6.6.1 DEDVI Line Break Analysis

Table 6-2 and Figures 6-28 through 6-36 present a comparison of the DEDVI line break cases. The results demonstrate the effect of the change in the reactor internals design (including the revised Pressurizer). For the DEDVI line break, the observed changes in response can be attributed primarily to the change in the Pressurizer rather than to the reactor internals changes. The largest observed changes are the decreases in both the Pressurizer Mixture Level (Figure 6-31) and Pressurizer Void Fraction (Figure 6-32). These subsequently reduce the Automatic Depressurization System (ADS) Stage 1-3 liquid discharge (Figure 6-33) and increase the ADS 1-3 vapor discharge (Figure 6-34). The overall effect is slightly larger Reactor Coolant System (RCS) inventory for a majority of the transient simulation period (Figure 6-36), with about the same predicted minimum RCS inventory. No significant changes in the RCS Pressure (Figures 6-28 and 6-29), Core Mixture Level (Figure 6-30), and DVI injection characteristics (Figure 6-35) are observed as a result of the design changes implemented.

6.6.2 Two Inch Cold Leg Break Analysis

Table 6-3 and Figures 6-37 through 6-46 present a comparison of the 2-Inch Cold Leg break simulations. The results again demonstrate the effect of the change in the reactor internals design (including the revised Pressurizer). Decreases in both the Pressurizer Mixture Level (Figure 6-40) and Pressurizer Void Fraction (Figure 6-41) were observed as a result of the Pressurizer change. For the 2-Inch Cold Leg Break, this subsequently reduces both the Automatic Depressurization System (ADS) Stage 1-3 liquid discharge (Figure 6-42) and ADS 1-3 vapor discharge (Figure 6-43). The overall effect of these design changes is slight variations in the Reactor Coolant System (RCS) inventory over the transient simulation period (Figure 6-46) with about the same predicted minimum RCS inventory. No significant changes in the RCS Pressure (Figures 6-37 and 6-38), Core Mixture Level (Figure 6-39), and DVI injection characteristics (Figures 6-44 and 6-45) are observed as a result of the design change implementation.

6.6.3 Conclusions

As discussed in the previous subsections, the effects of the reactor internals design changes described in Reference 1 (including the previously assessed pressurizer design changes) on the AP1000 SBLOCA analyses presented in Chapter 15 of DCD Rev. 16 (Reference 3) are small. As a result, these SBLOCA analyses remain valid with respect to the changes.

6.7 Post-LOCA Long-Term Cooling

The AP1000 SBLOCA re-analyses performed in support of this RAI response (Section 6.6) consider the Reference 1 reactor internals design changes. The DEDVI line break is among the analyses re-performed with the revised reactor internals design. It represents the limiting break for long term cooling due to its early switchover to containment recirculation, and therefore is the loss of coolant accident long term cooling (LOCA LTC) case analyzed in References 13 and 3. The DEDVI line break SBLOCA re-analysis demonstrates that at its conclusion, at the time that continuous injection of water from the IRWST has been established, both the Reactor Coolant System (RCS) inventory and the injection flow rate from the IRWST are essentially identical to those in the DEDVI break transient simulation results presented in References 13 and 3. The reactor fluid conditions at the start of the LOCA LTC analysis as performed are therefore almost identical, including the condition that the downcomer and lower plenum regions are largely filled with liquid.

None of the Reference 1 reactor internals design changes has a significant impact on the AP1000 LOCA LTC analysis. In the AP1000, long-term core cooling is accomplished by introducing and maintaining liquid in the reactor vessel by gravity flow from the IRWST and/or containment sump in conjunction with the operation of ADS Stage 4 to vent steam generated in cooling the core. Low fluid velocities prevail within the reactor vessel during LOCA LTC, so the impact of a slight change in the reactor vessel hydraulic resistance (pressure drop) is negligible. Moreover, the energy initially stored in the downcomer / lower internals metal is largely removed prior to the LTC phase of the DEDVI break, so the impact of slight changes in the reactor vessel internals mass is unimportant.

Thus, the effect of the reactor internals design changes described in Reference 1 on the AP1000 LOCA LTC analysis presented in Chapter 15 of DCD Rev. 16 (Reference 3) is insignificant. As a result, that LOCA LTC analysis remains valid with respect to the changes.

7. Radioactive Release from a Subsytem or Component

The following radiological consequence is analyzed in Section 15.7 of the DCD:

- Fuel handling accident

The radiological consequence analysis does not specifically model the vessel geometry and as such is not impacted by the design changes.

8. Anticipated Transients Without SCRAM

A deterministic analysis of the loss of normal feedwater event without SCRAM was performed. The loss of normal feedwater ATWS case represents the most limiting "loss of heat sink" event. This transient was performed using assumptions compatible with analyses of Westinghouse plants in determining the ATWS rule (References 4 and 5).

The acceptance criterion assumed for the ATWS transient is that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in ASME Service Level C. A pressure of 3200 psig corresponds to the maximum allowable pressure for the weakest component in the reactor coolant system. Mitigation of the event is provided by activation of pressurizer and steam generator safety valves, actuation of the Passive Residual Heat Removal (PRHR) system, and actuation of the Core Makeup Tanks (CMT). The Diverse Actuation System (DAS) is assumed to operate to actuate the PRHR, and CMTs and trip the turbine. DAS also includes a diverse reactor trip function which de-energizes power to the RCCA motor-generator set. However, the diverse reactor trip function is not credited in the analysis presented here.

Analysis results are provided for the original AP1000 plant configuration in Revision 16 of the DCD (Reference 3). Results are also provided for the updated AP1000 plant configuration with the modified reactor vessel internals. Figures for both cases are provided in Figures 8-1 through 8-5. A sequence of events is provided for both cases in Table 8-1. The results show that the thermal hydraulic response trend remains unchanged. With the original AP1000 plant configuration, a maximum reactor coolant pressure of 2712.19 psia is reached. The updated AP1000 configuration results in a maximum reactor coolant pressure of 2756.67 psia, which is well below the acceptance criterion of 3200 psig.

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21. DCP/NRC1802, "AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-016, Revision 0," November 17, 2006

Table 1-1 - Sequence of Events for Full Double Ended Steam Line Break		
Event	Time (sec)	
	Original AP1000	Revised AP1000
Break initiated, startup feedwater started, PRHR started	0.0	0.0
Low steam line pressure setpoint reached	1.4	1.4
Reactor coolant pumps tripped on low steam line pressure signal	7.4	7.4
Main feedwater isolation valves and main steam line isolation valves close on low steam line pressure signal	13.4	13.4
Low cold leg temperature setpoint reached	18.1	17.7
Core Makeup Tanks (CMT) actuated on low steam line pressure signal	18.4	18.4
Criticality reached	28.0	28.0
Startup feedwater isolated on low cold leg temperature signal	30.1	29.7
Boron begins reaching the core	30.2	35.3
Pressurizer empties	58.2	52.6
Accumulators begin injecting	259.6	271.6

Table 2-1 Sequence of Events for Turbine Trip			
Accident Case	Event	Time (sec)	
		Original AP1000	Revised AP1000
Case C.2 Without pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0	3.0
	Reactor coolant pump underspeed trip point reached	3.472	3.526
	Rods begin to drop	4.239	4.293
	Maximum RCS pressure occurs	6.300 (2693.65 psia, 185.7 bar)	6.600 (2642.77 psia, 182.2 bar)
	Steam generator safety valve setpoint reached	14.000	11.700
Case A.2. With pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.468	3.522
	Rods begin to drop	4.235	4.289
	Minimum DNBR occurs	6.000 (1.57)	6.300 (1.58)
	Steam generator safety valve setpoint reached	18.700	15.300

Table 3-1 - Sequence of Events for a Partial Loss of Forced Reactor Coolant Flow (Loss of Two Pumps with Four Pumps Running)		
Event	Time (sec)	
	Original AP1000	Revised AP1000
Coastdown begins	0.0	0.0
Low flow reactor trip setpoint reached	1.61	1.93
Reactor Trip - RCCA insertion begins	3.06	3.38
Minimum DNBR occurs	4.70	4.90

Table 4-1 Boron Dilution			
Mode of Operation	Active System Mixing Volume, ft^3		Difference in Active Volume, %
	Updated AP1000	Original AP1000	
1	8463.9	8126.5	+4.2
2	8463.9	8126.5	+4.2
3	7636.2	7300.7	+4.6
4	7636.2	2805.0	> +100% *
5	7636.2	2402.0	> +100% *

* A change in Technical Specification Section 3.4.8 now requires at least one reactor coolant pump in operation in Modes 3, 4 and 5 whenever the reactor trip breakers are open and with unborated water sources not isolated from the RCS. This precludes boron dilutions from occurring in Modes 3, 4 and 5 when no reactor coolant pumps are running. When a reactor coolant pump is running, the active mixing volume increases substantially to include the active loop and steam generator.

Table 5-1 - Sequence of Events for a Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory		
Event	Time (sec)	
	Original AP1000	Revised AP1000
Transient started; spurious "CVS" starts injecting borated water	10.0	10.0
"S" signal; low T_{cold} setpoint reached	1088	1073
Reactor trip - RCCA insertion begins	1090	1075
Turbine trip begins	1090	1075
Loss of offsite power and reactor coolant pumps start to coast down	1093	1078
CVS, main steam and feed lines are isolated; CMT valves fully open	1100	1085
PRHR actuated on "S" signal (valve fully open)	1105	1090
Pressurizer safety valves open	1424	1236
Pressurizer water volume 30 minutes after reactor trip	1415 ft^3 (2900 sec)	1429 ft^3 (2876 sec)
High-2 Pressurizer level setpoint reached (CVS isolation)	3720	3650
Pressurizer safety valves close	15,088	14,816
Peak pressurizer water volume occurs	2140 ft^3 (15,262. sec)	2171 ft^3 (15,304. sec)
PRHR matches decay heat	14,720	15,200
CMTs stop recirculating	20,200	21,048

Table 6-1 Sequence of Events for Reference Transient DECLG Break*	
Event	Time (sec)
Break occurs with coincident loss of offsite power	0
Blowdown PCT occurs	12 (approx.)
Accumulator injection begins	15
End of blowdown	47
Liquid level reaches bottom of active fuel; reflood begins	55 (approx.)
Calculated PCT occurs	109.6
Core quench occurs	238

*See also Table 15.6.5-6 in of the DCD (Reference 3).

Table 6-2 Sequence of Events for Double-Ended Injection Line Break– 20 psia		
Event	Original AP1000 Time (seconds)	Updated AP1000 Time (seconds)
Break opens	0.0	0.0
Reactor trip signal	13.1	13.0
Steam turbine stop valves close	19.1	18.1
"S" signal	18.6	18.0
Main feed isolation valves begin to close	20.6	20.0
Reactor coolant pumps start to coast down	24.6	24.0
ADS Stage 1	182.5	180.1
ADS Stage 2	252.5	250.1
Intact accumulator injection starts	254	249.2
ADS Stage 3	372.5	370.1
ADS Stage 4	492.5	490.1
Intact accumulator empties	600.0	594.5
Intact loop IRWST injection starts*	1470	1695
Intact loop core makeup tank empties	2123	2100

*Continuous injection period

Table 6-3 Sequence of Events for 2-Inch Cold Leg Break in CLBL Line		
Event	Original AP1000 Time (seconds)	Updated AP1000 Time (seconds)
Break opens	0.0	0.0
Reactor trip signal	54.7	54.6
Steam turbine stop valves close	60.7	59.7
"S" signal	61.9	62.0
Main feed isolation valves begin to close	63.9	64.0
Reactor coolant pumps start to coast down	67.9	68.0
ADS Stage 1	1334.1	1329.8
Accumulator injection starts	1405	1394.3
ADS Stage 2	1404.1	1399.8
ADS Stage 3	1524.1	1519.8
Accumulator empties	1940.2	1921.3
ADS Stage 4	2418.6	2422
Core makeup tank empty	2895	2923
IRWST injection starts*	3280	3361

*Continuous injection period

Table 8-1 Sequence of Events for Loss of Normal Feedwater With Failure to SCRAM (ATWS)		
Event	Time (sec)	
	Original AP1000	Revised AP1000
All feedwater flow to steam generators is lost	0 to 4	0 to 4
Pressurizer safety valves open	50.0	45.0
Reactor coolant pumps tripped on DAS signal	66.1	62.7
Turbine is tripped on DAS signal	70.1	66.7
PRHR actuated on DAS signal	74.1	70.7
CMTs actuated on DAS signal	74.1	70.7
Pressurizer fills	92.0	88.0
Maximum reactor coolant pressure reached	132.0 (2712.19 psia, 187 bar)	124.0 (2756.67 psia, 190.07 bar)
Pressurizer safety valves close	197.0	197.0

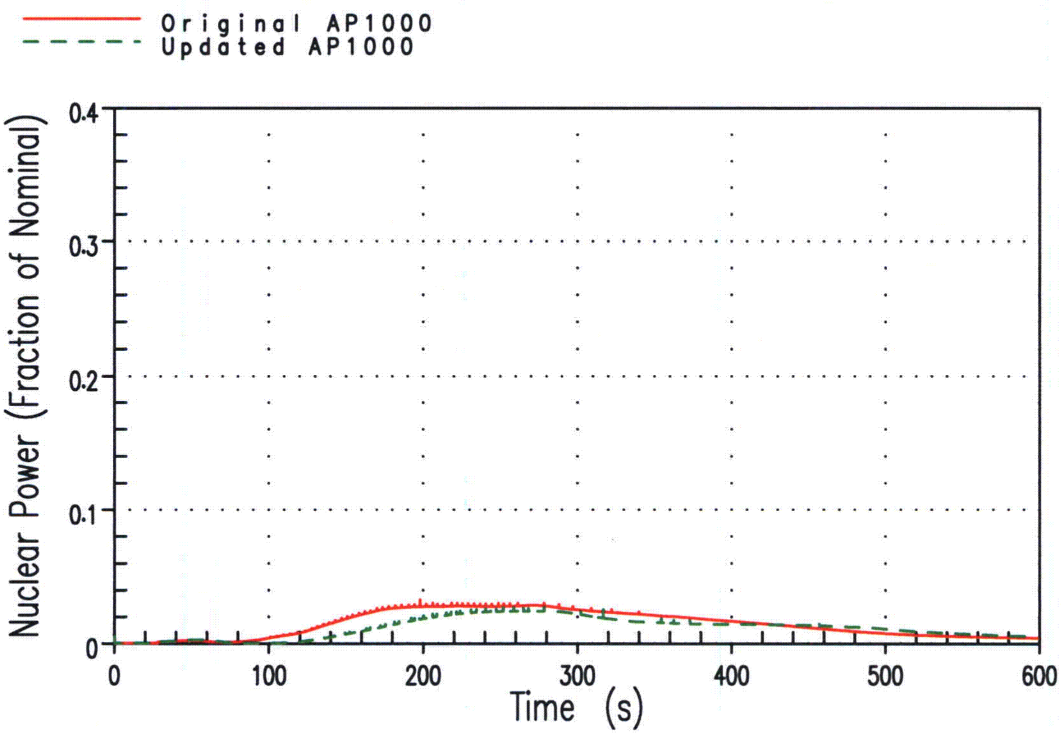


Figure 1-1
Steam Line Break

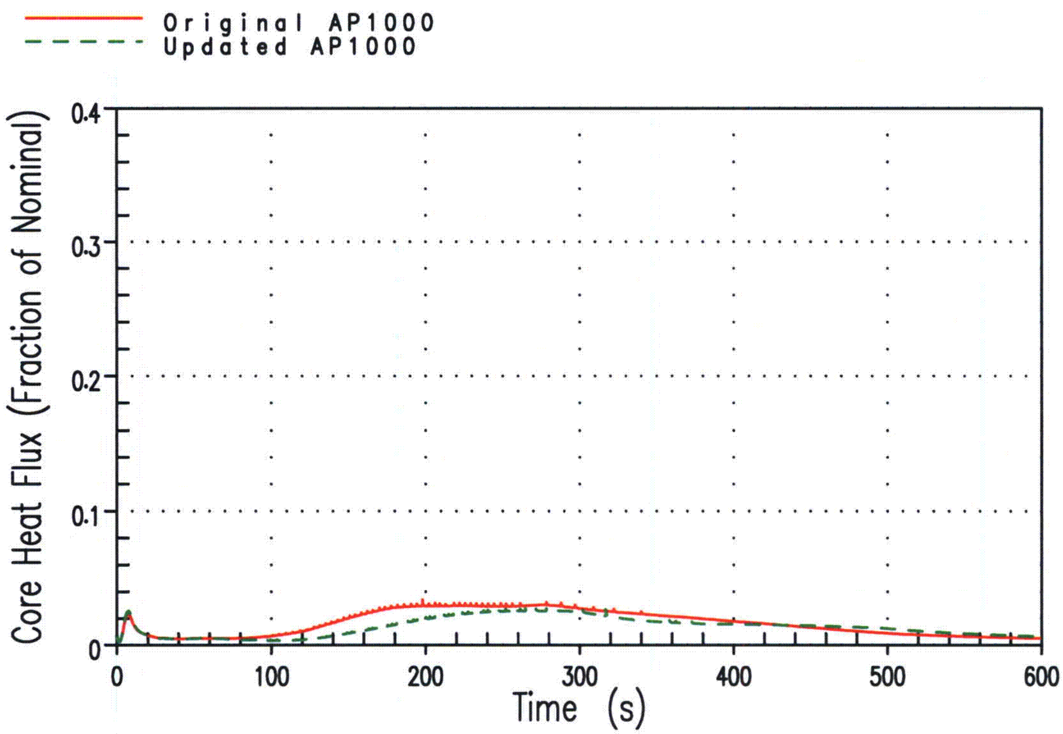


Figure 1-2
Steam Line Break

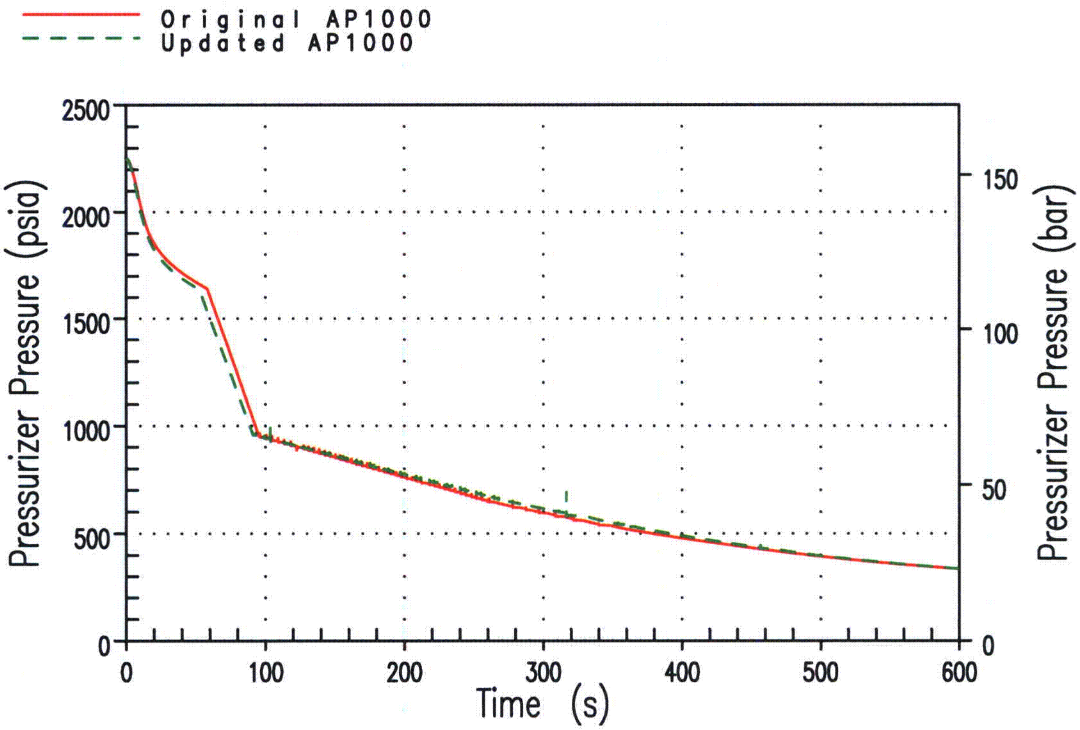


Figure 1-3
Steam Line Break

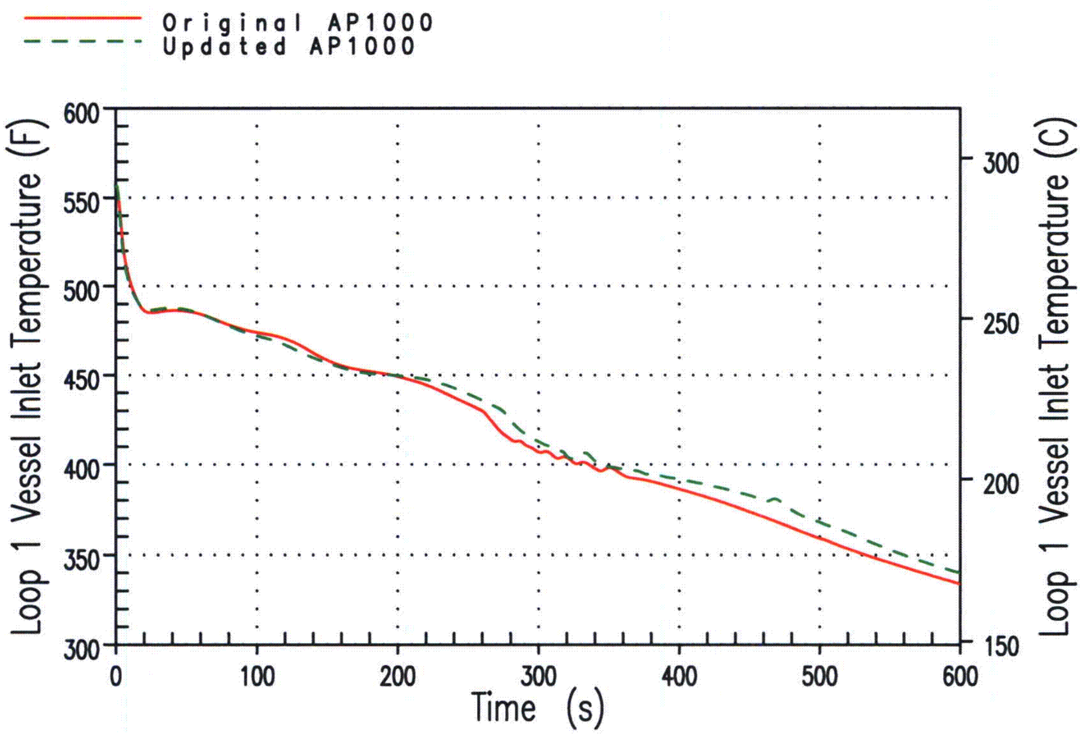


Figure 1-4
Steam Line Break

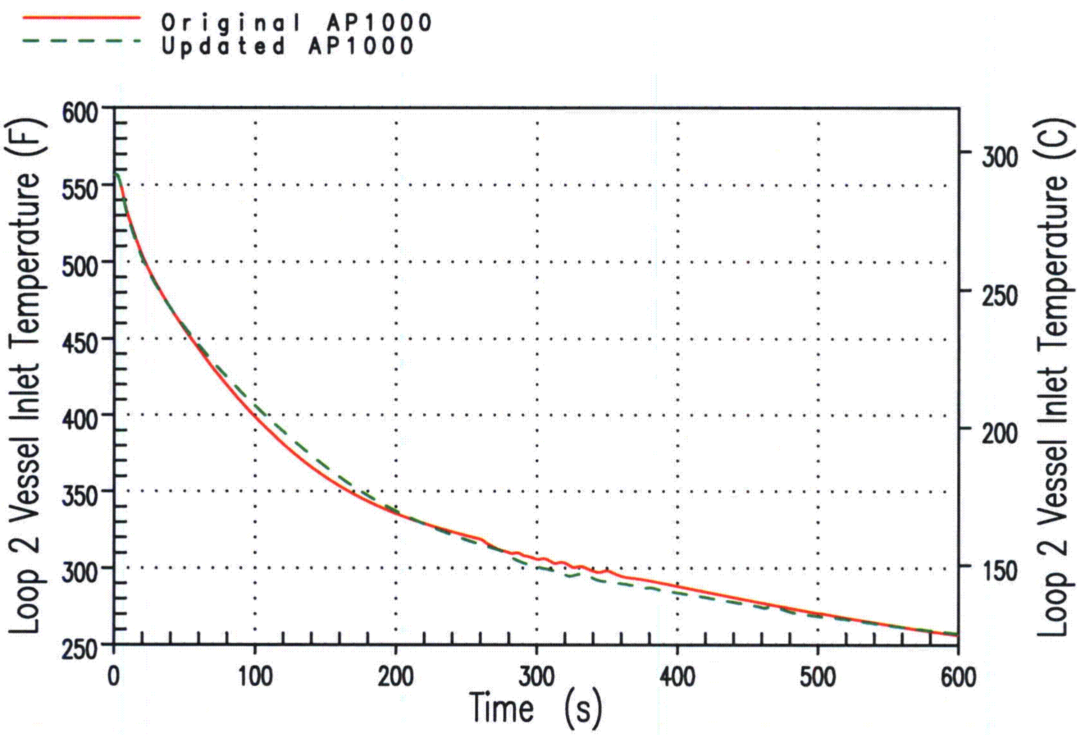


Figure 1-5
Steam Line Break

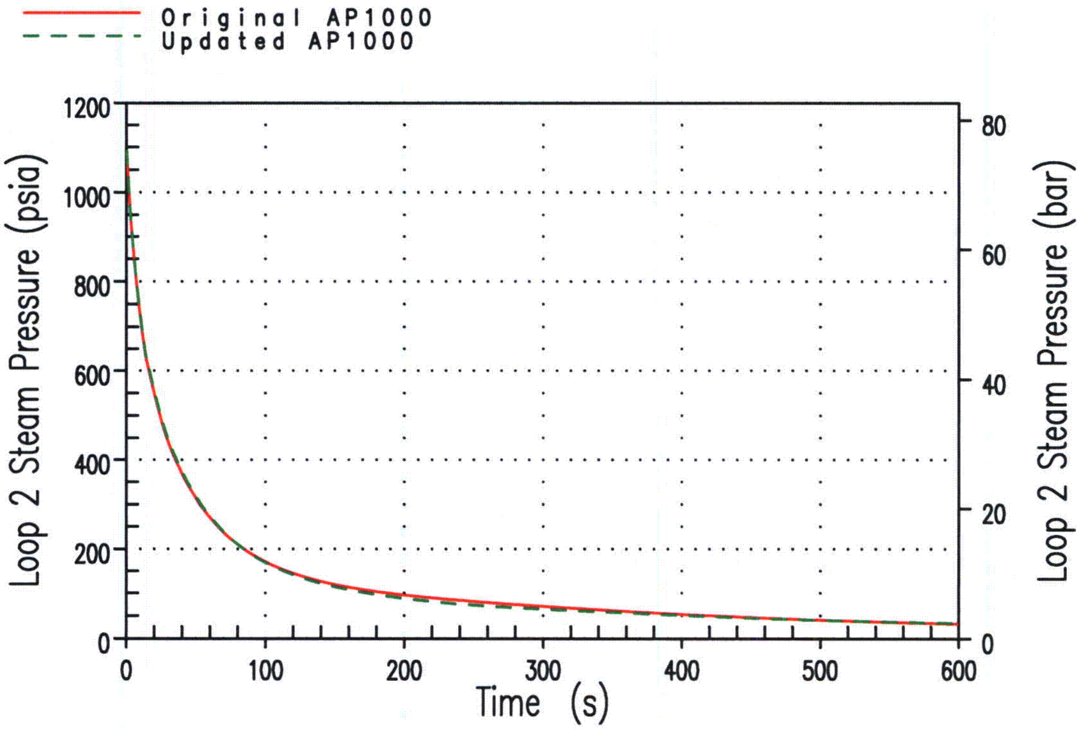


Figure 1-6
Steam Line Break

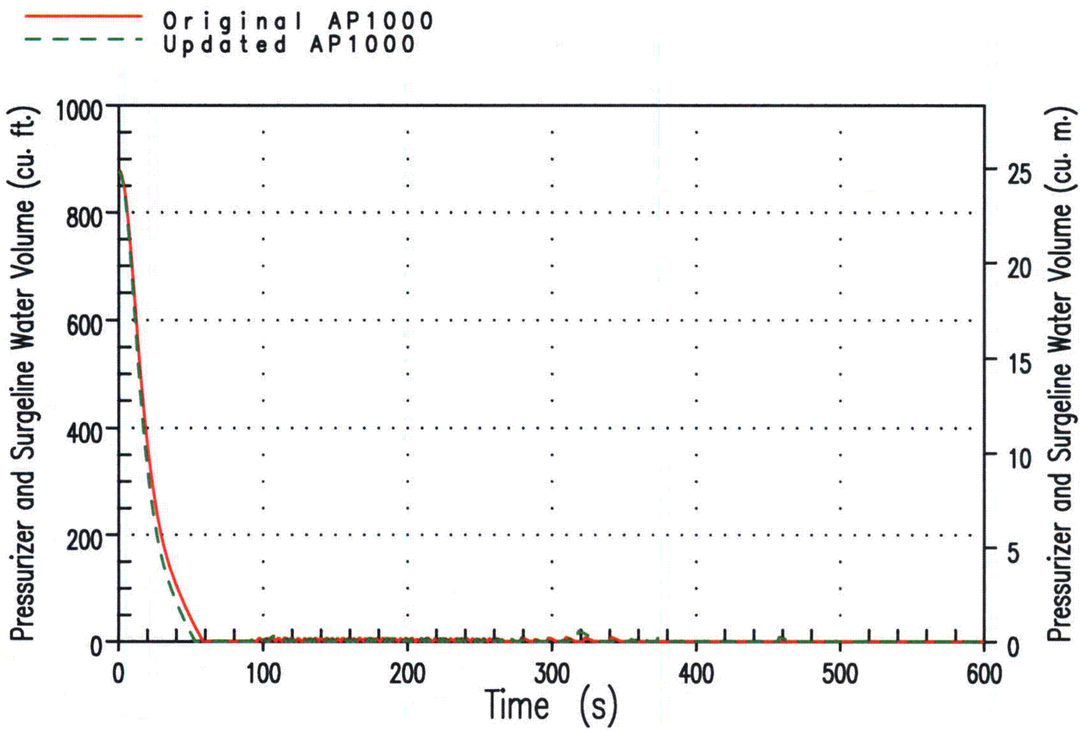


Figure 1-7
Steam Line Break

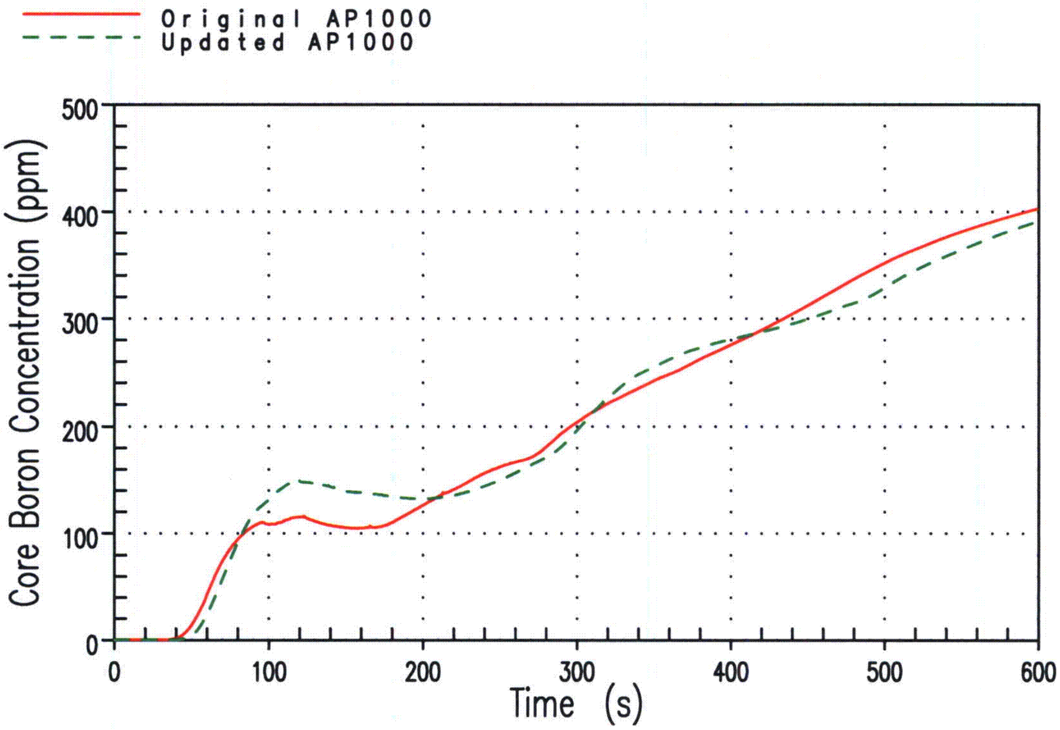


Figure 1-8
Steam Line Break

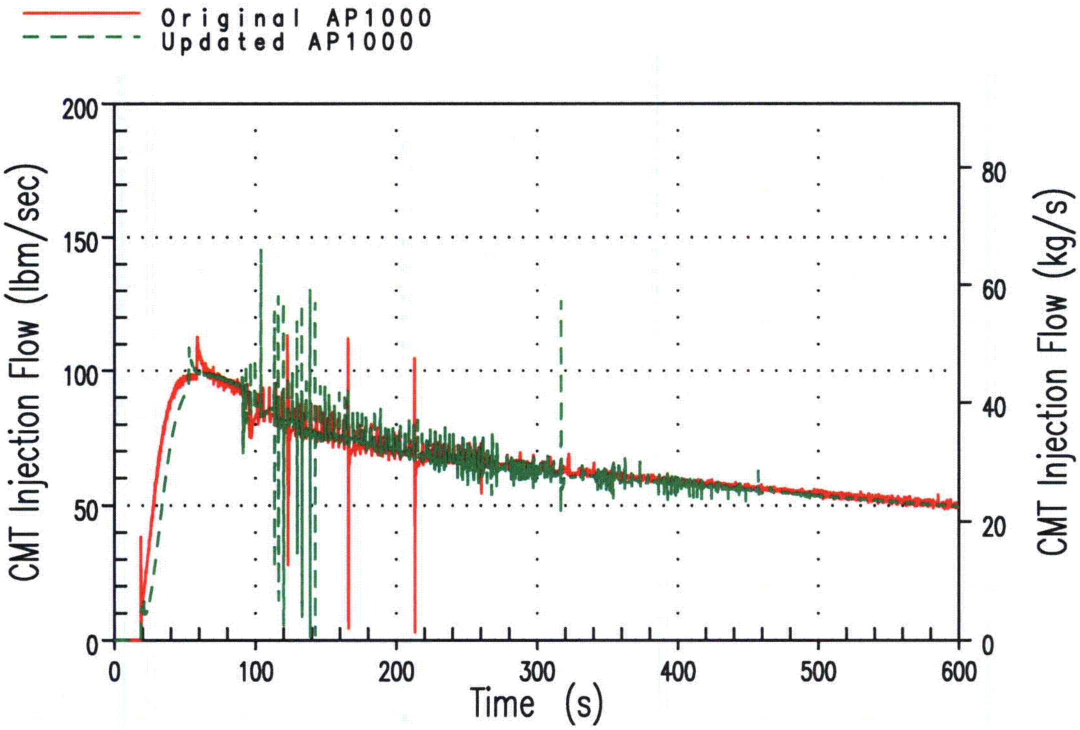


Figure 1-9
Steam Line Break

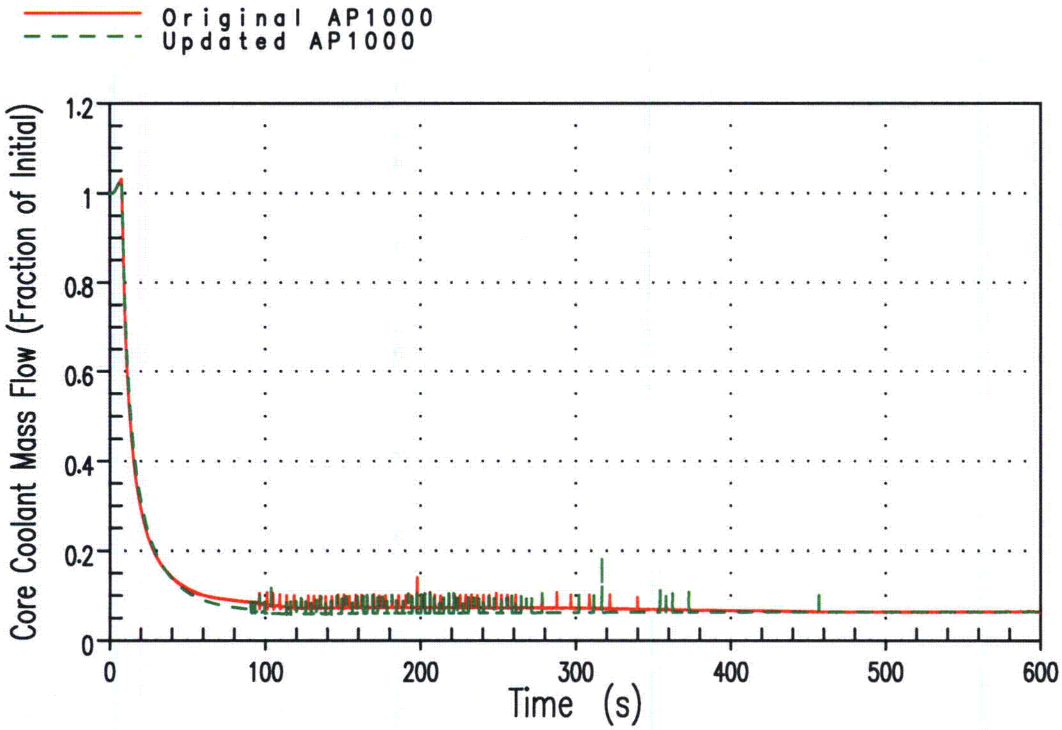


Figure 1-10
Steam Line Break

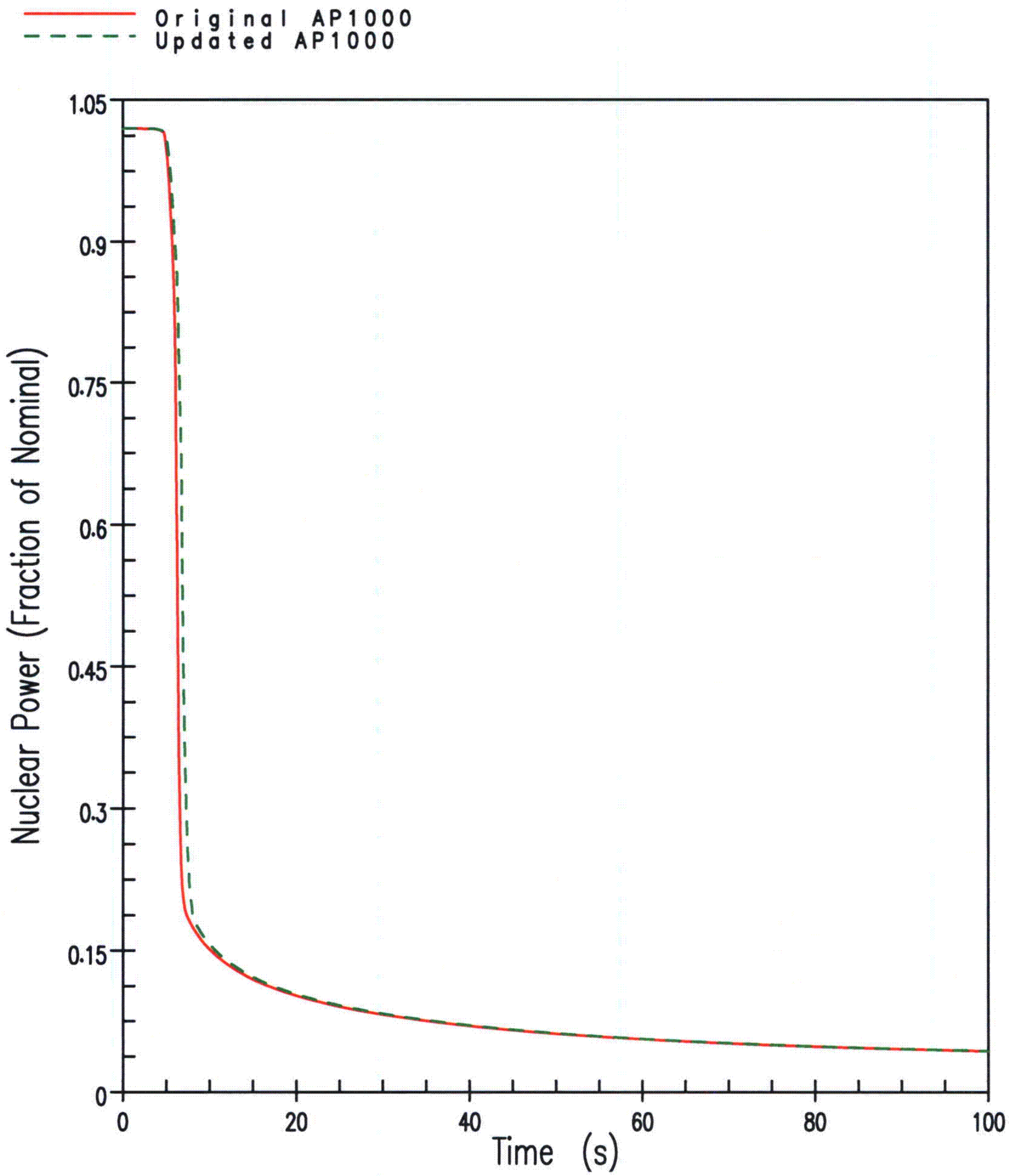


Figure 2-1
Turbine Trip
(Peak Pressure Case)

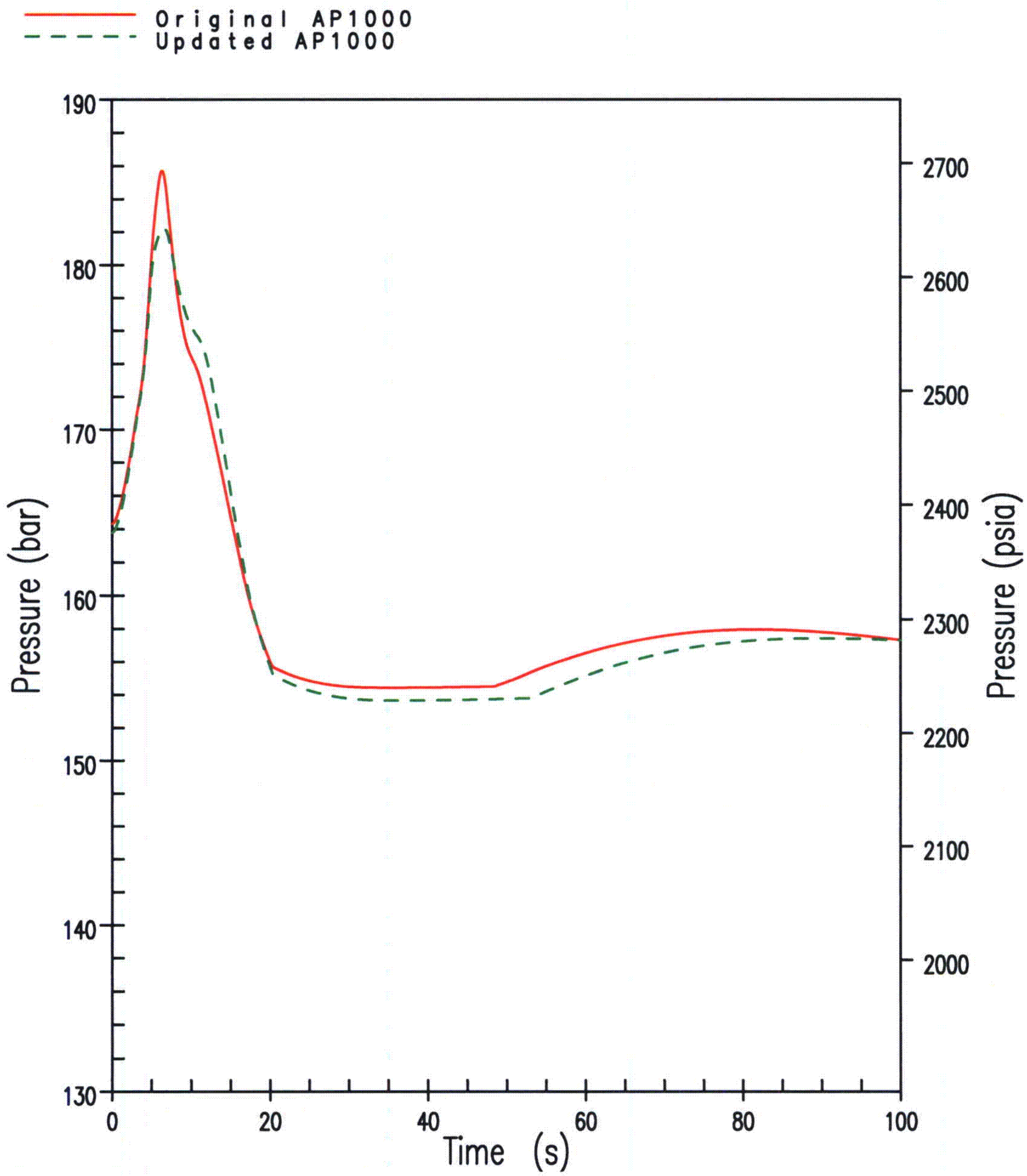


Figure 2-2 RCP Outlet Pressure
Turbine Trip
(Peak Pressure Case)

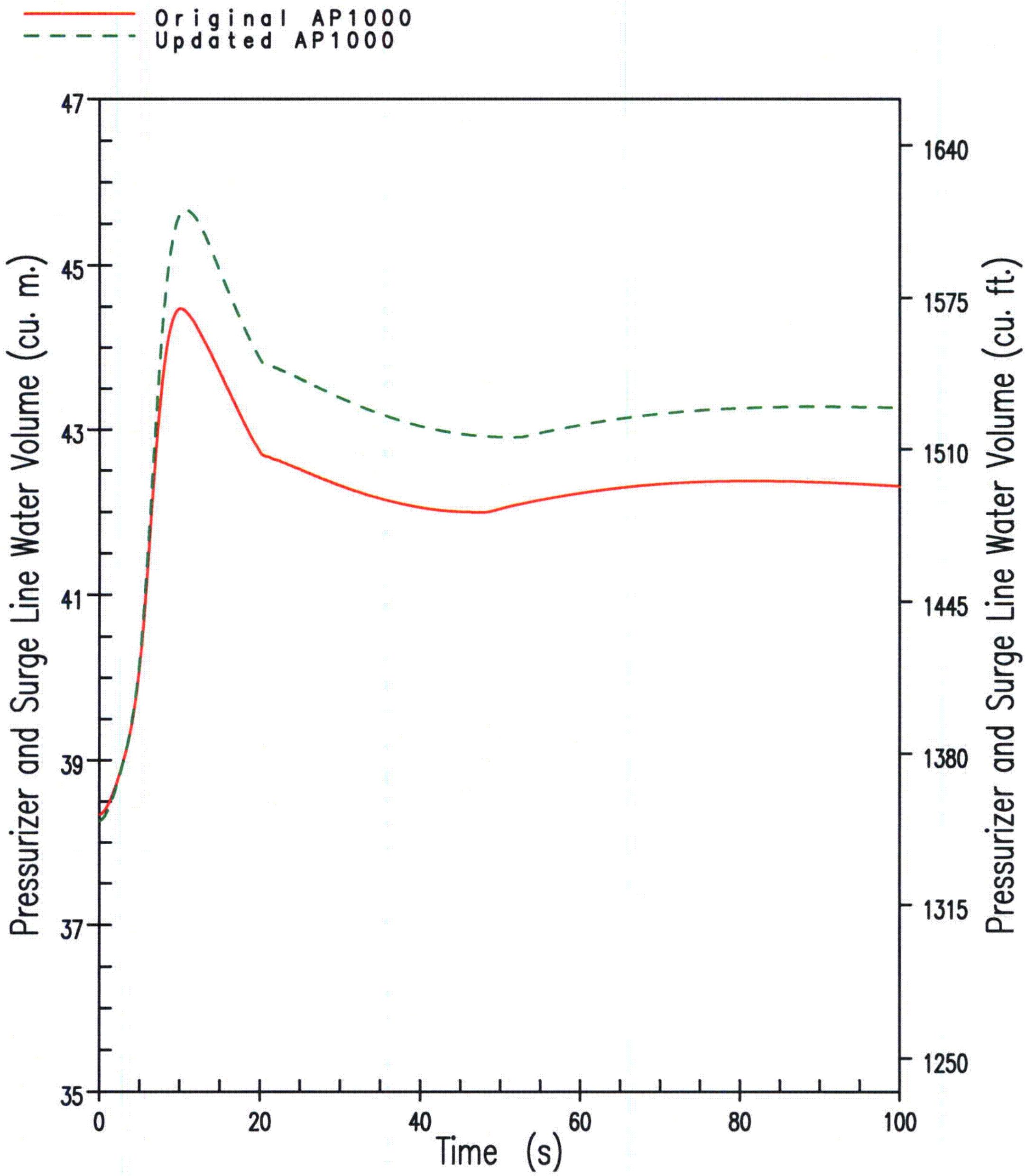


Figure 2-3
Turbine Trip
(Peak Pressure Case)

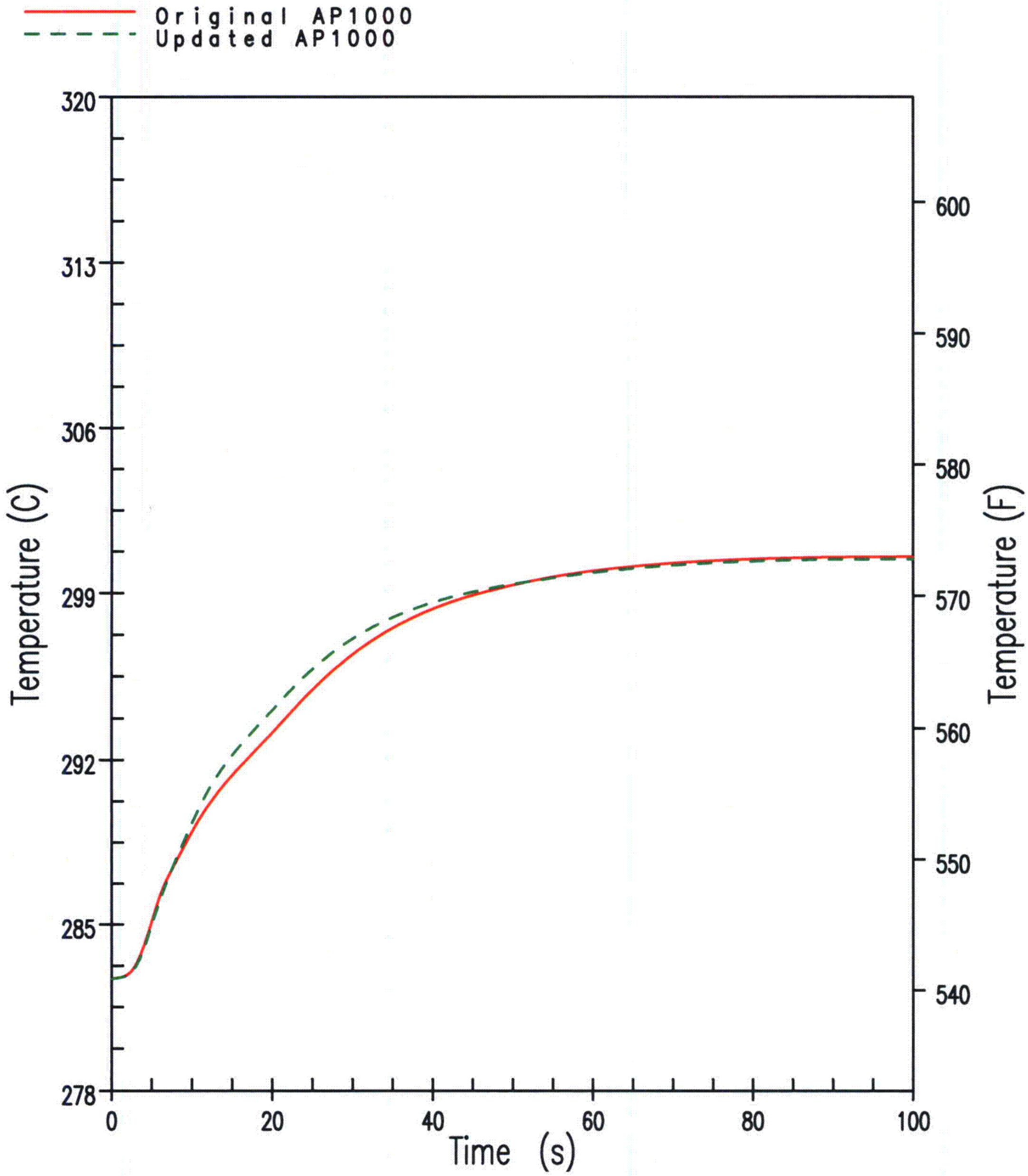


Figure 2-4 Vessel Inlet Temperature
Turbine Trip
(Peak Pressure Case)

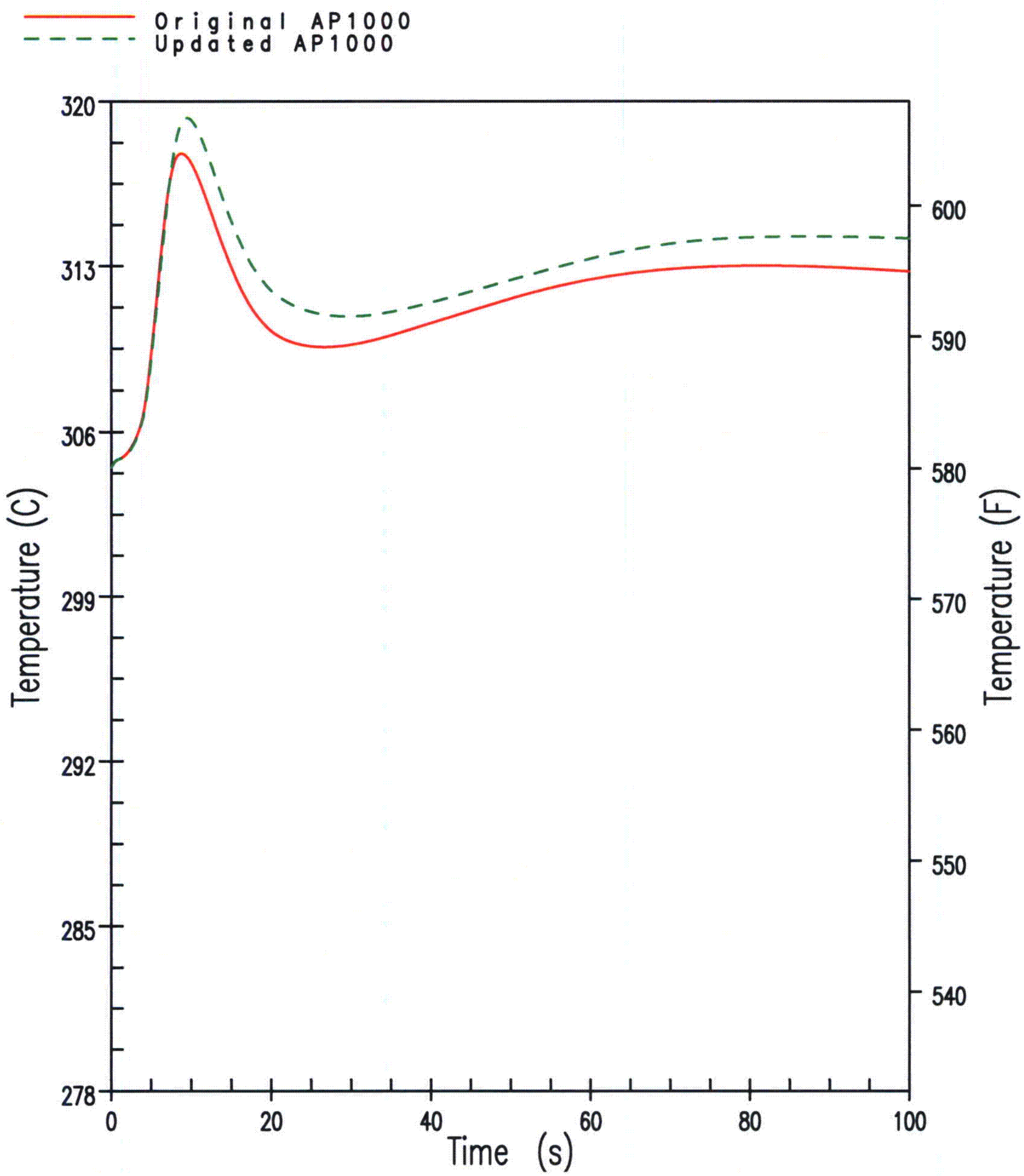


Figure 2-5 Vessel Average Temperature
Turbine Trip
(Peak Pressure Case)

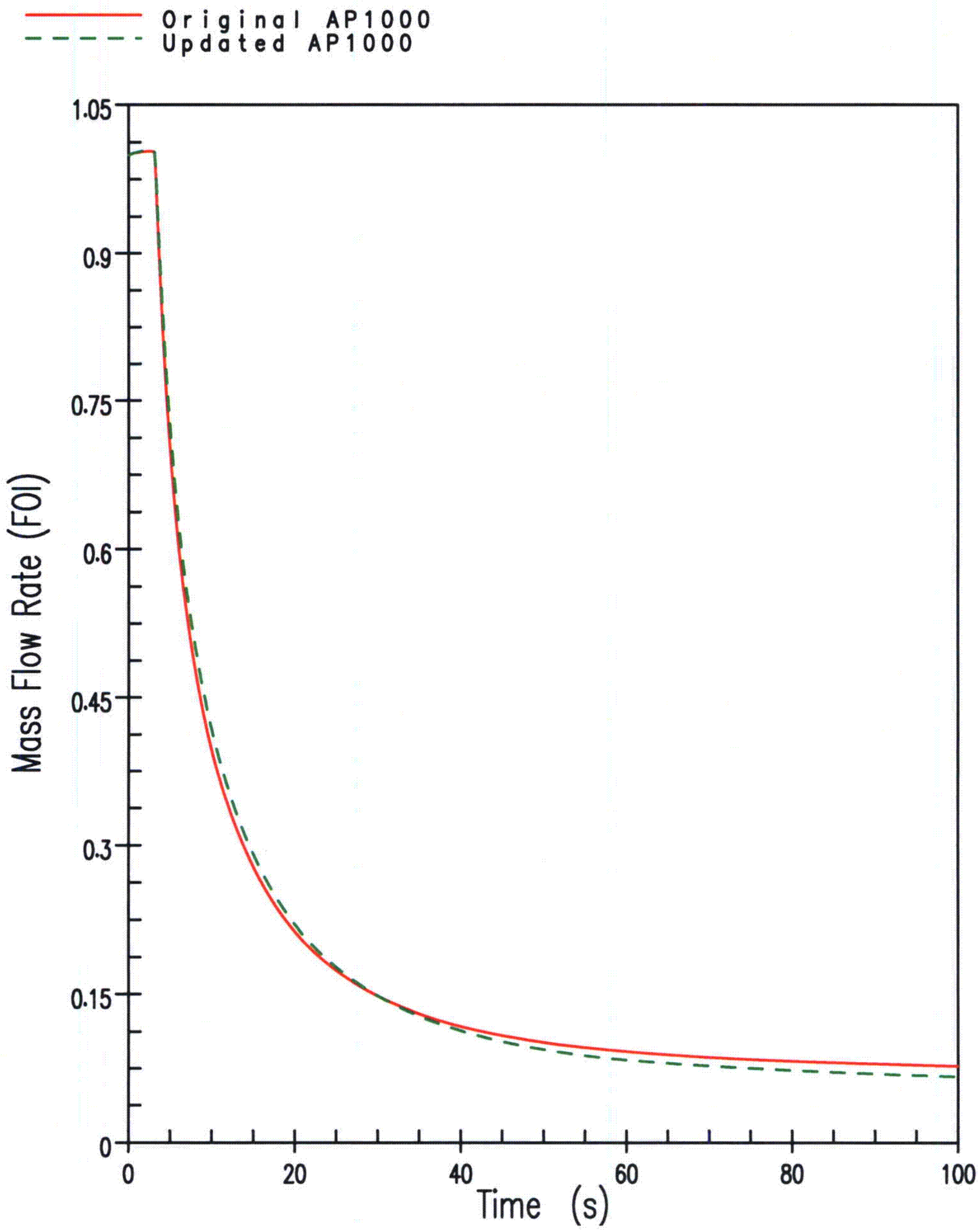


Figure 2-6 Core Coolant Mass Flow
Turbine Trip
(Peak Pressure Case)

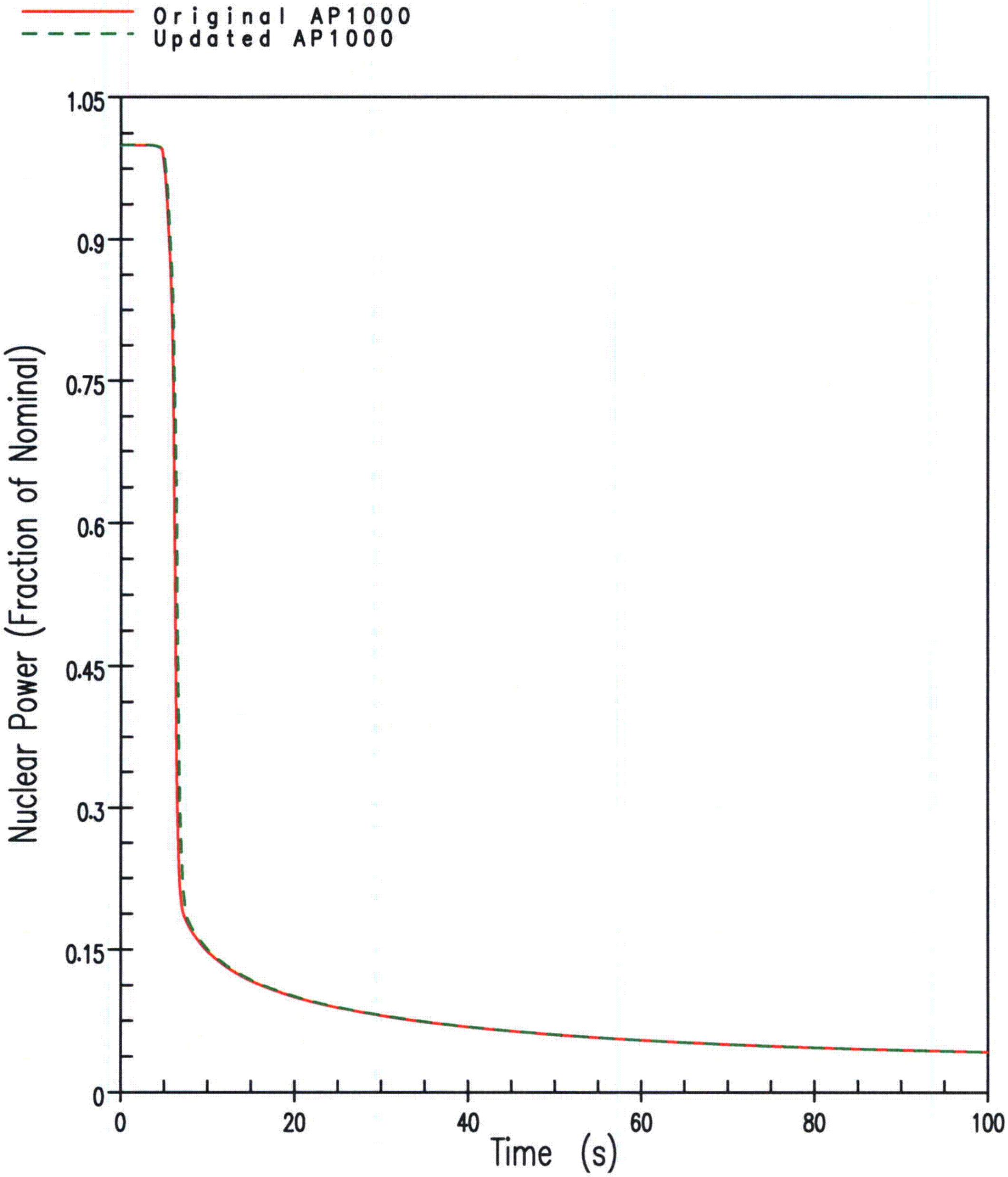


Figure 2-7
Turbine Trip
(Minimum DNBR Case)

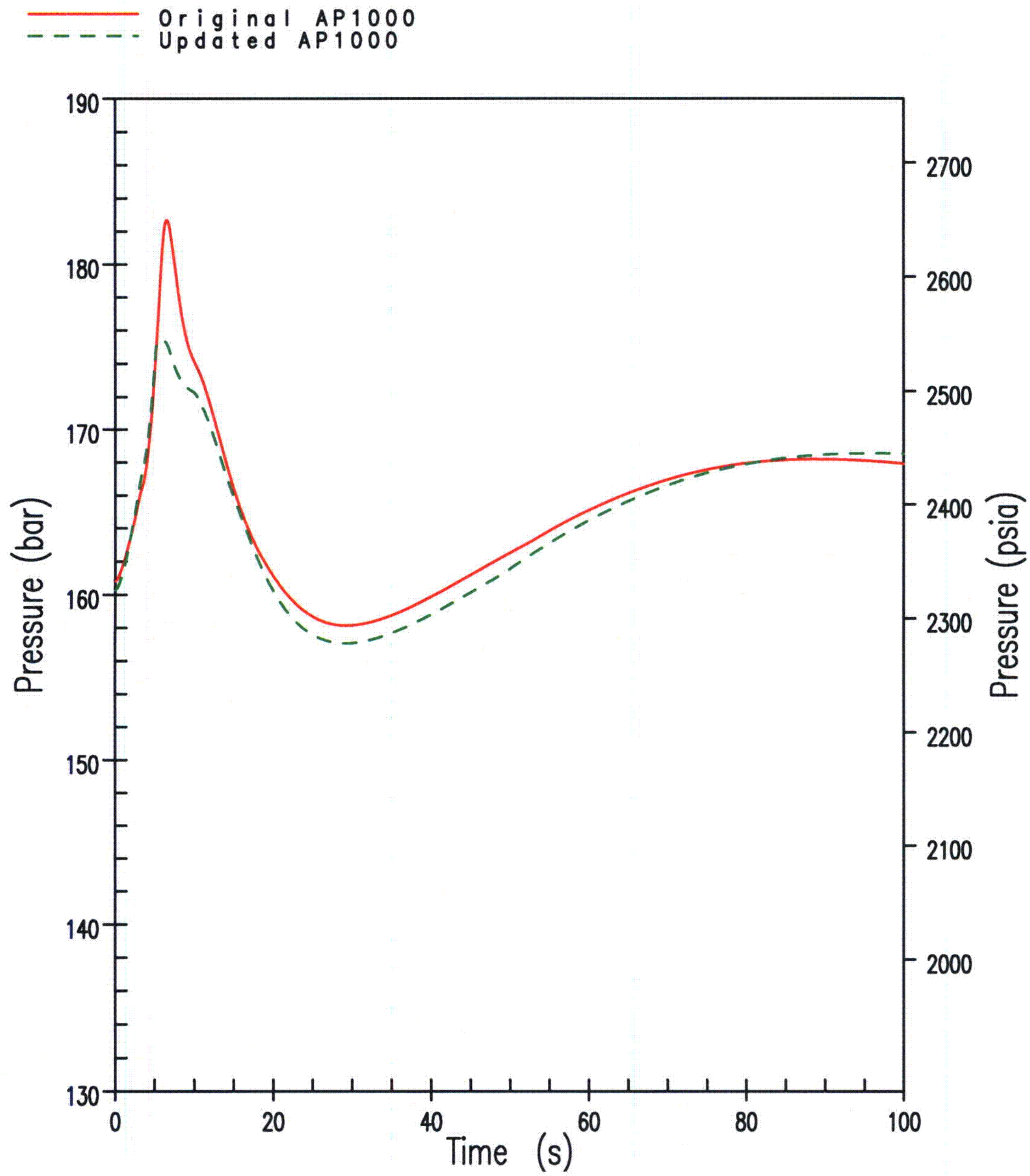


Figure 2-8 RCP Outlet Pressure
Turbine Trip
(Minimum DNBR Case)

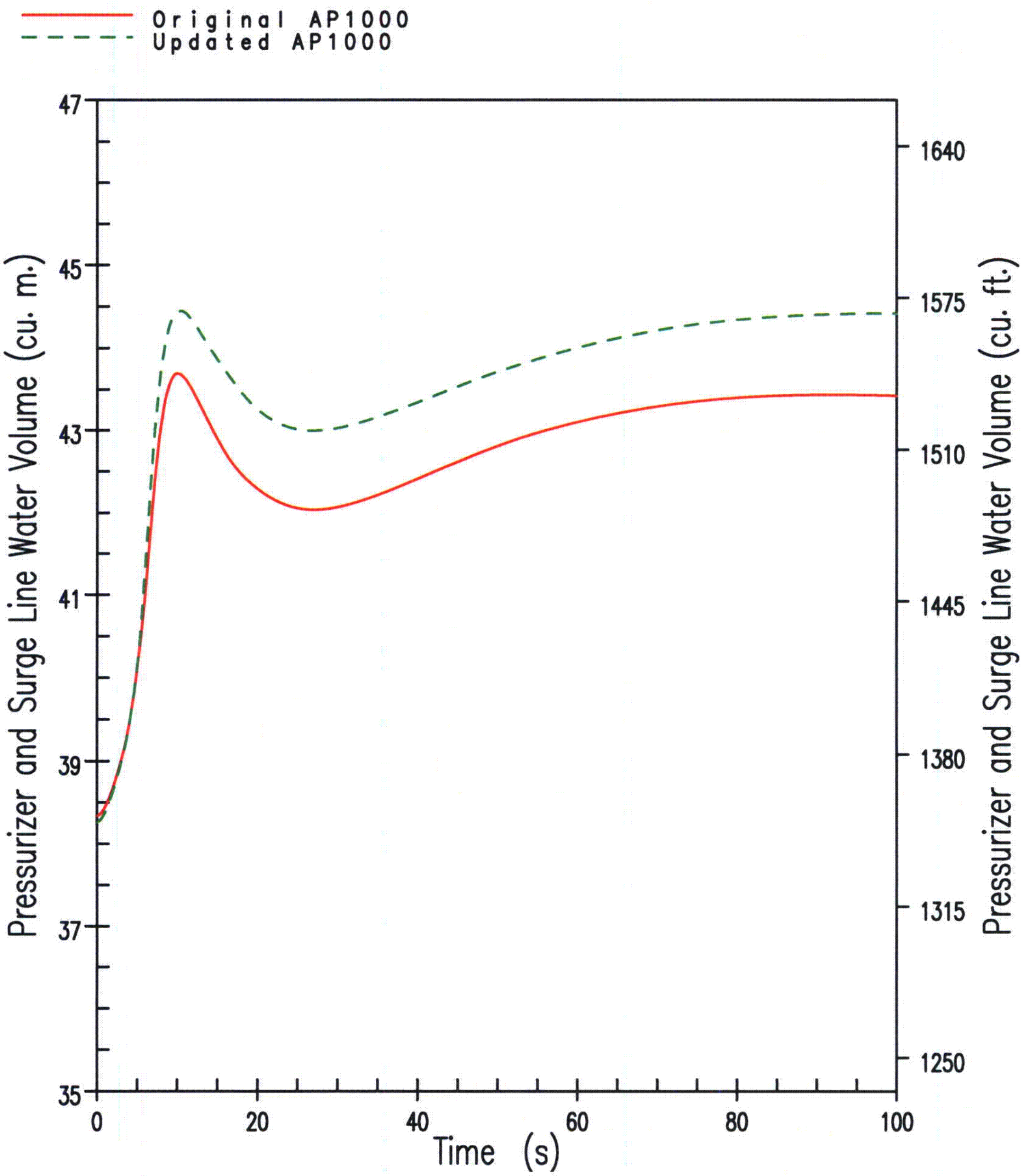


Figure 2-9 Pressurizer and Surgeline Water Volume
Turbine Trip
(Minimum DNBR Case)

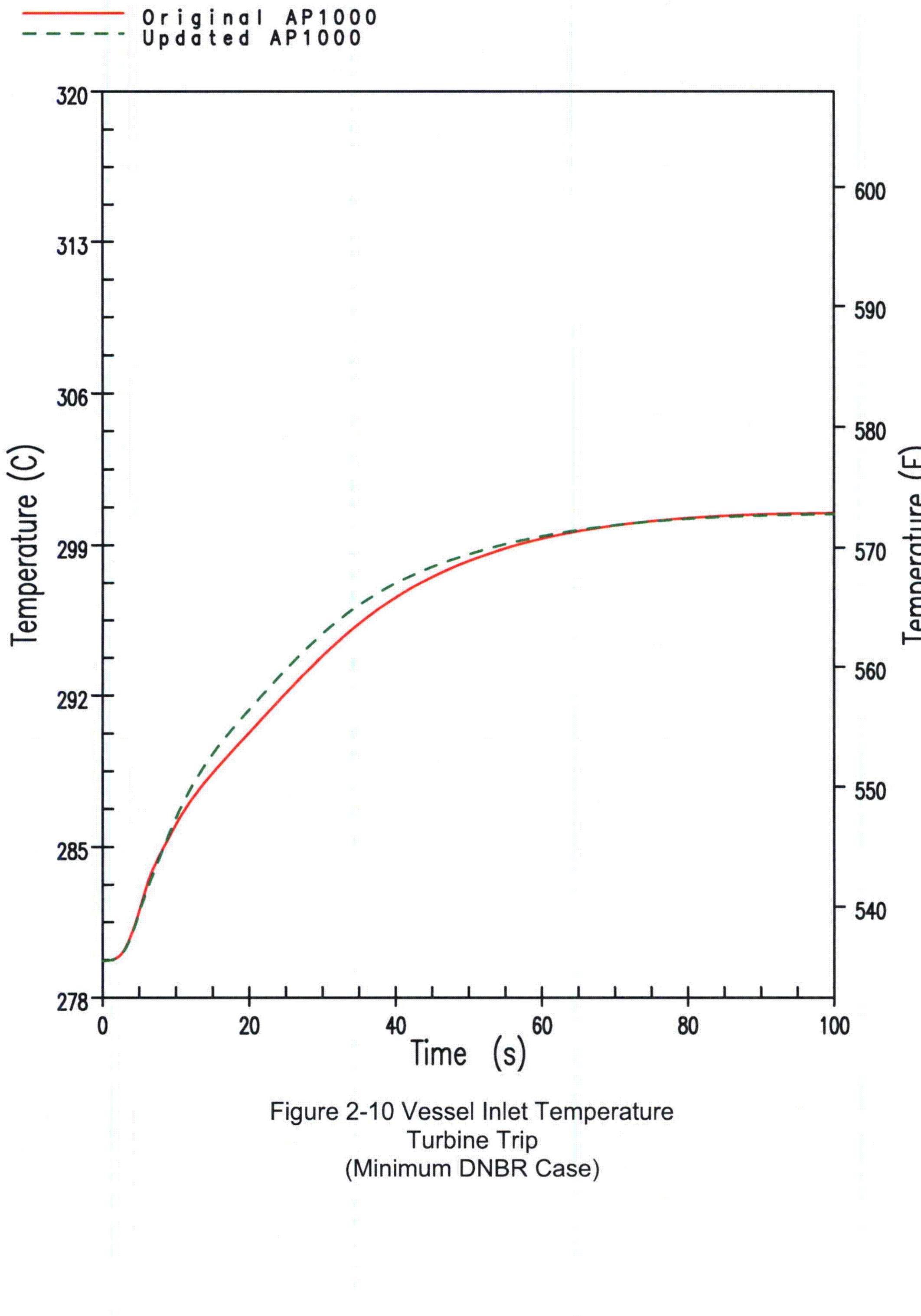


Figure 2-10 Vessel Inlet Temperature
Turbine Trip
(Minimum DNBR Case)

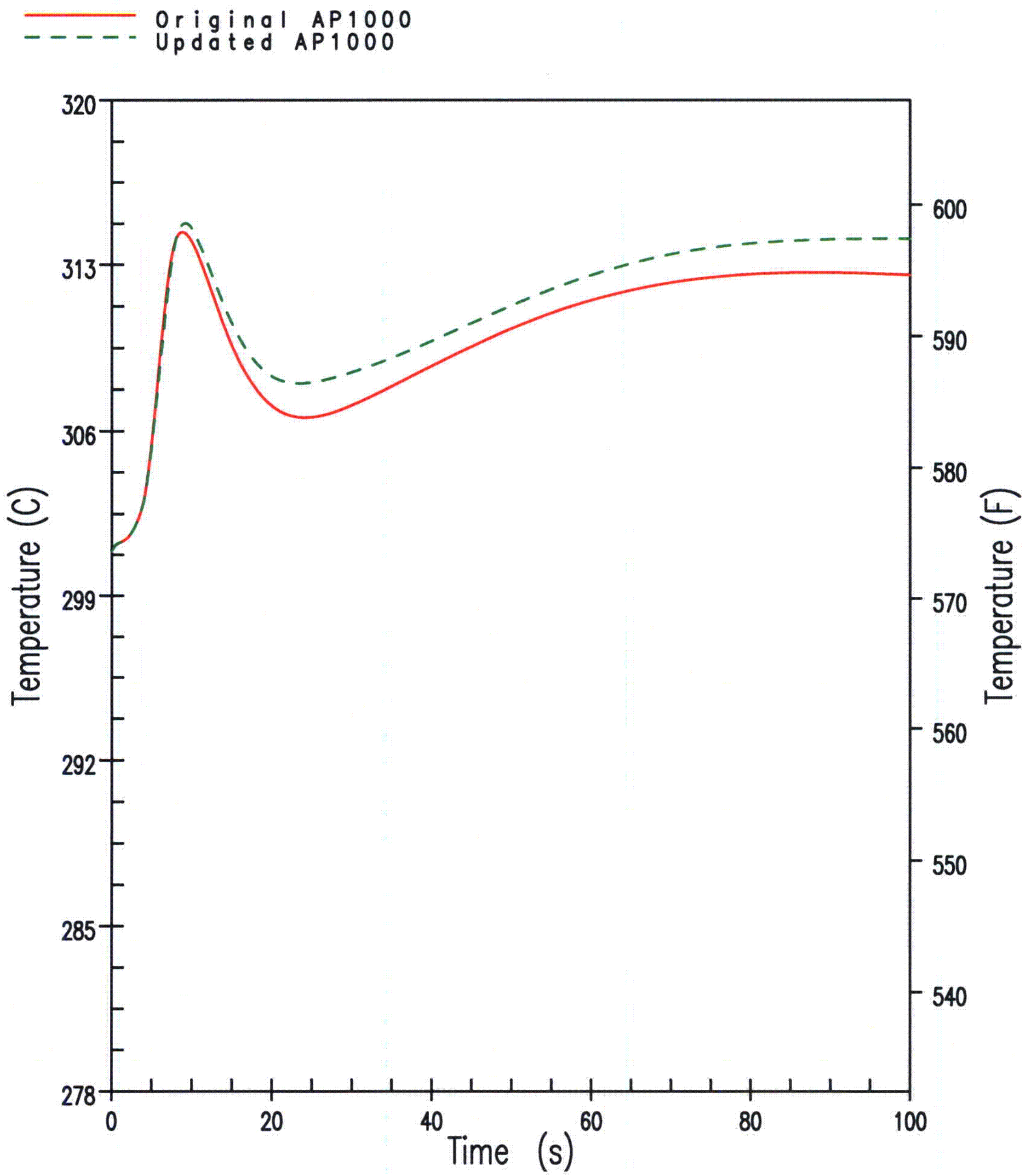


Figure 2-11 Vessel Average Temperature
Turbine Trip
(Minimum DNBR Case)

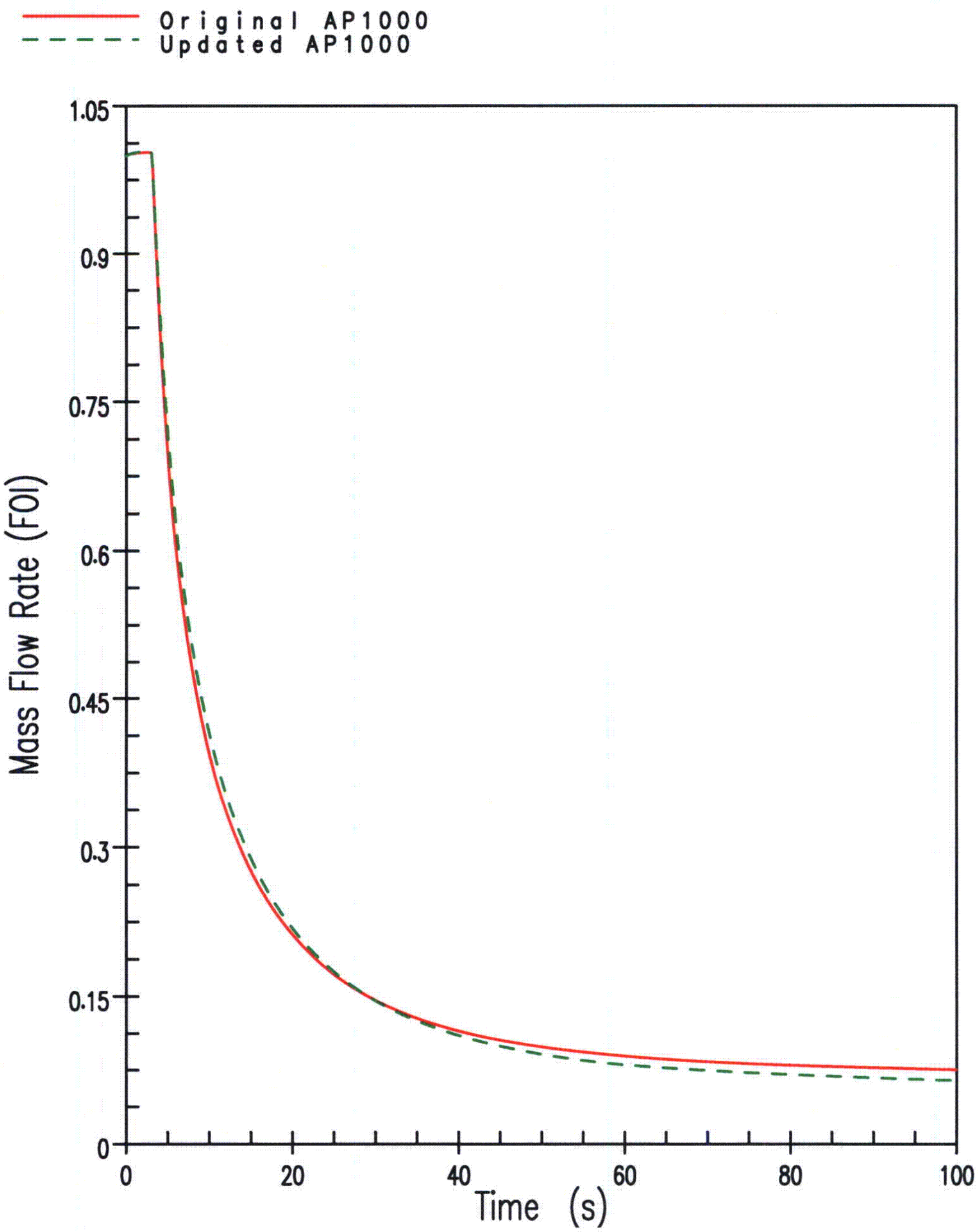


Figure 2-12 Vessel Mass Flow
Turbine Trip
(Minimum DNBR Case)

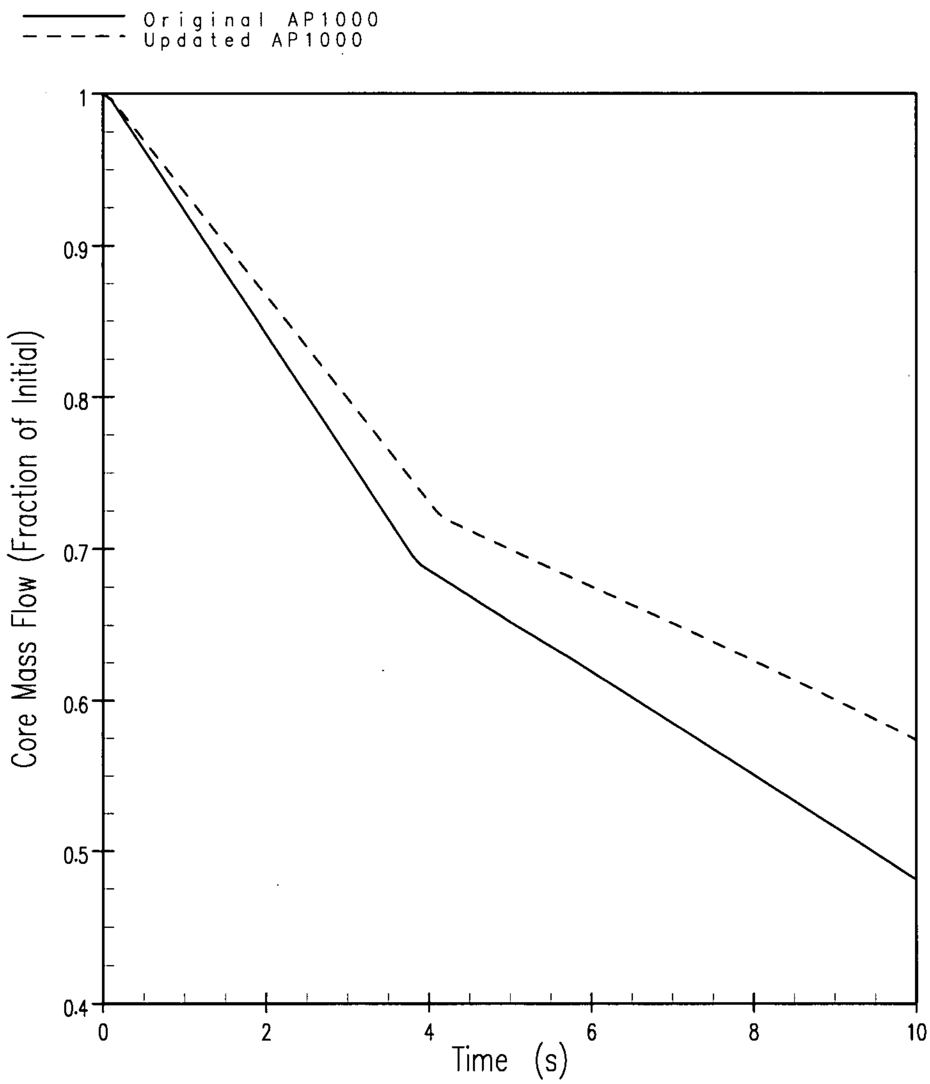


Figure 3-1
Four Cold Legs in Operation, Two Pumps Coasting Down

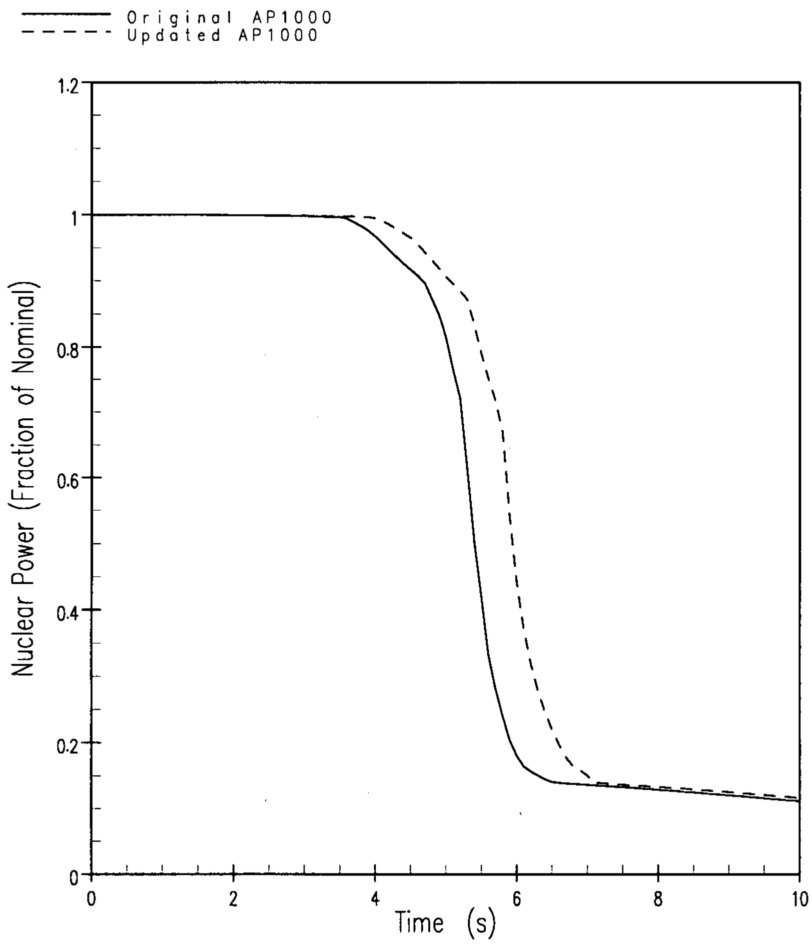


Figure 3-2
Four Cold Legs in Operation, Two Pumps Coasting Down

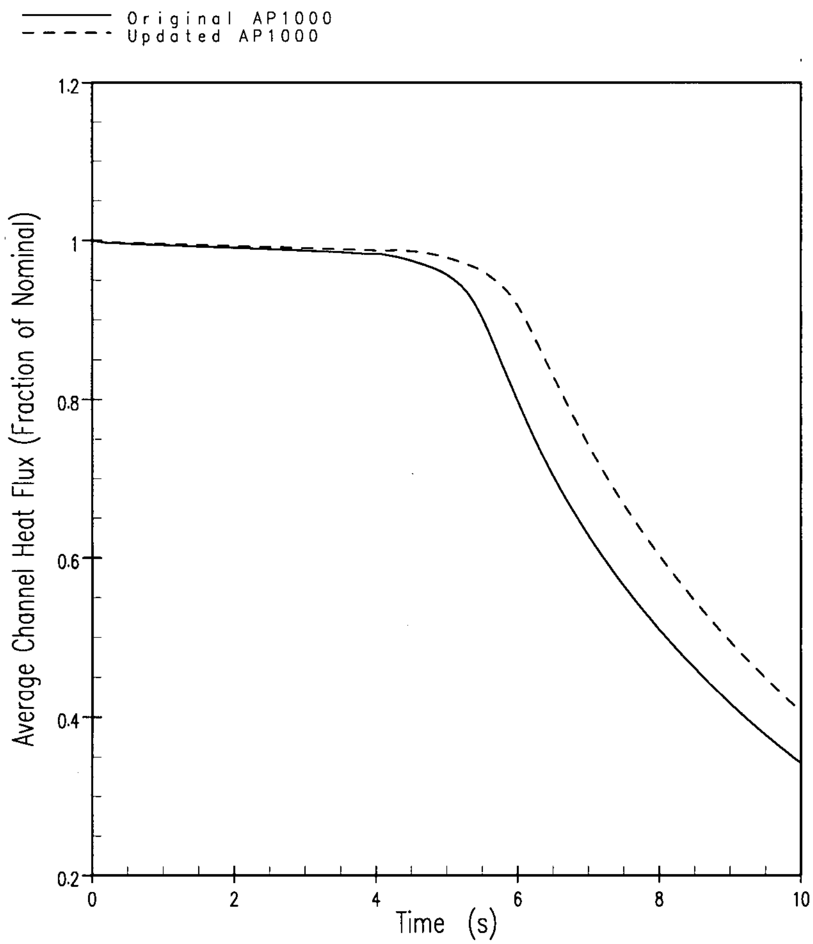


Figure 3-3
Four Cold Legs in Operation, Two Pumps Coasting Down

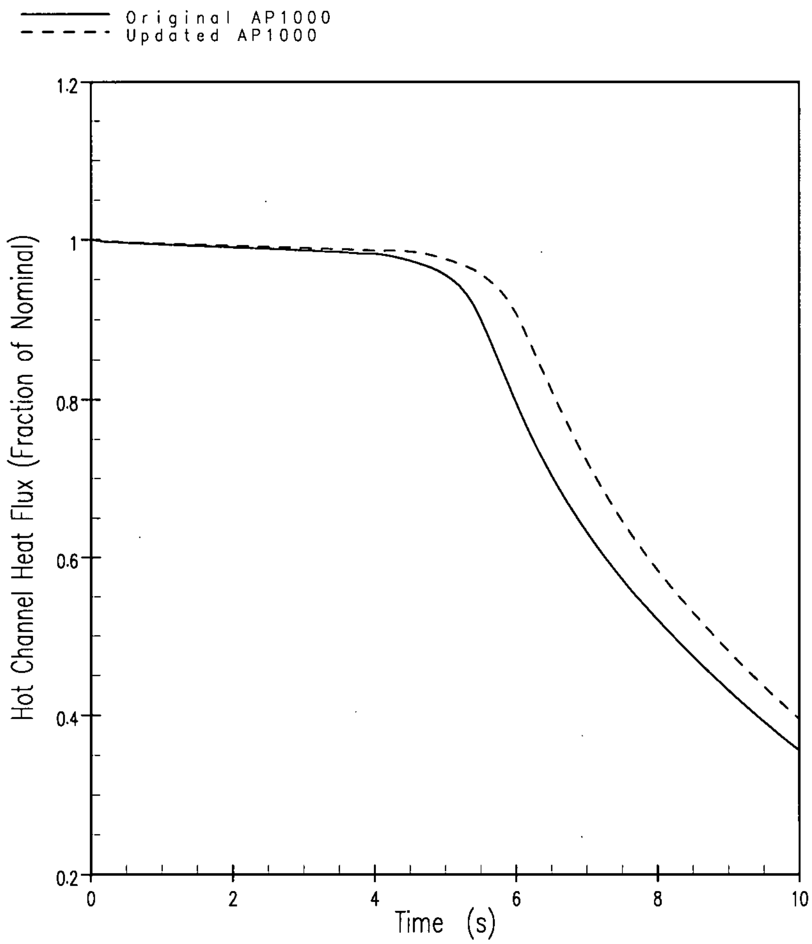


Figure 3-4
Four Cold Legs in Operation, Two Pumps Coasting Down

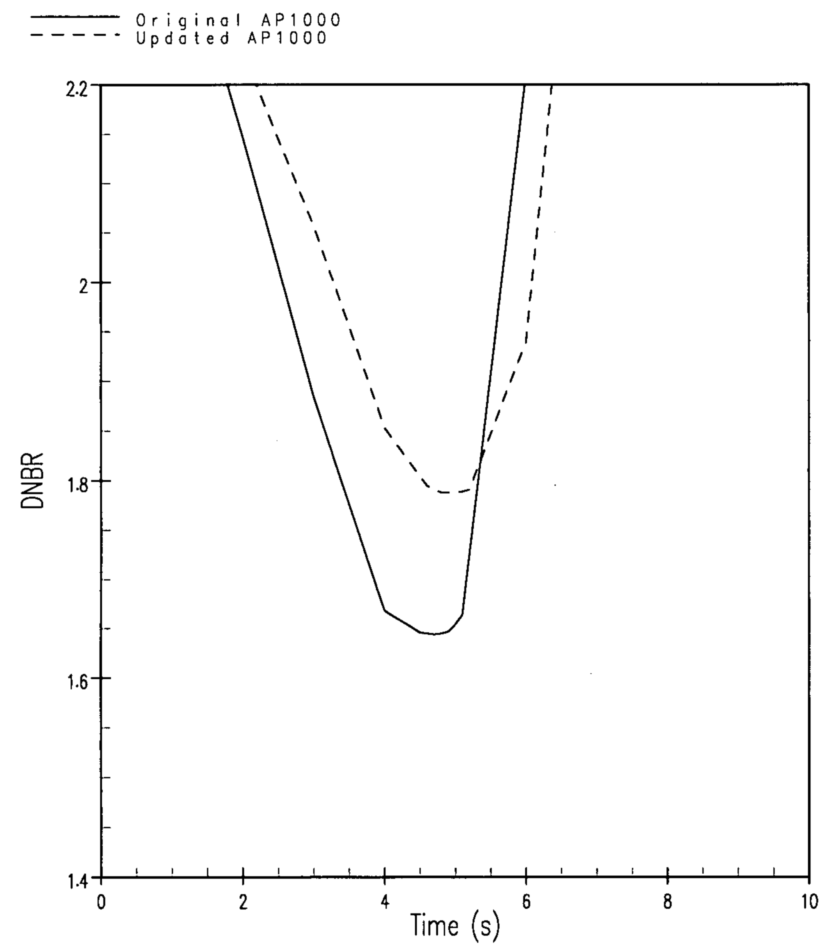


Figure 3-5
Four Cold Legs in Operation, Two Pumps Coasting Down

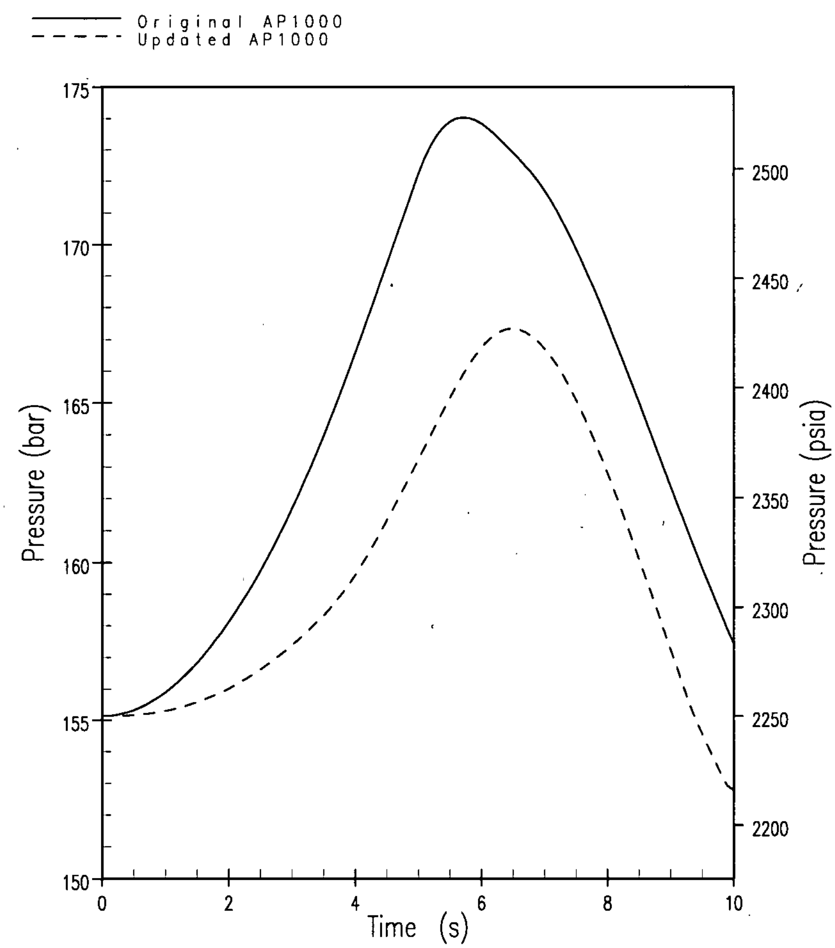


Figure 3-6
Four Cold Legs in Operation, Two Pumps Coasting Down

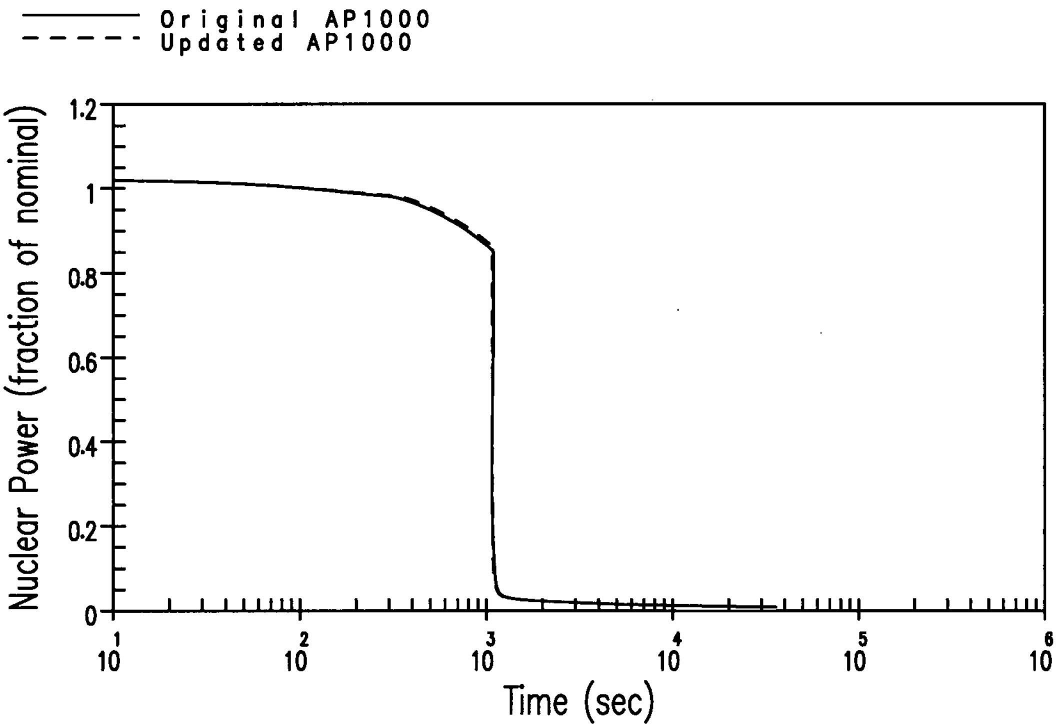


Figure 5-1
Nuclear Power
CVS Malfunction

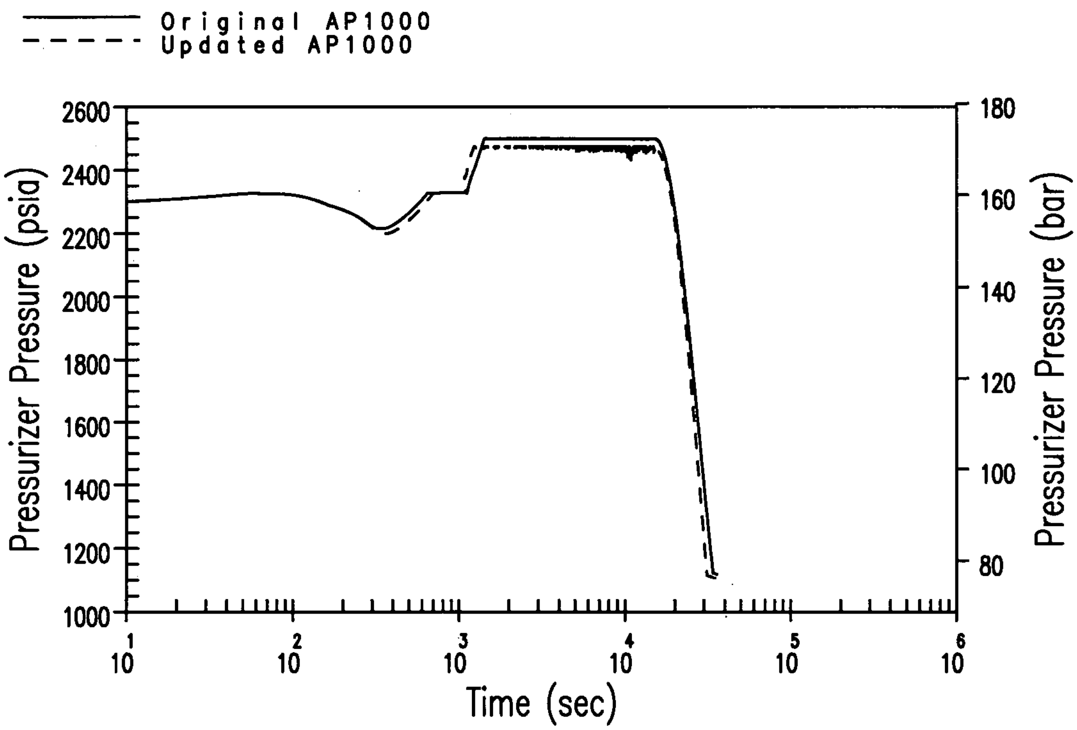


Figure 5-2
Pressurizer Pressure
CVS Malfunction

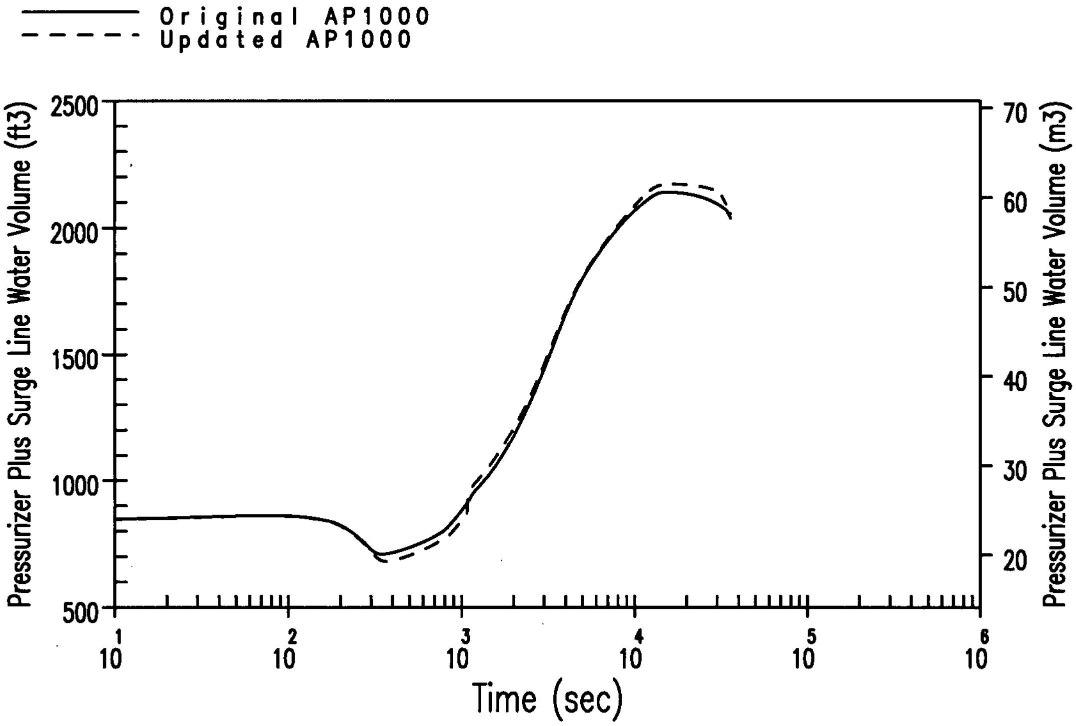


Figure 5-3
Pressurizer Water Volume
CVS Malfunction

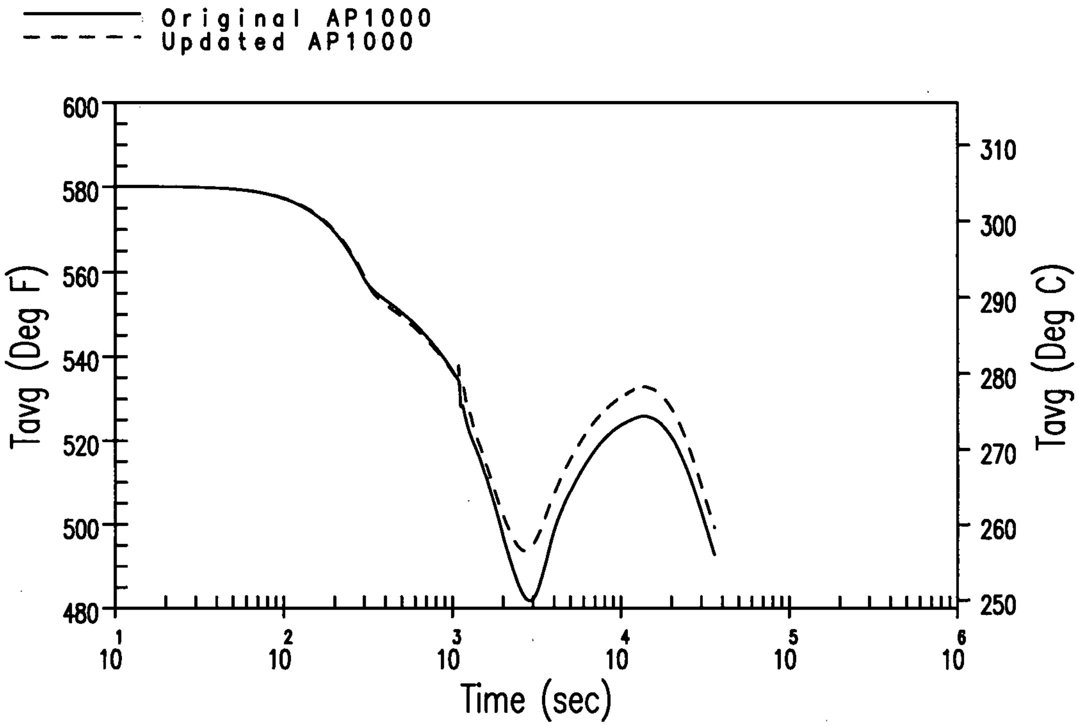


Figure 5-4
RCS Average Temperature
CVS Malfunction

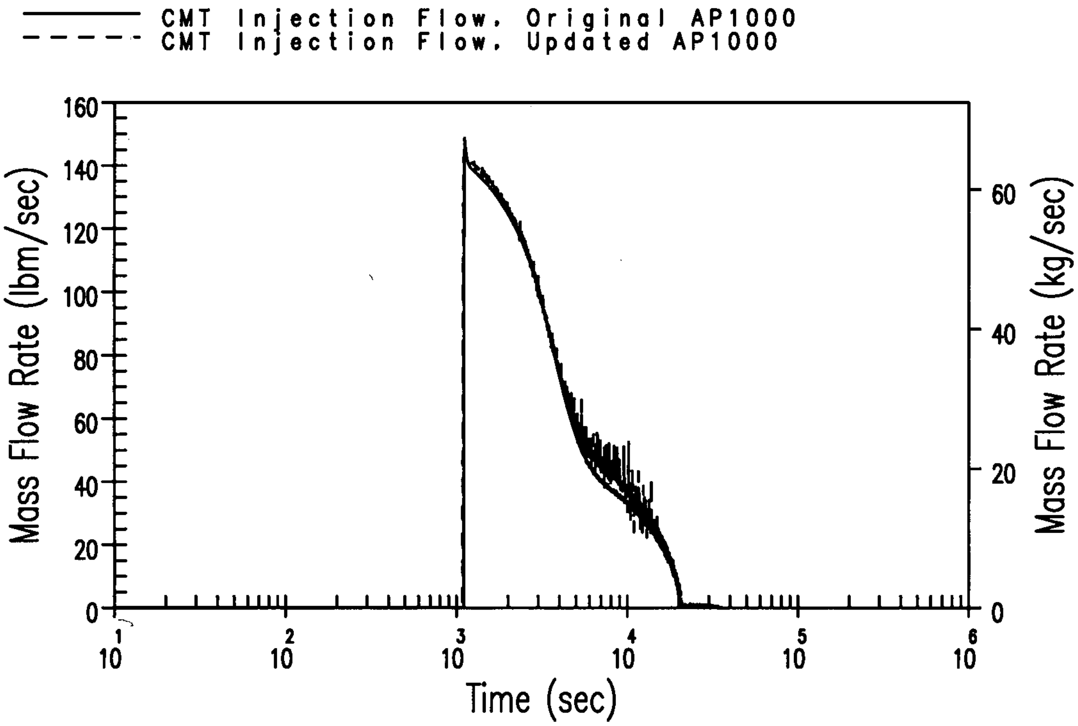


Figure 5-5
CMT Injection Flow Rate
CVS Malfunction

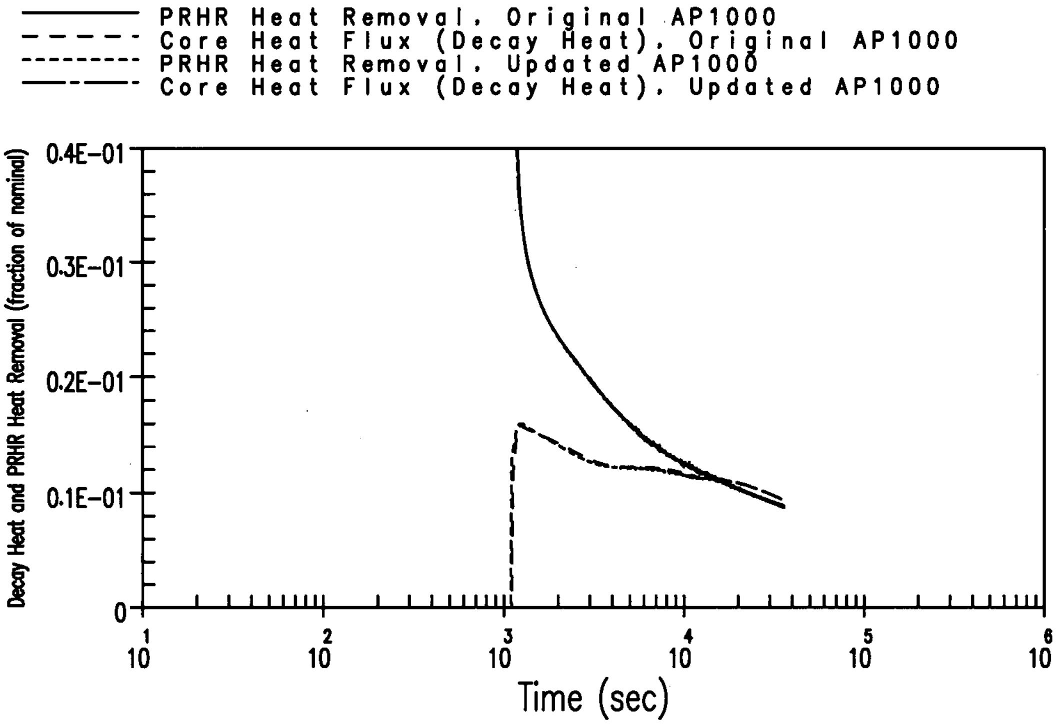


Figure 5-6
Decay Heat and PRHR Heat Removal
CVS Malfunction

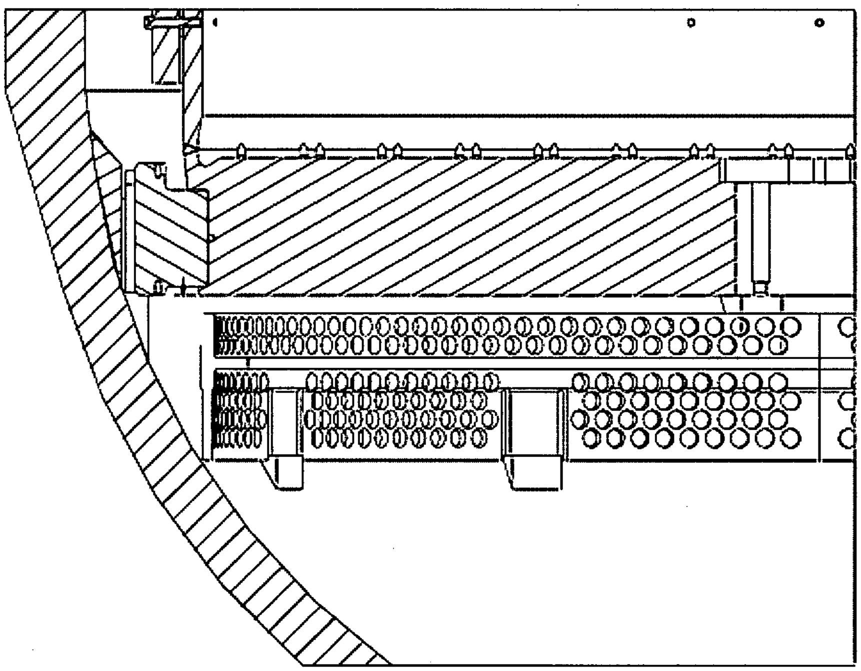


Figure 6-1
Elevation View of Reactor Bottom Vessel Head, Flow Skirt, and Lower Core Support Plate
(from WCAP-1617-NP Figure 3-1)

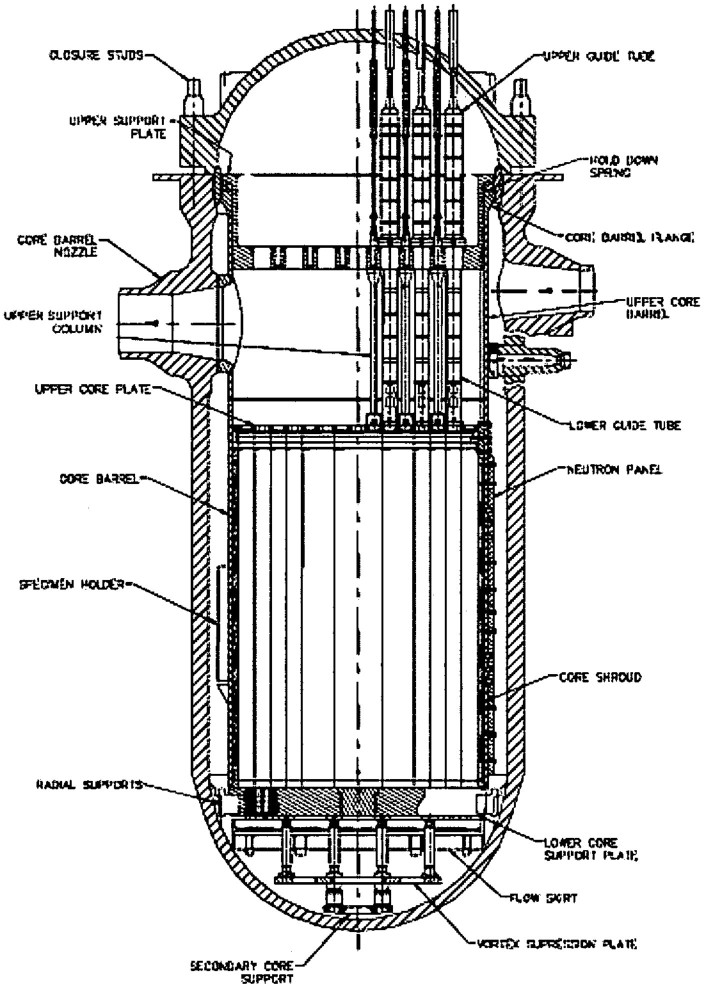


Figure 6-2
Changes to Reactor Internals (from WCAP-16716-NP Figure 4-1)

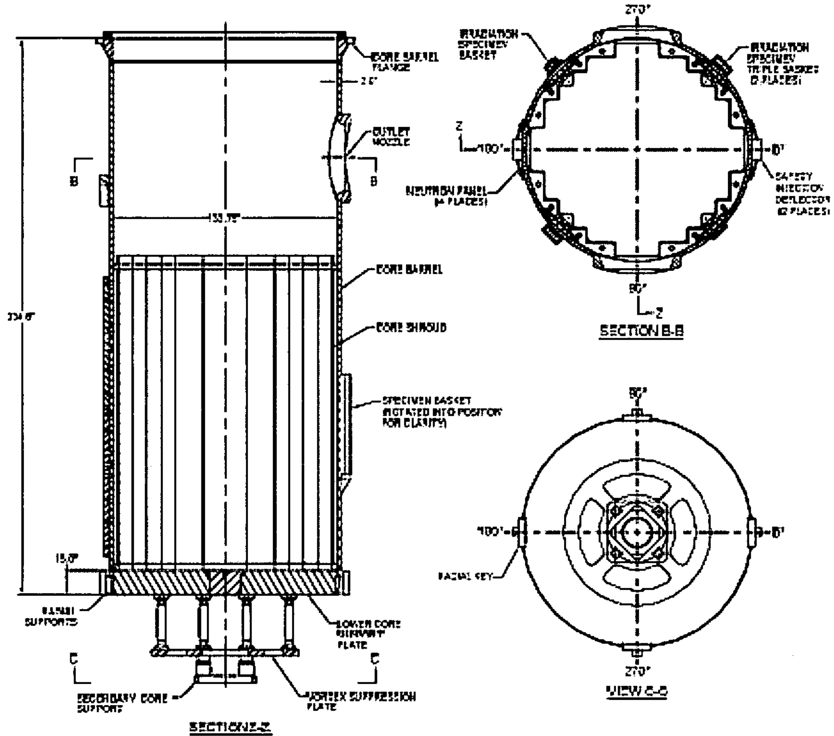


Figure 6-3
Changes to Reactor Internals (from WCAP-1617-NP Figure 4-2)

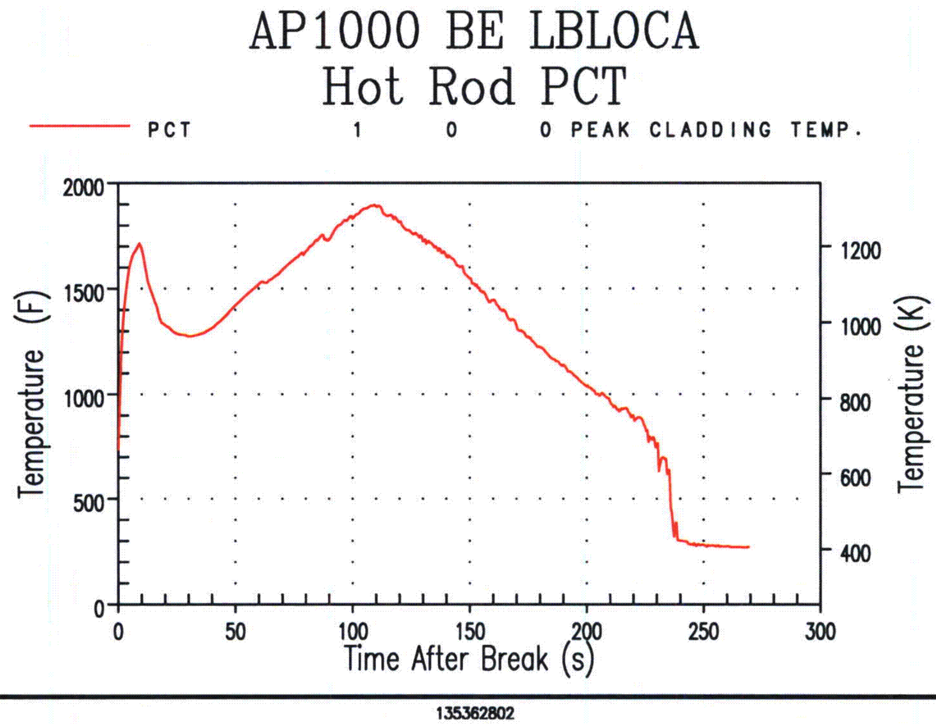


Figure 6-4
Hot Rod PCT (first data series from DCD, Fig. 15.6.5.4A-1)

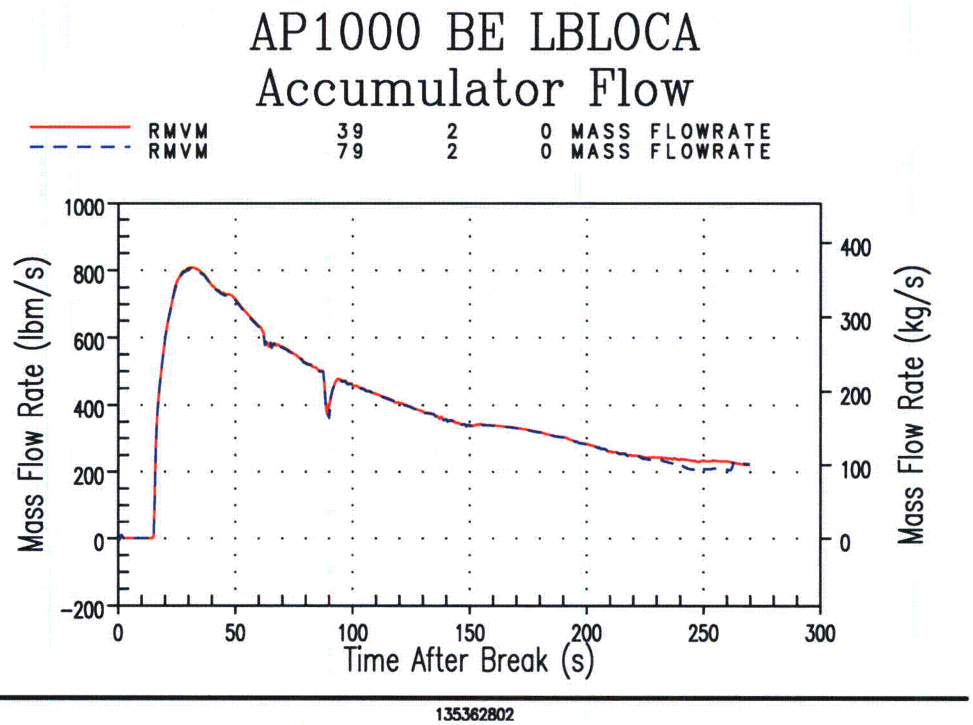


Figure 6-5
Accumulator Flows
(compare to Figure 15.6.5.4A-5 of DCD)

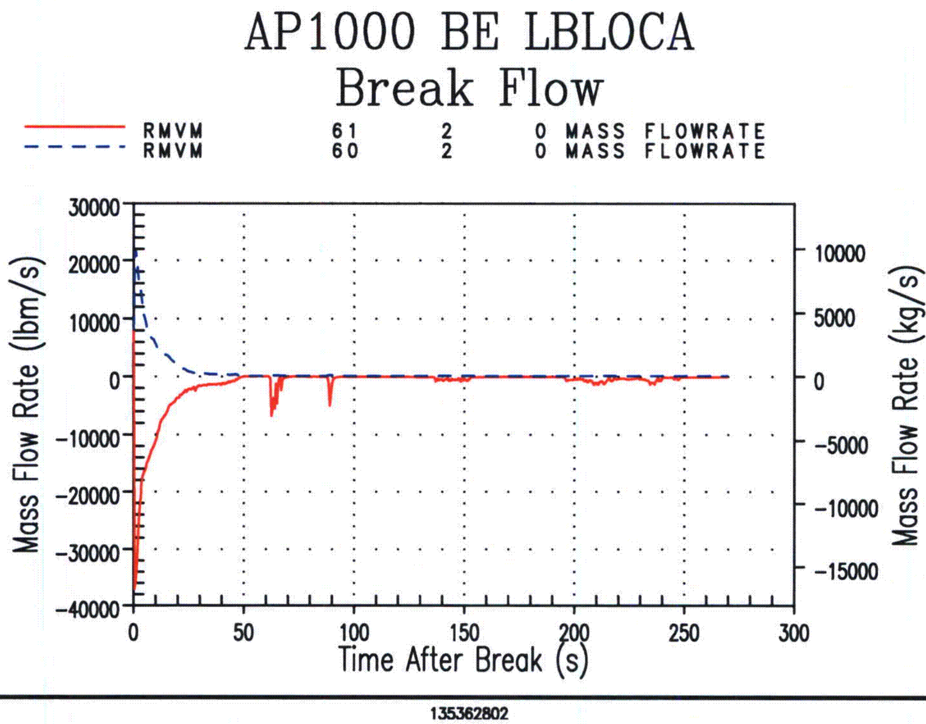


Figure 6-6
Break Flows (Component 61 is vessel side, Component 60 is pump side; both flows are out of RCS loop)

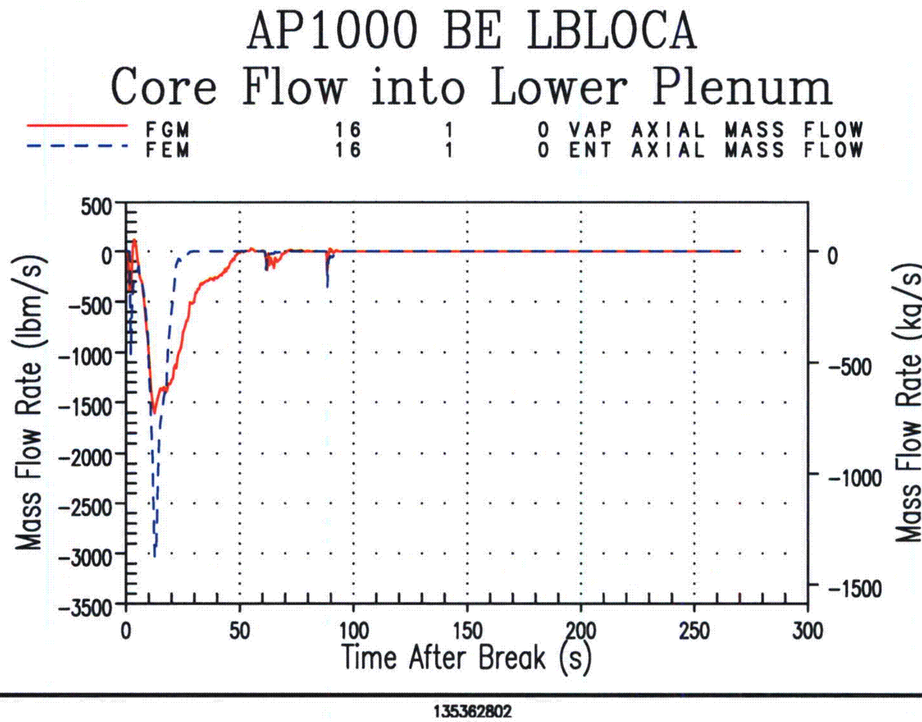


Figure 6-7
Vapor and Entrained Liquid Flows across the lower core plate
(negative is downward into the lower plenum)

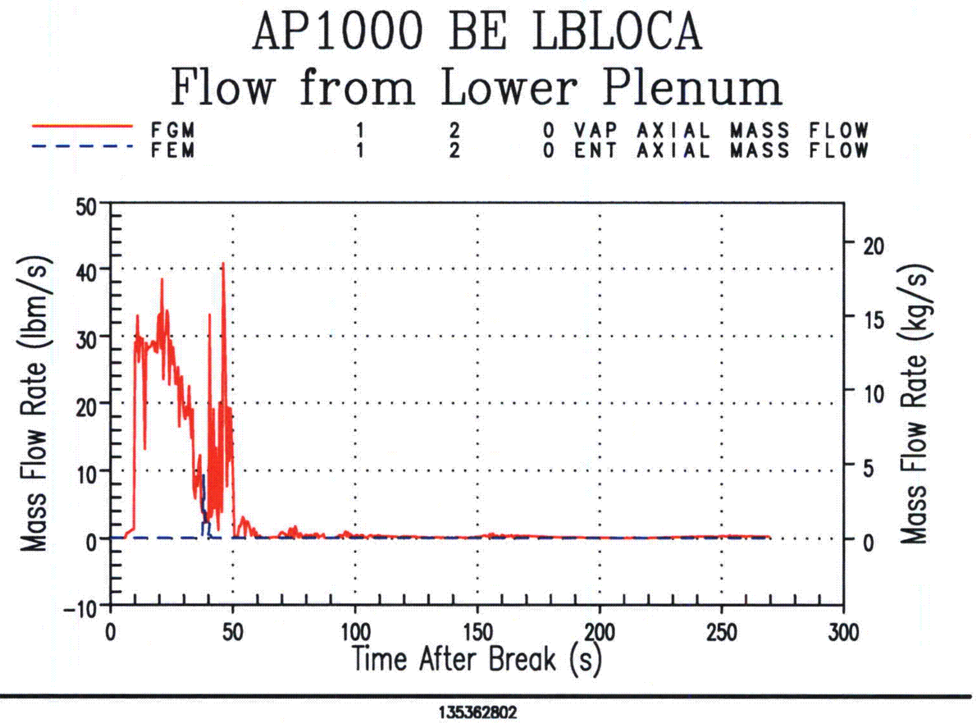


Figure 6-8
Vapor and Entrained Liquid Flows from the vessel bottom to the upper region of the lower plenum (flow skirt elevation)

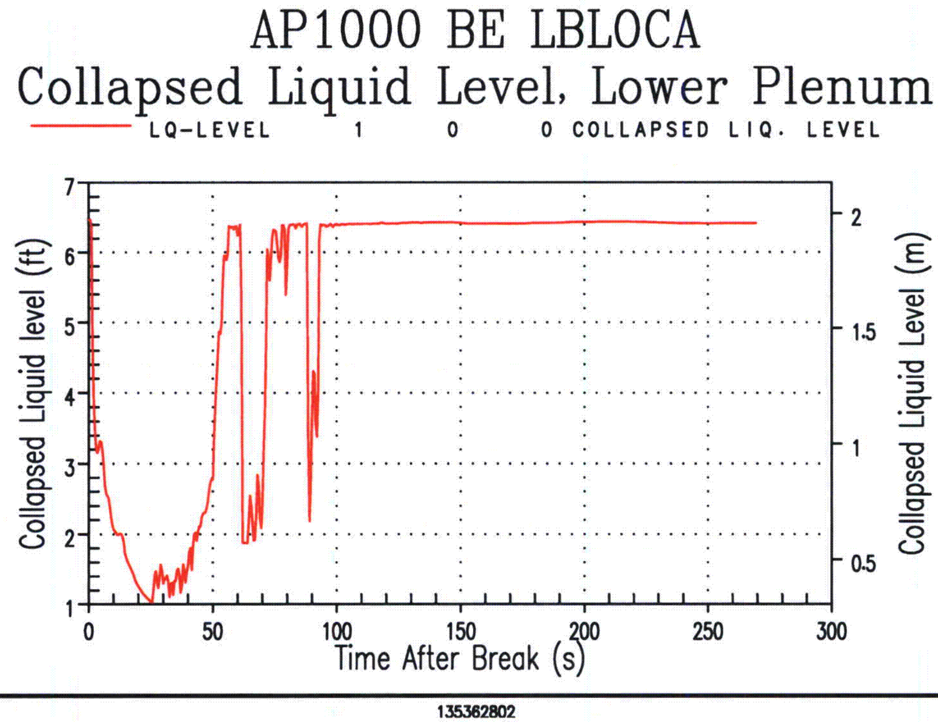


Figure 6-9
Lower Plenum Collapsed Liquid Level

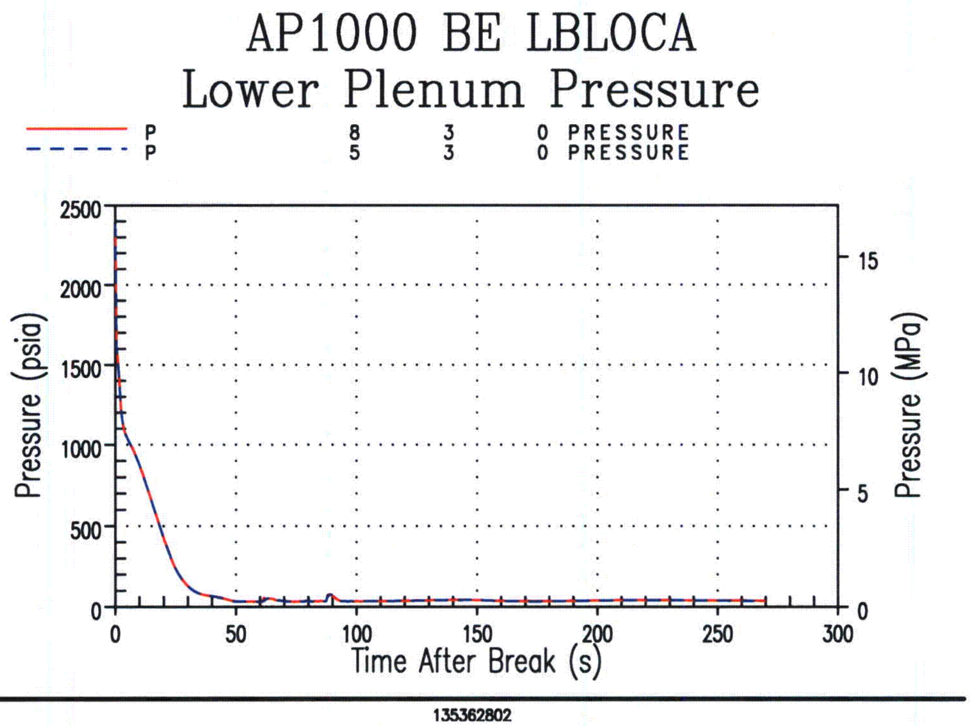


Figure 6-10
Pressure in the Lower Plenum

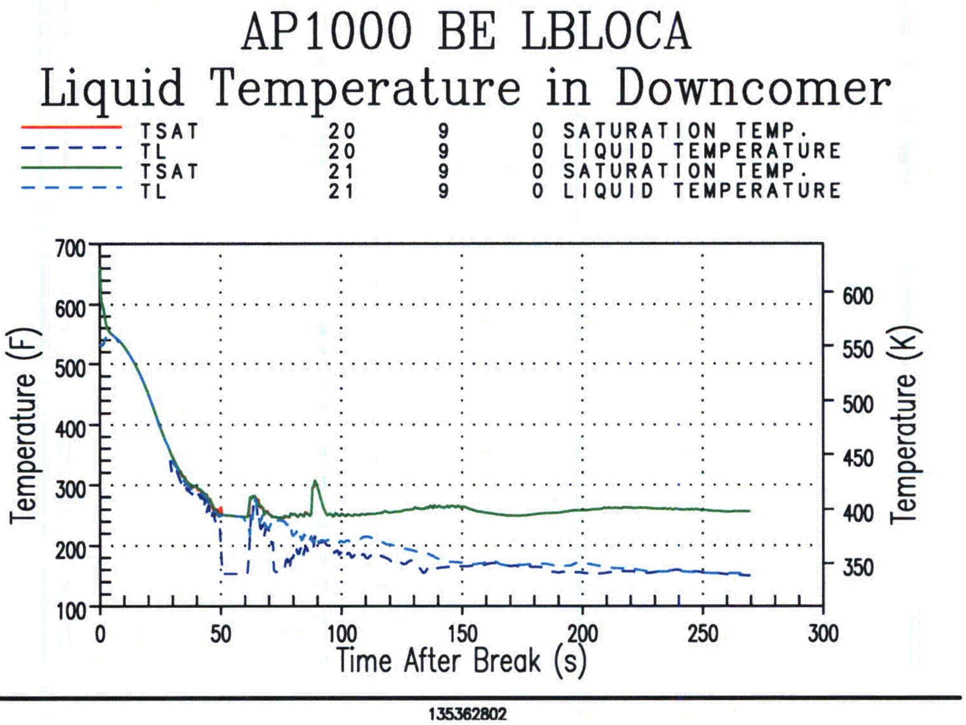


Figure 6-11
Liquid and Saturation Temperatures in mid-core downcomer elevations below and adjacent to broken cold leg

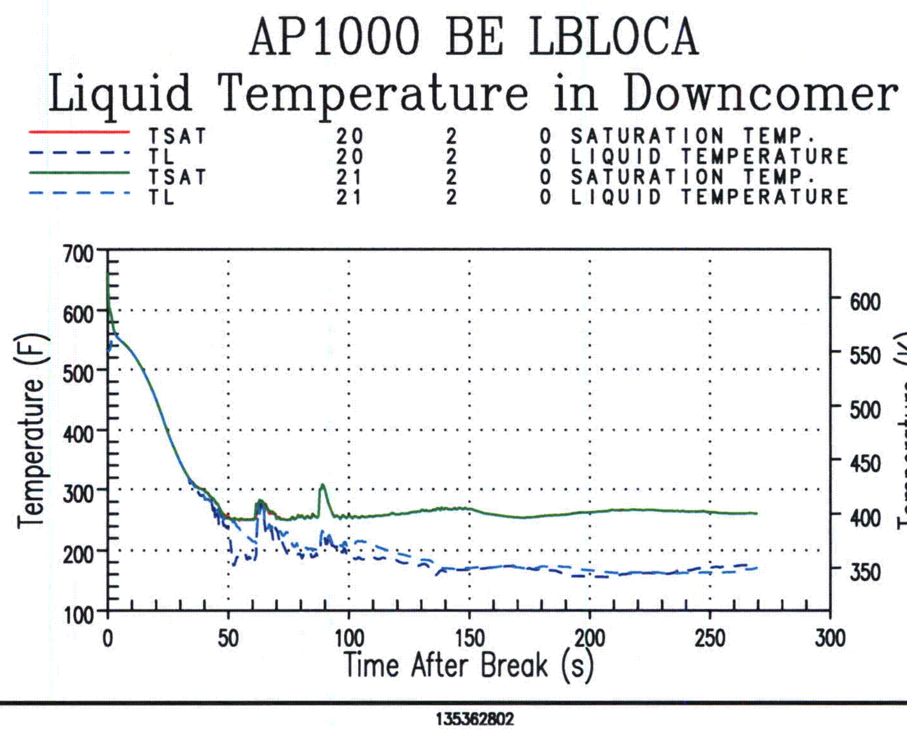


Figure 6-12
Liquid and Saturation Temperatures in low-core downcomer elevations below and adjacent to broken cold leg

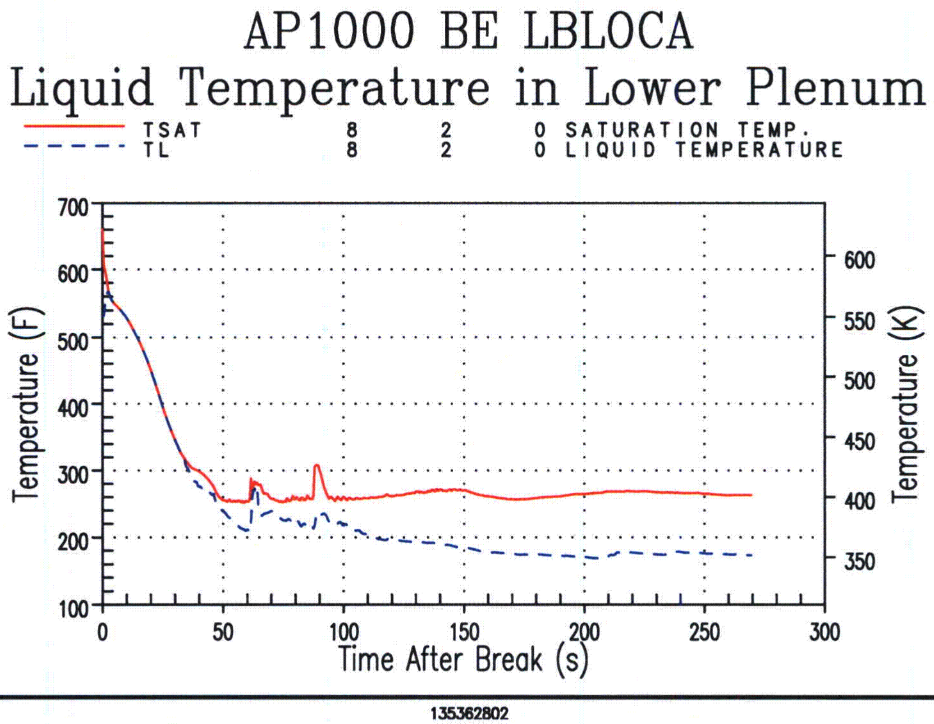


Figure 6-13
Liquid and Saturation Temperatures in the lower plenum

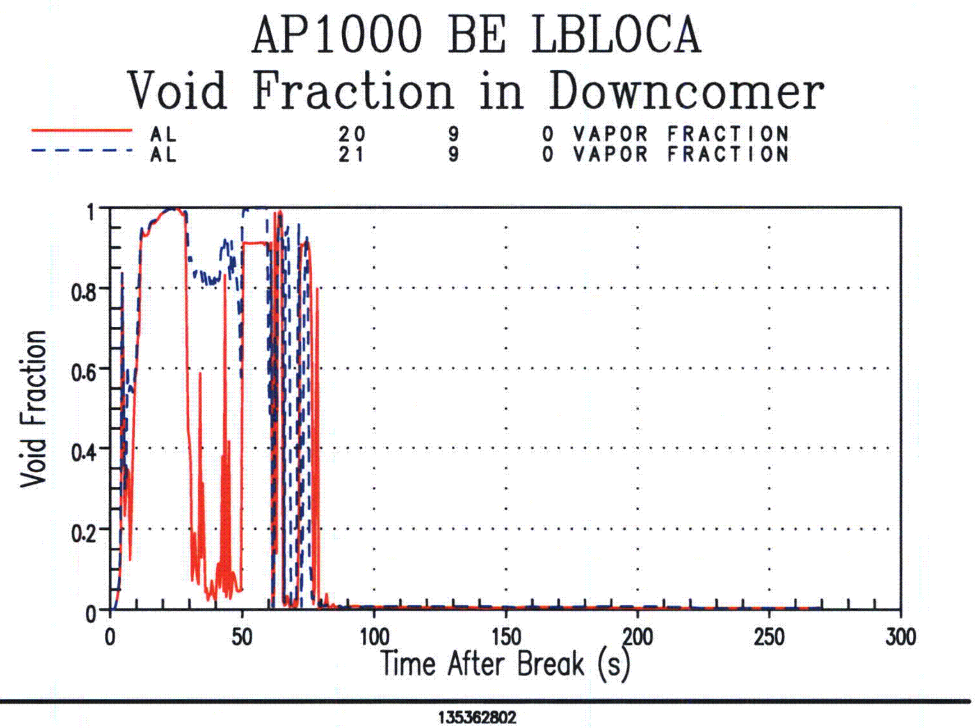


Figure 6-14
Void Fraction in mid-core downcomer elevations below and adjacent to break

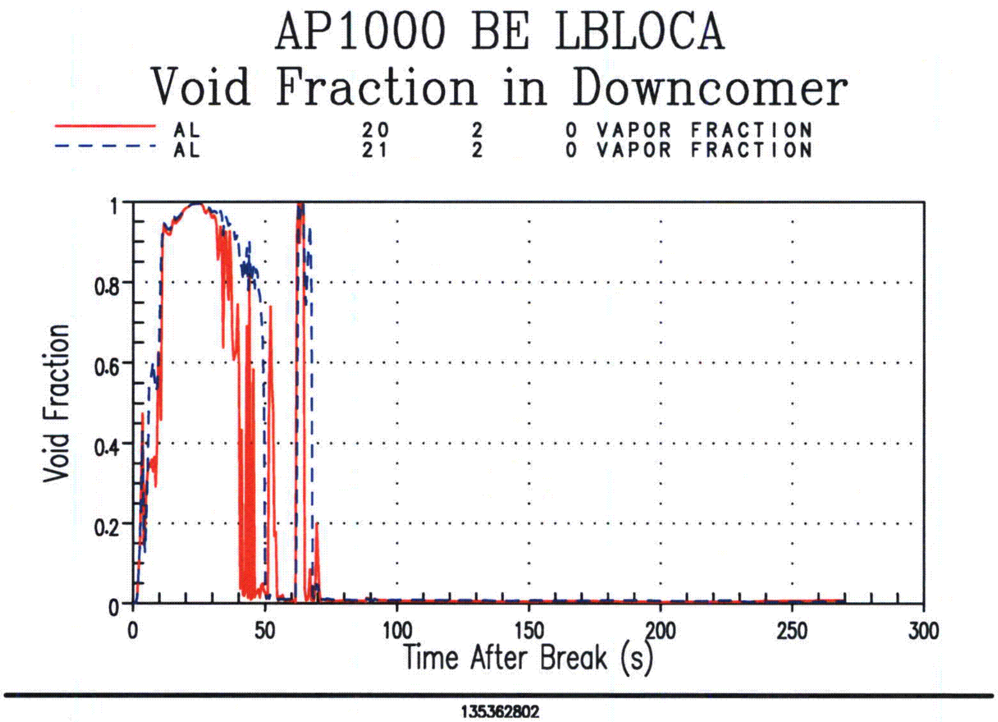
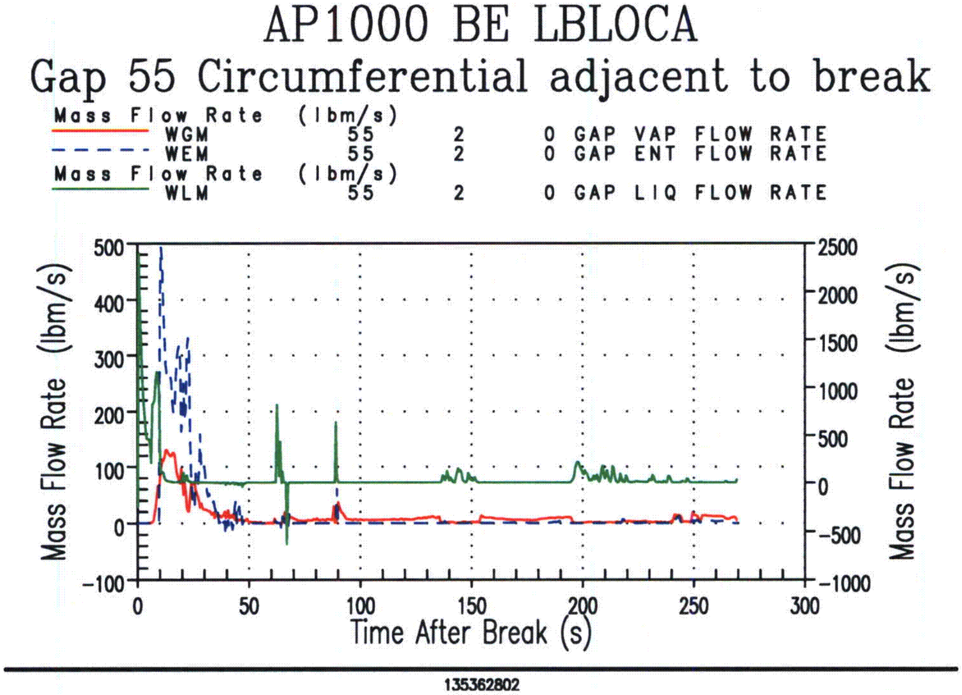


Figure 6-15
Void fraction in low-core downcomer elevation below and adjacent to break



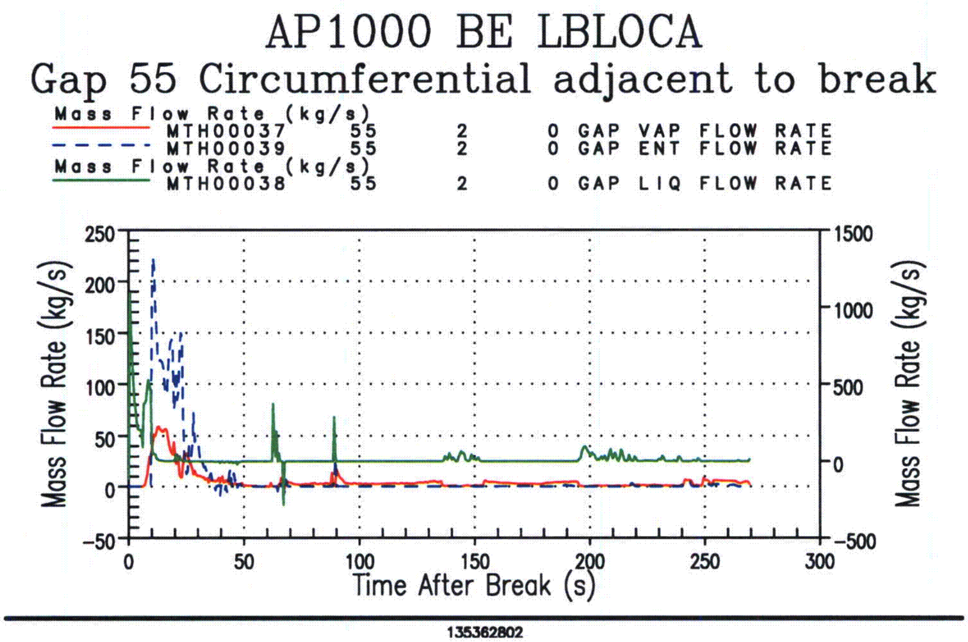


Figure 6-16
Circumferential Flow from the downcomer channel immediately above the DVI to the broken cold leg nozzle channel

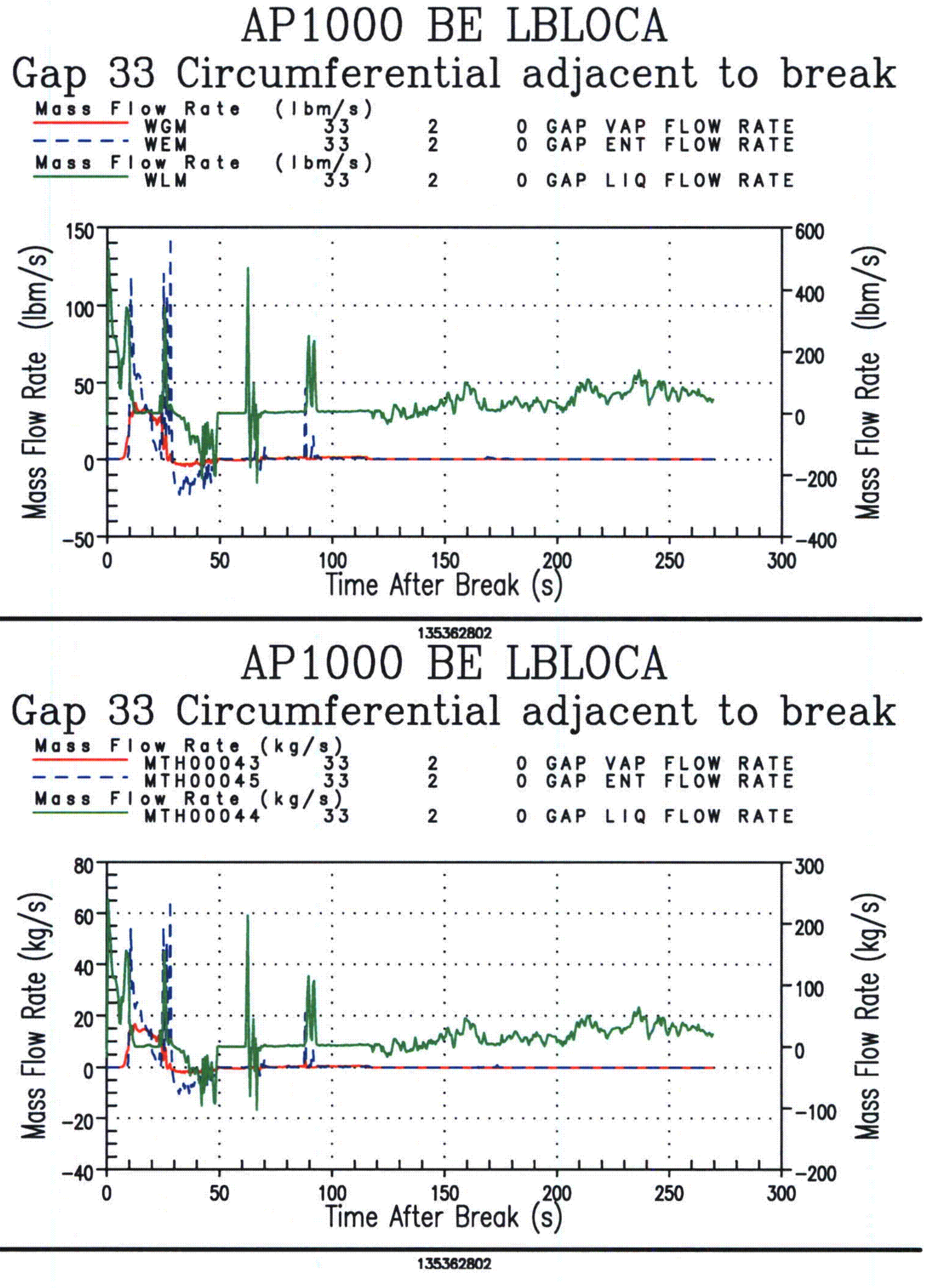
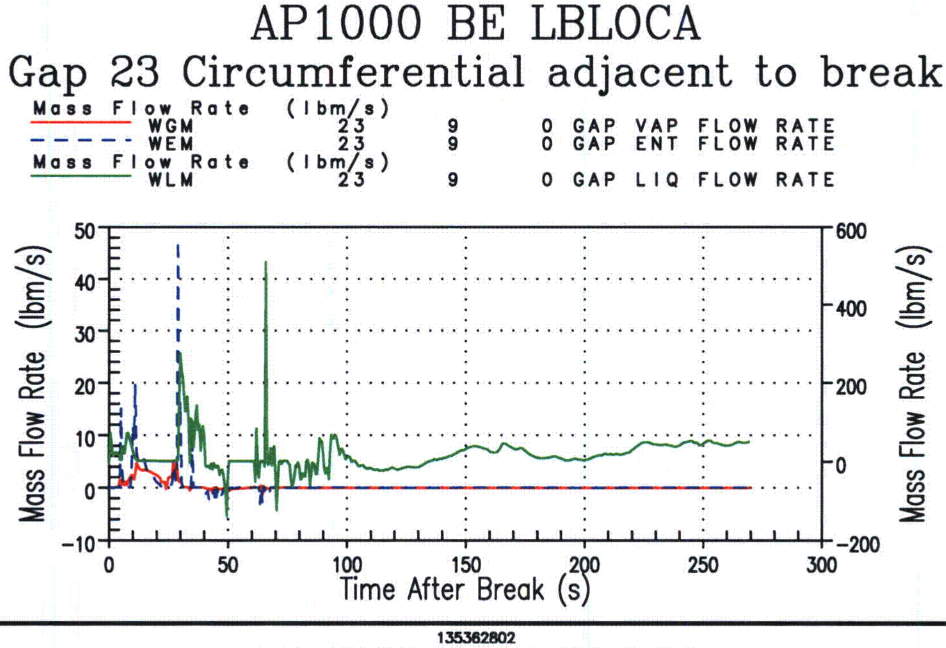


Figure 6-17
Circumferential flow from the downcomer channel below the DVI to the broken cold leg nozzle channel at the top fuel nozzle elevation (CCFL region)



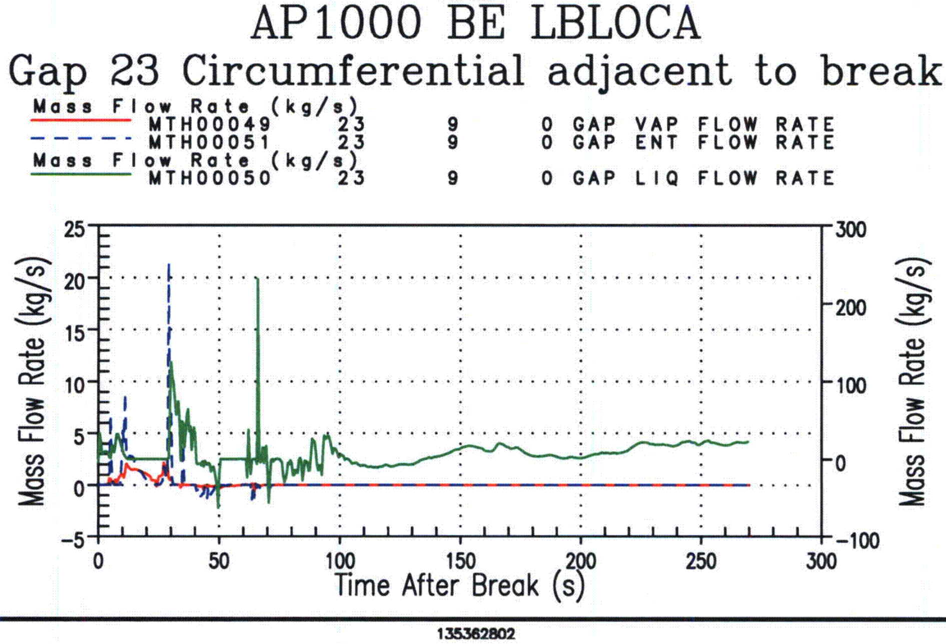
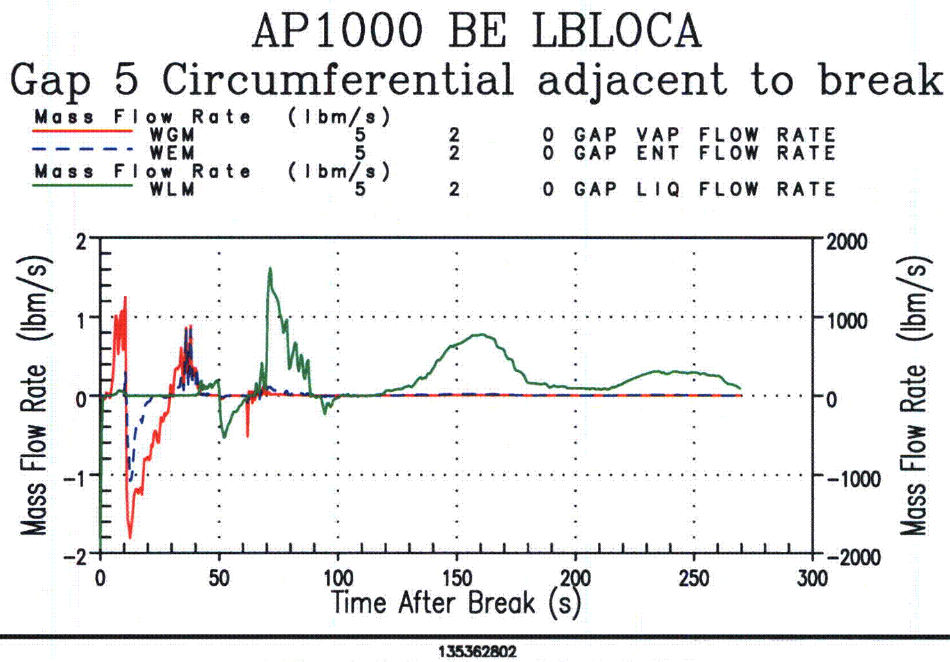


Figure 6-18
Circumferential flow from the downcomer channel below the DVI to the broken cold leg nozzle channel at the mid-core elevation



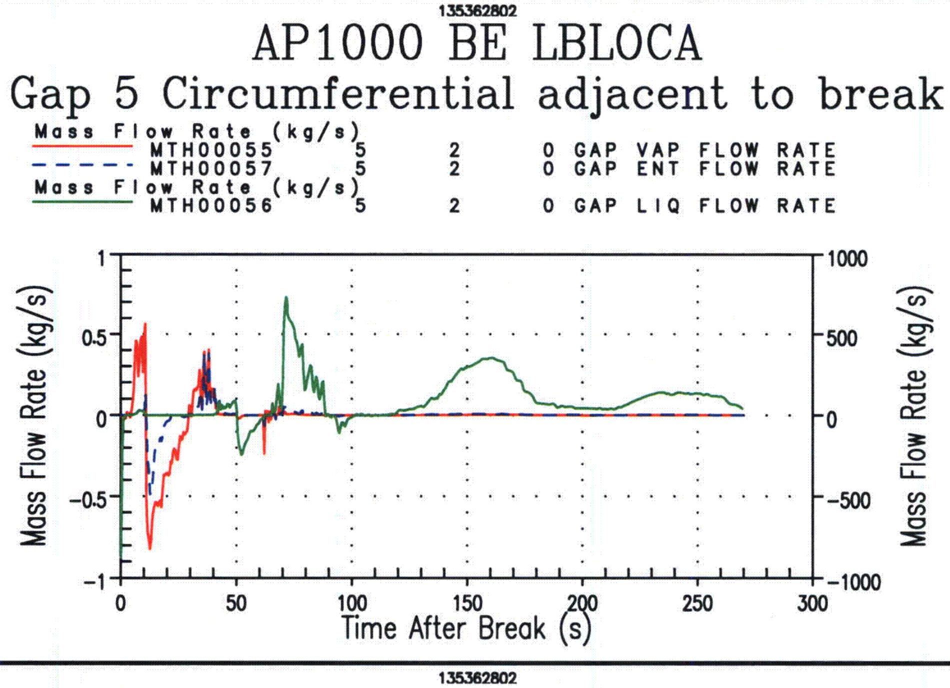
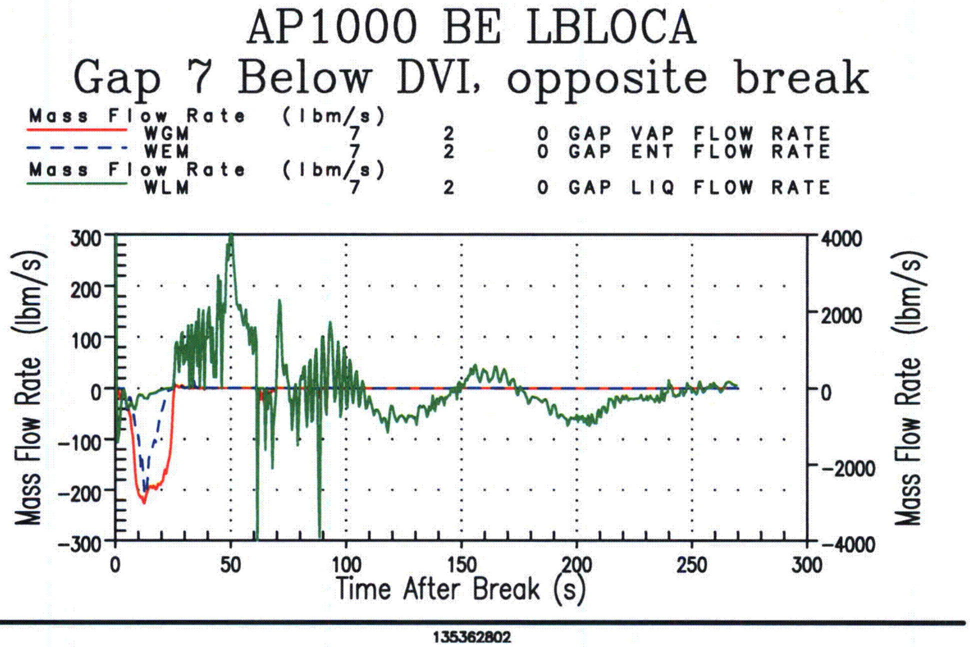


Figure 6-19
Circumferential flow from the downcomer channel below the DVI to the broken cold leg nozzle channel in the lower plenum



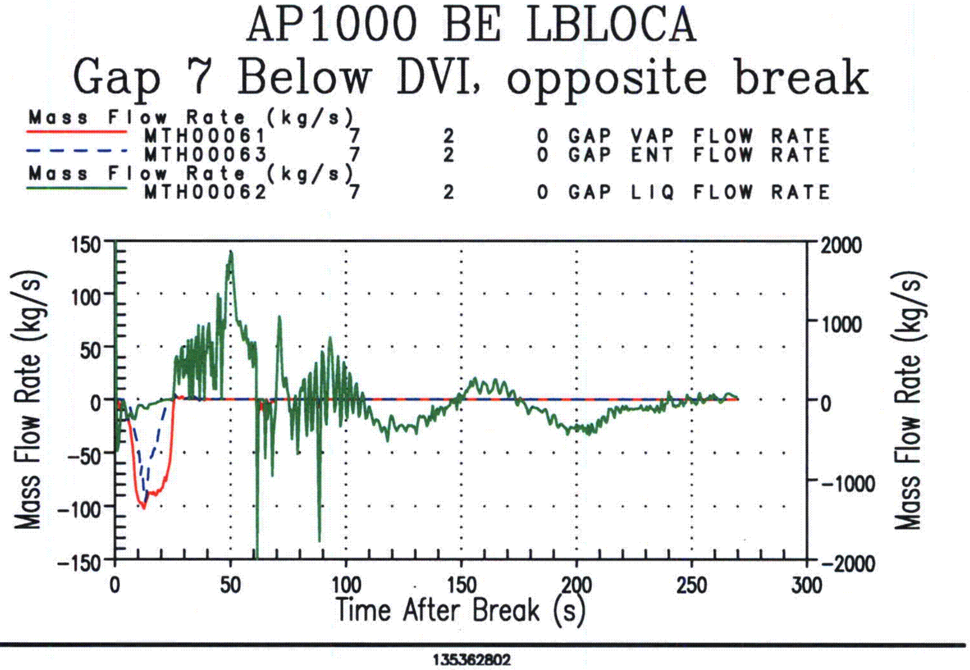
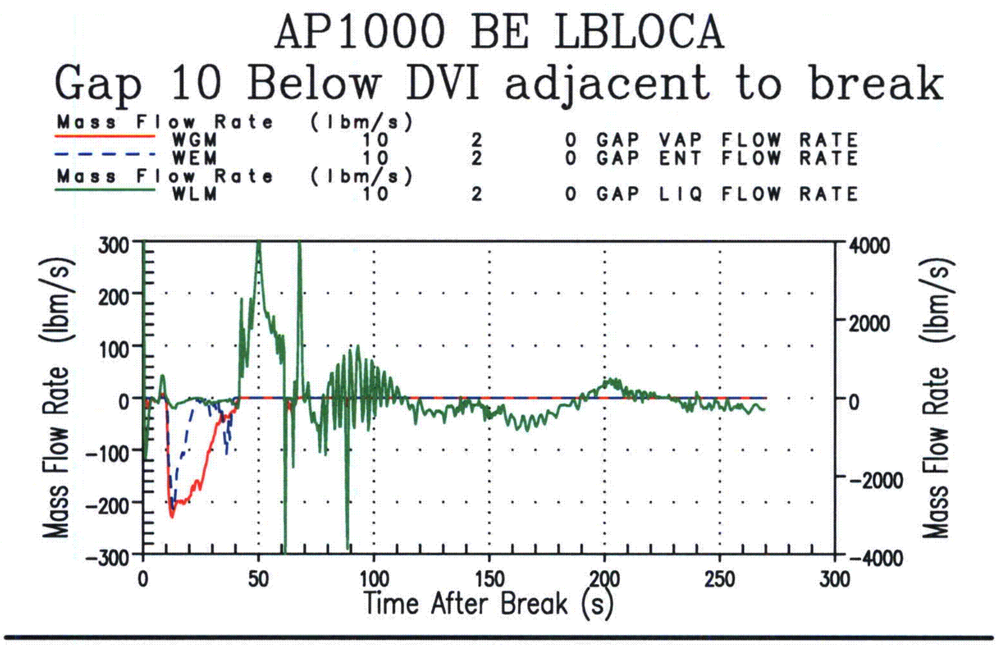


Figure 6-20
Vapor, Entrained Liquid, and Liquid flow into the lower plenum from the region below the DVI opposite the break



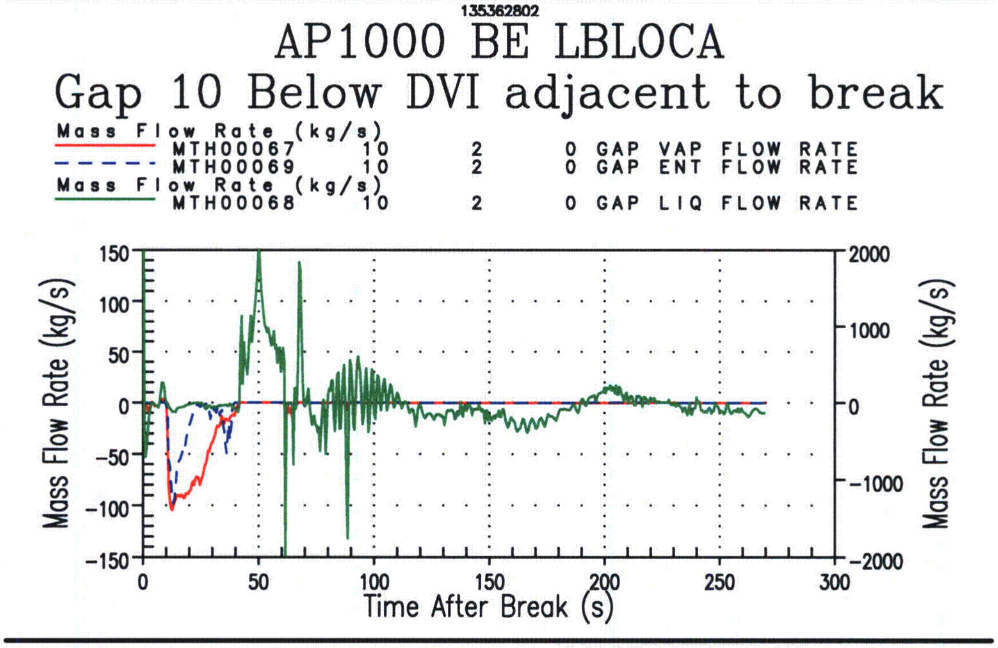
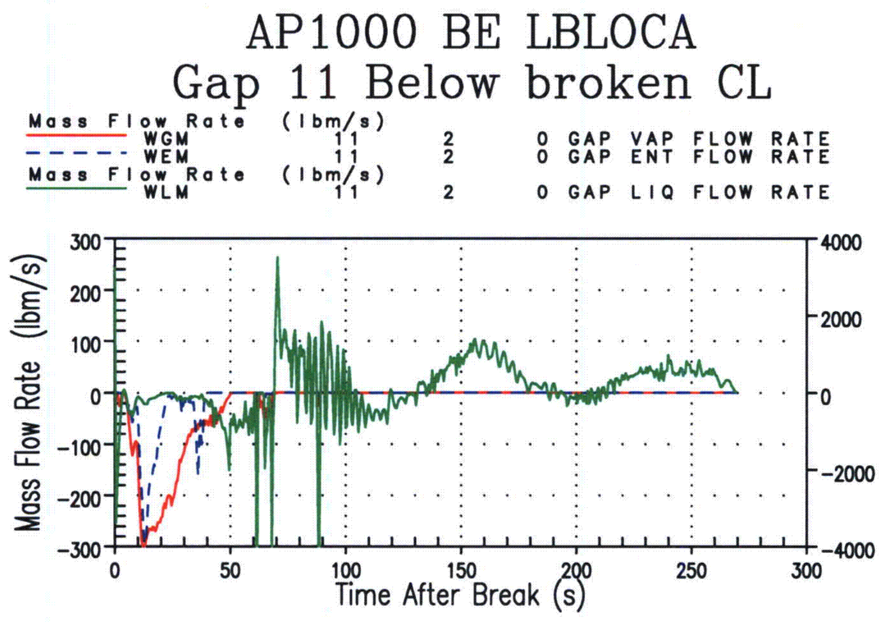


Figure 6-21
Vapor, Entrained Liquid, and Liquid flow into the lower plenum from the region below the DVI adjacent to the break



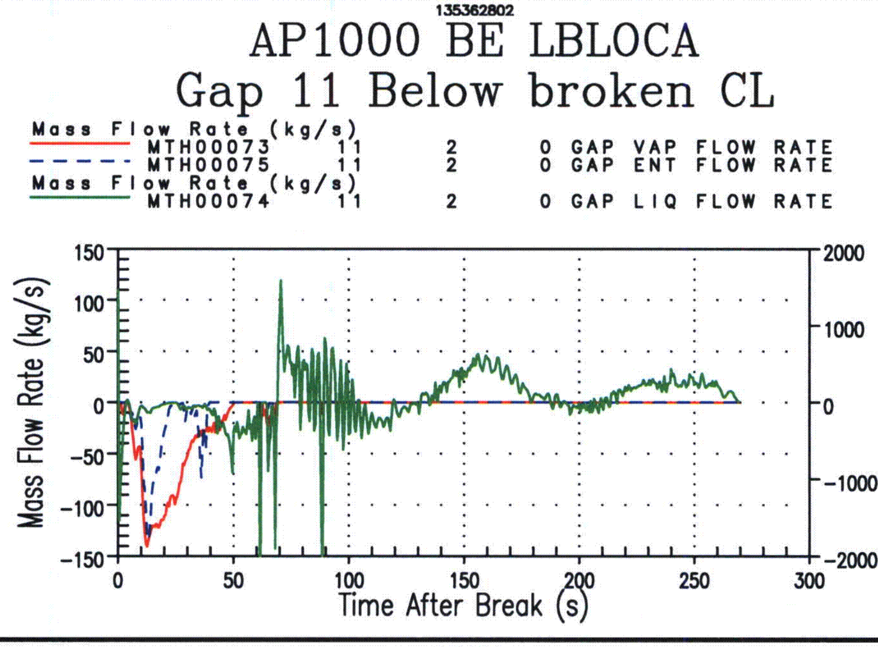
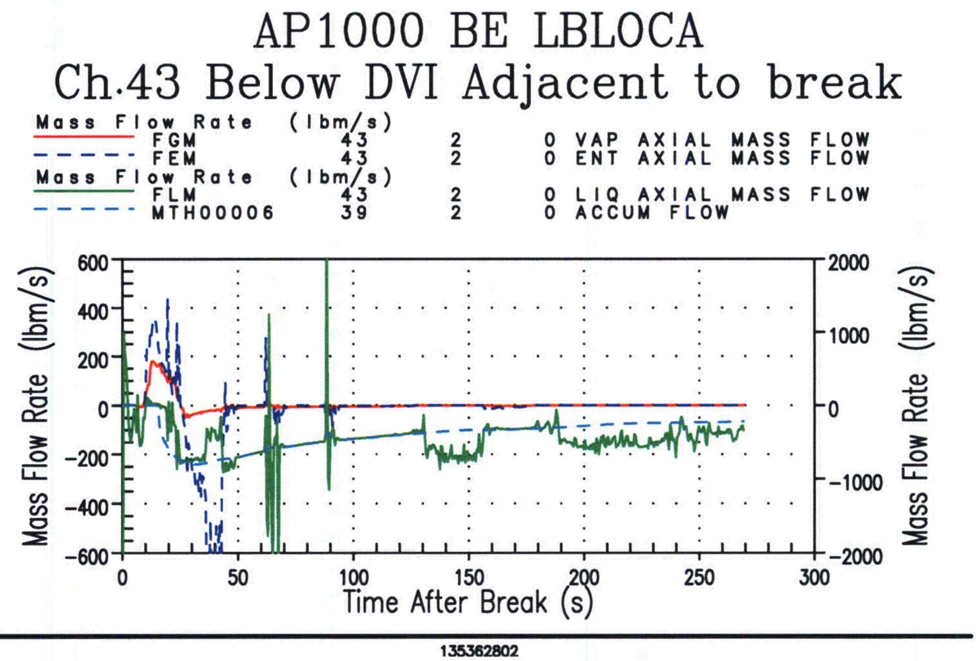


Figure 6-22
Vapor, Entrained Liquid, and Liquid into the lower plenum from the region below the broken cold leg



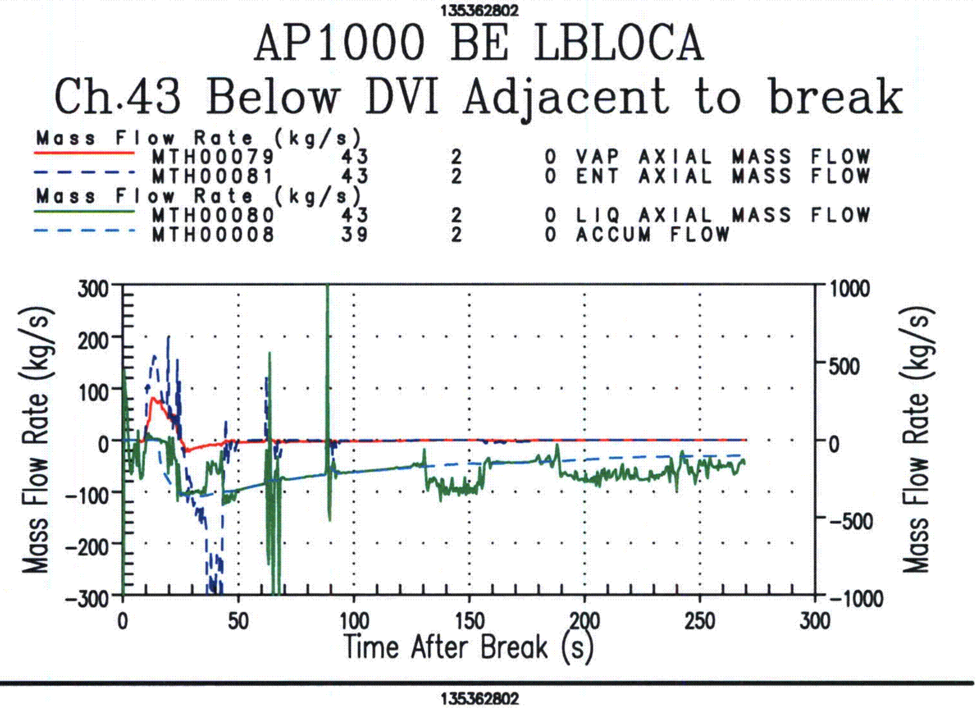


Figure 6-23
Vertical flow in the downcomer immediately below the DVI adjacent to the break

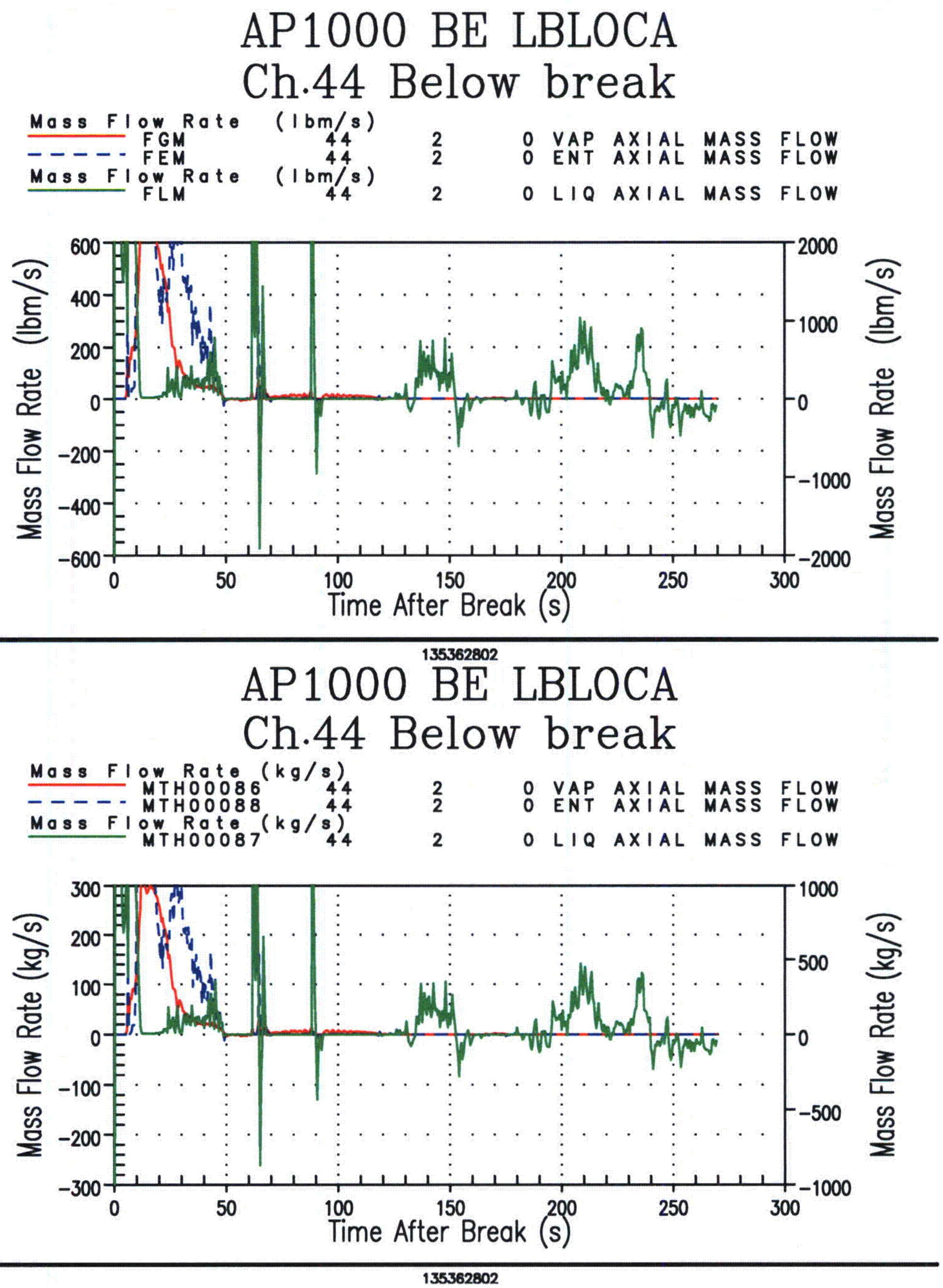


Figure 6-24
Vertical flow immediately below the broken cold leg

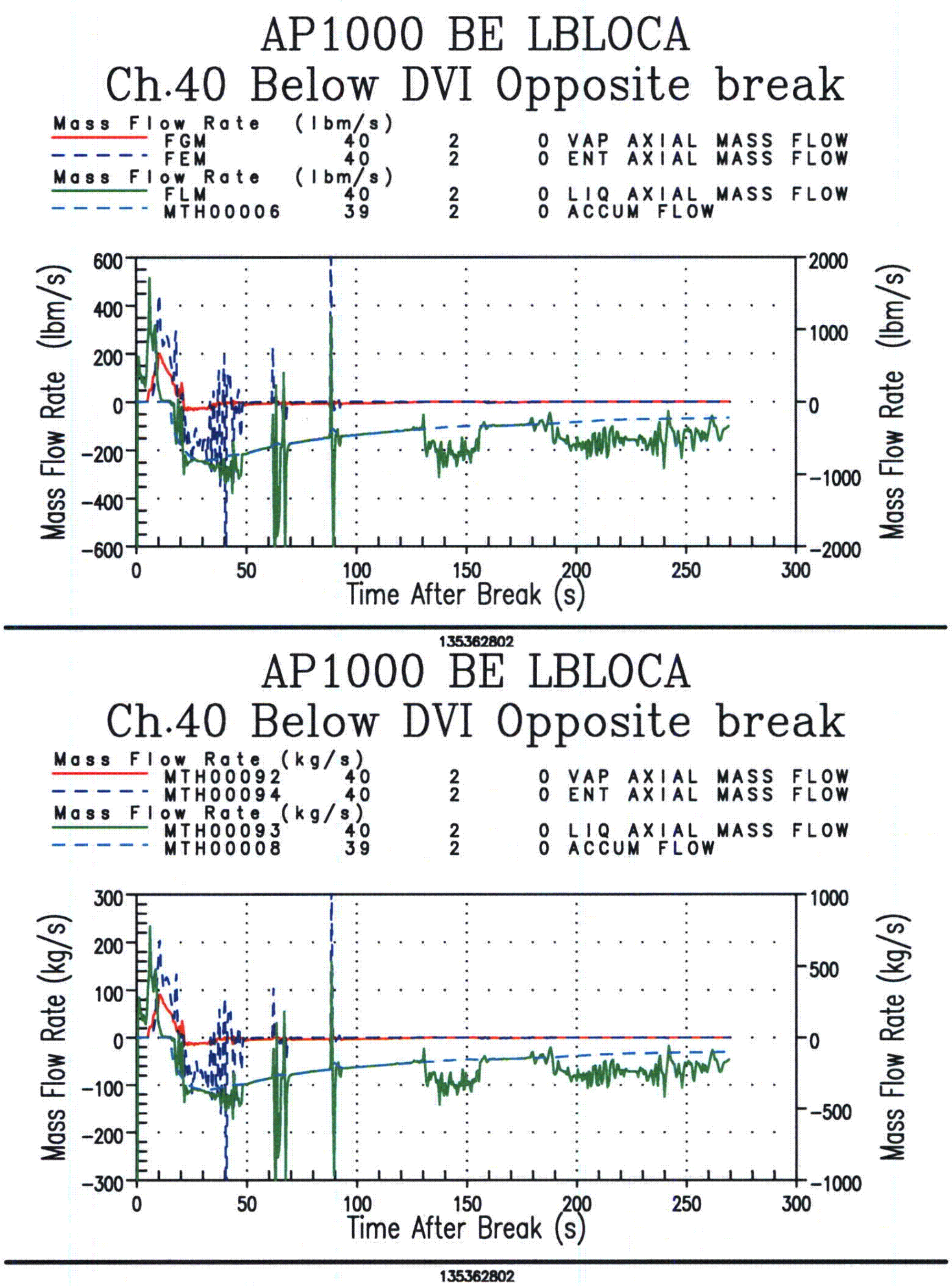


Figure 6-25
Vertical flow immediately below the DVI opposite the break

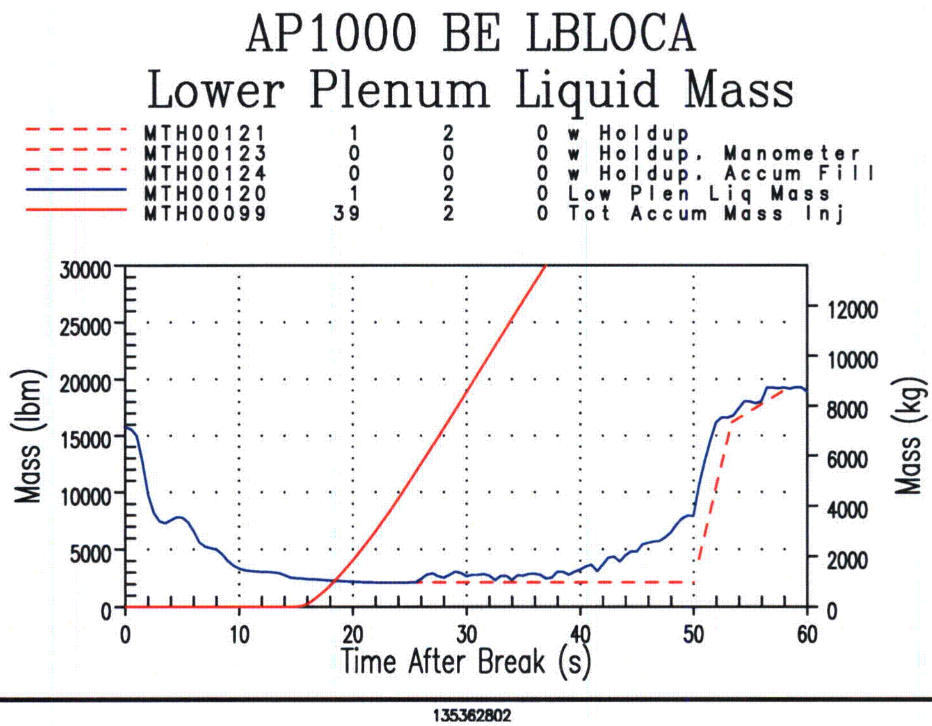


Figure 6-26
Lower Plenum Liquid Mass as a result of postulated complete liquid holdup at the flow skirt

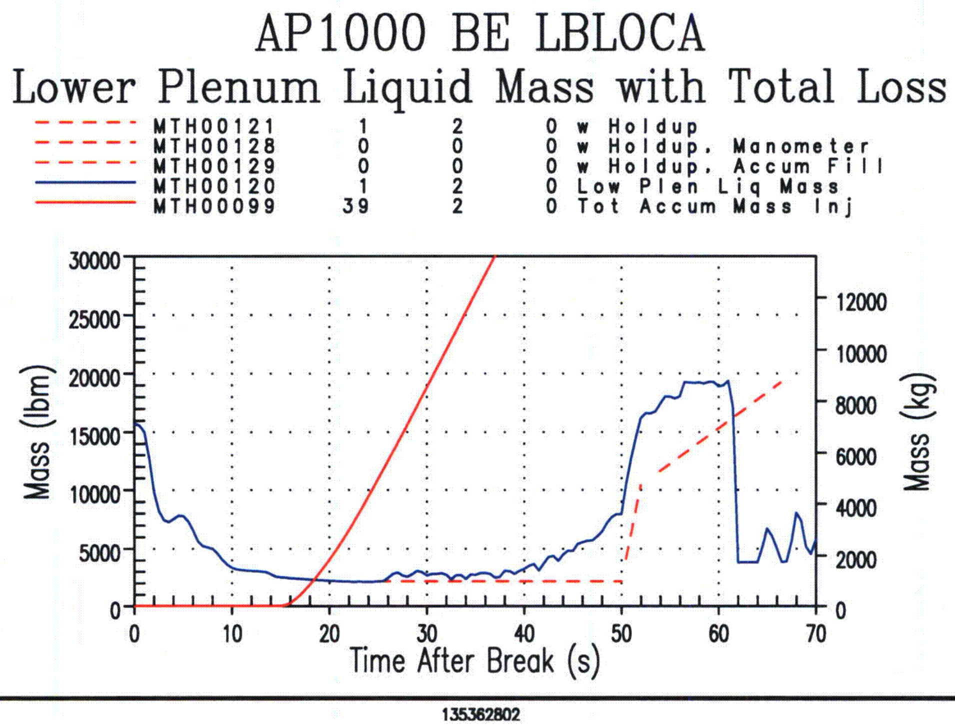


Figure 6-27
Lower Plenum Liquid Mass as a result of postulated complete liquid holdup and sweep out at the flow skirt

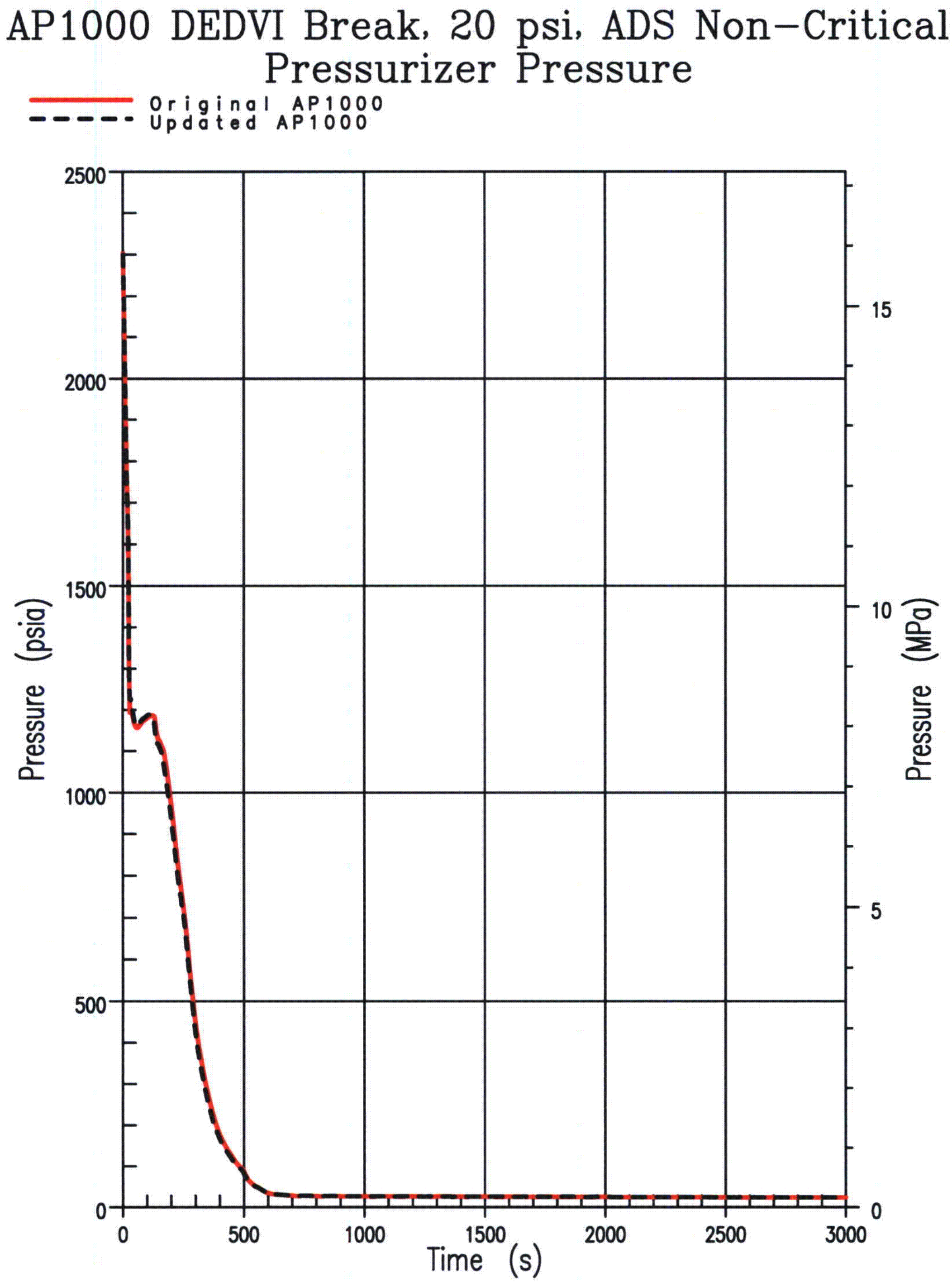


Figure 6-28
DEDVI Break, RCS Pressure

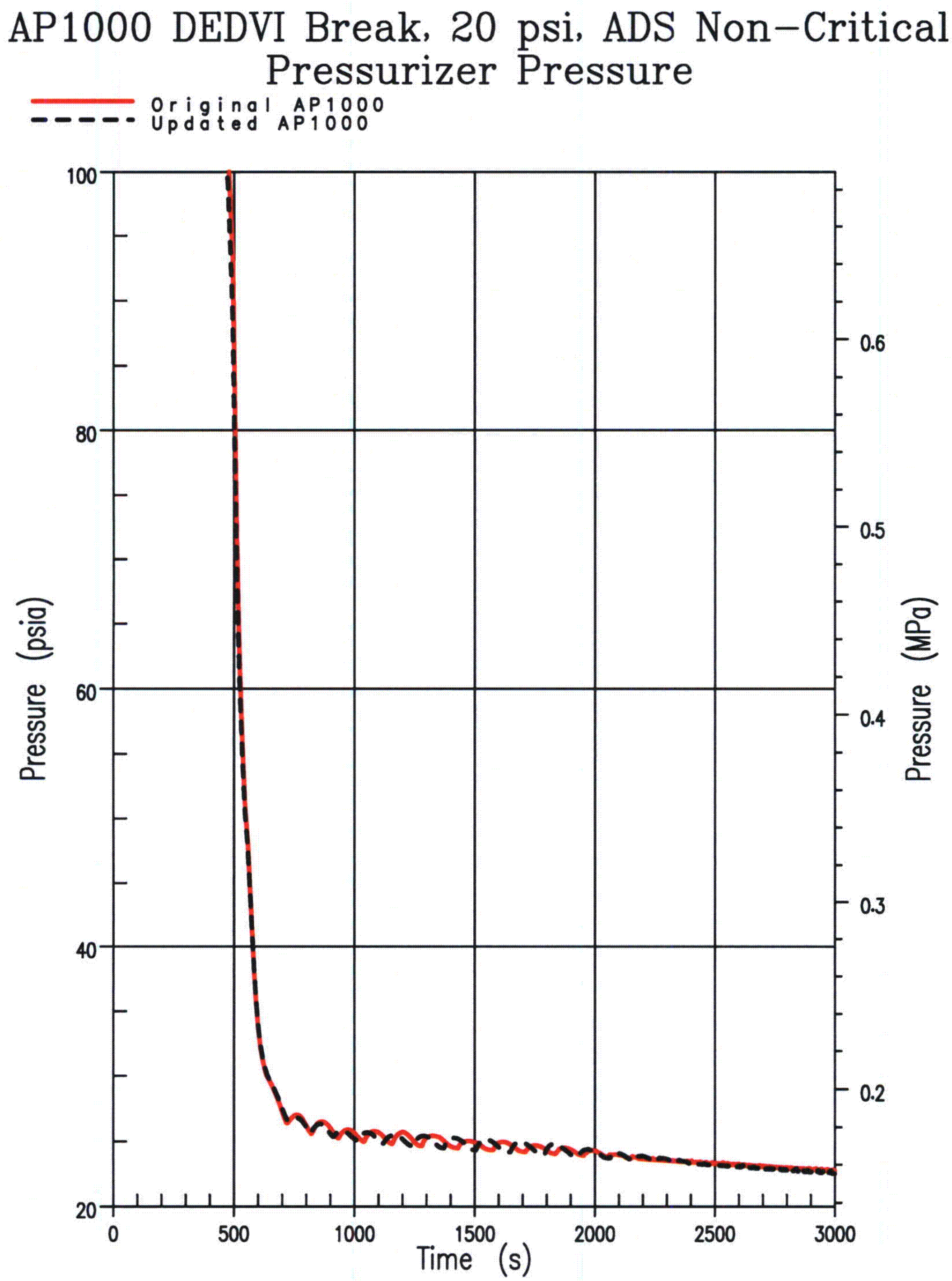


Figure 6-29
DEDVI, RCS Pressure, Low Pressure Region

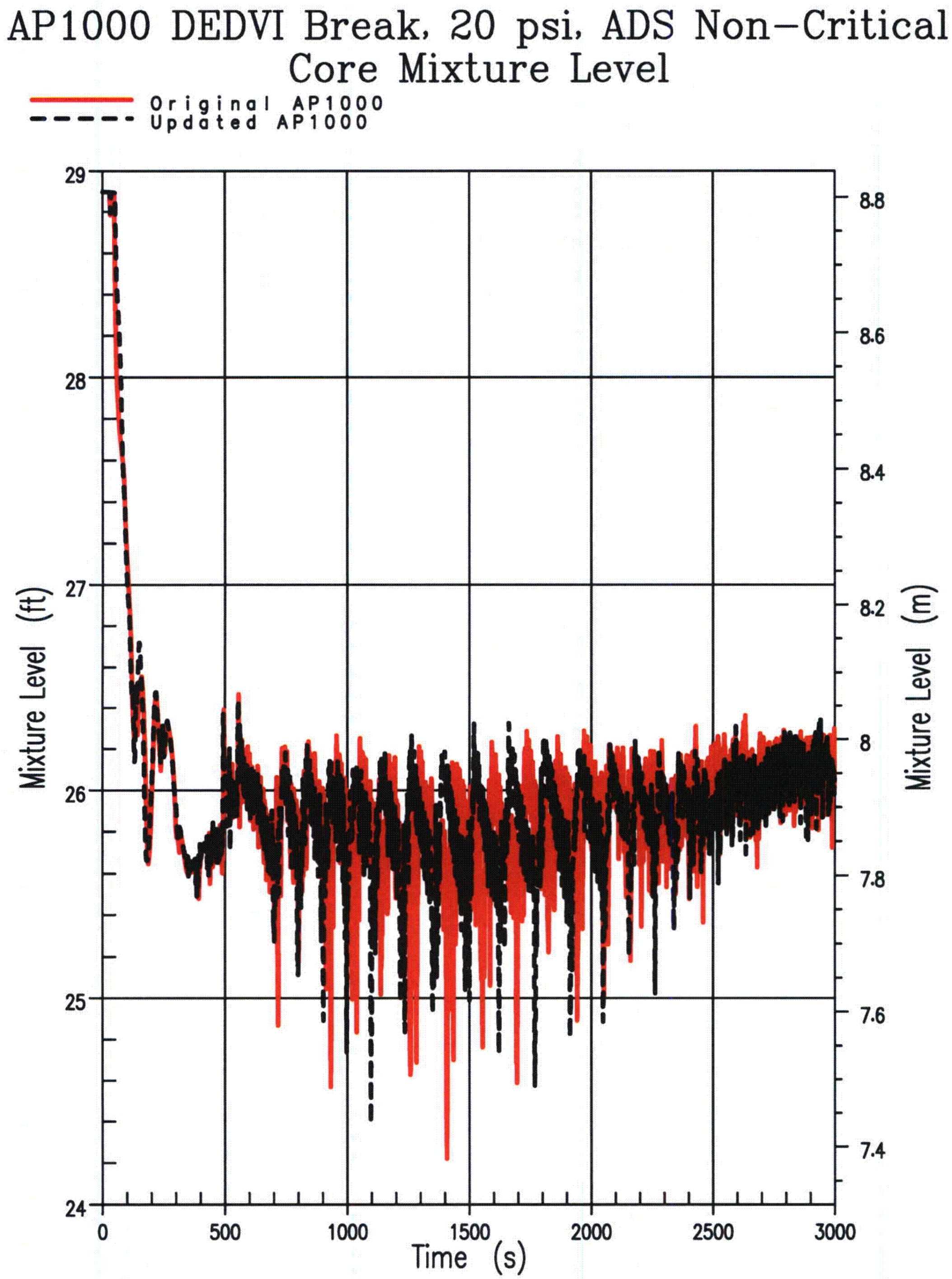


Figure 6-30
DEDVI, Core/Upper Plenum Mixture Level

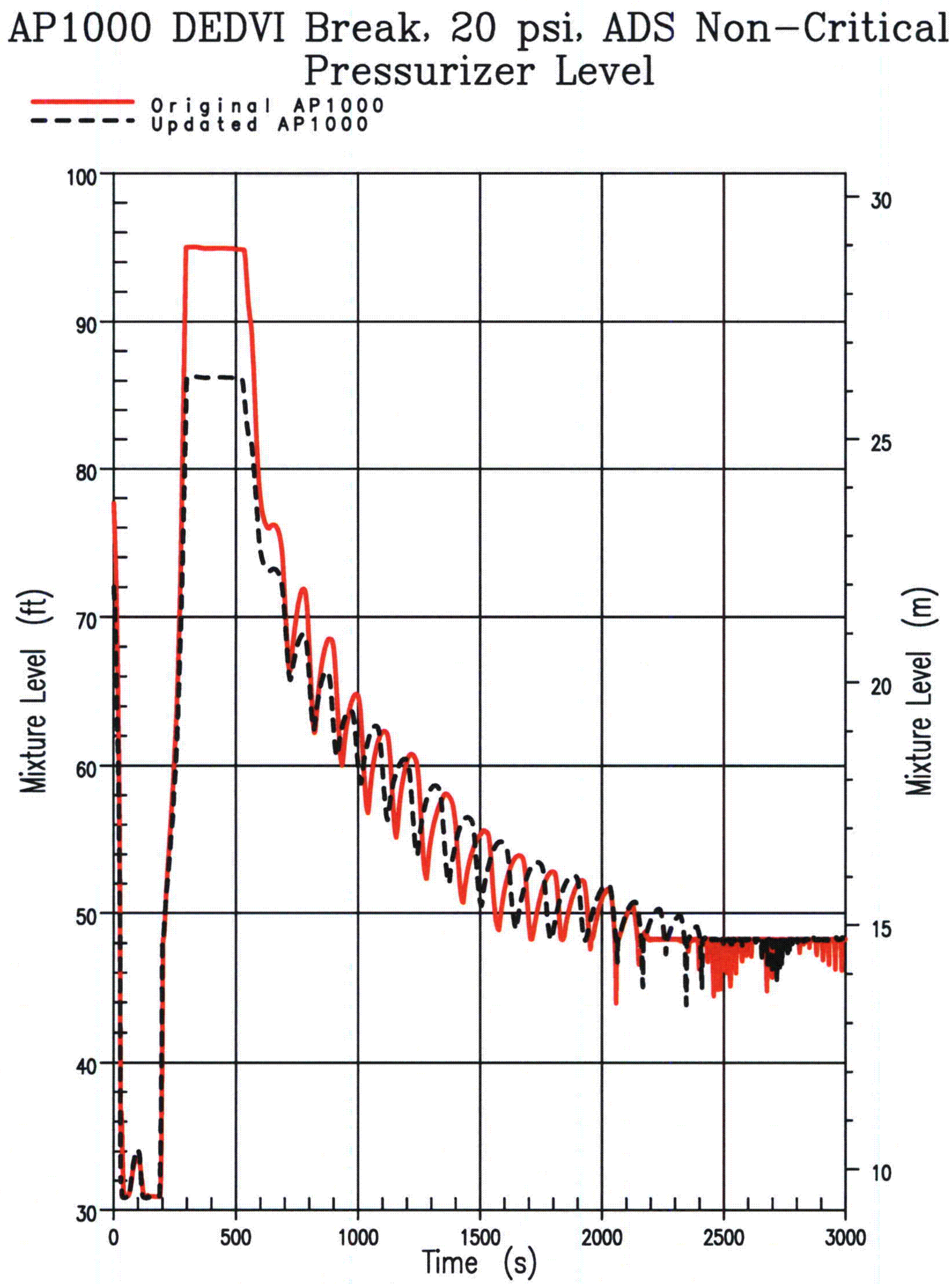


Figure 6-31
DEDVI, Pressurizer Mixture Level

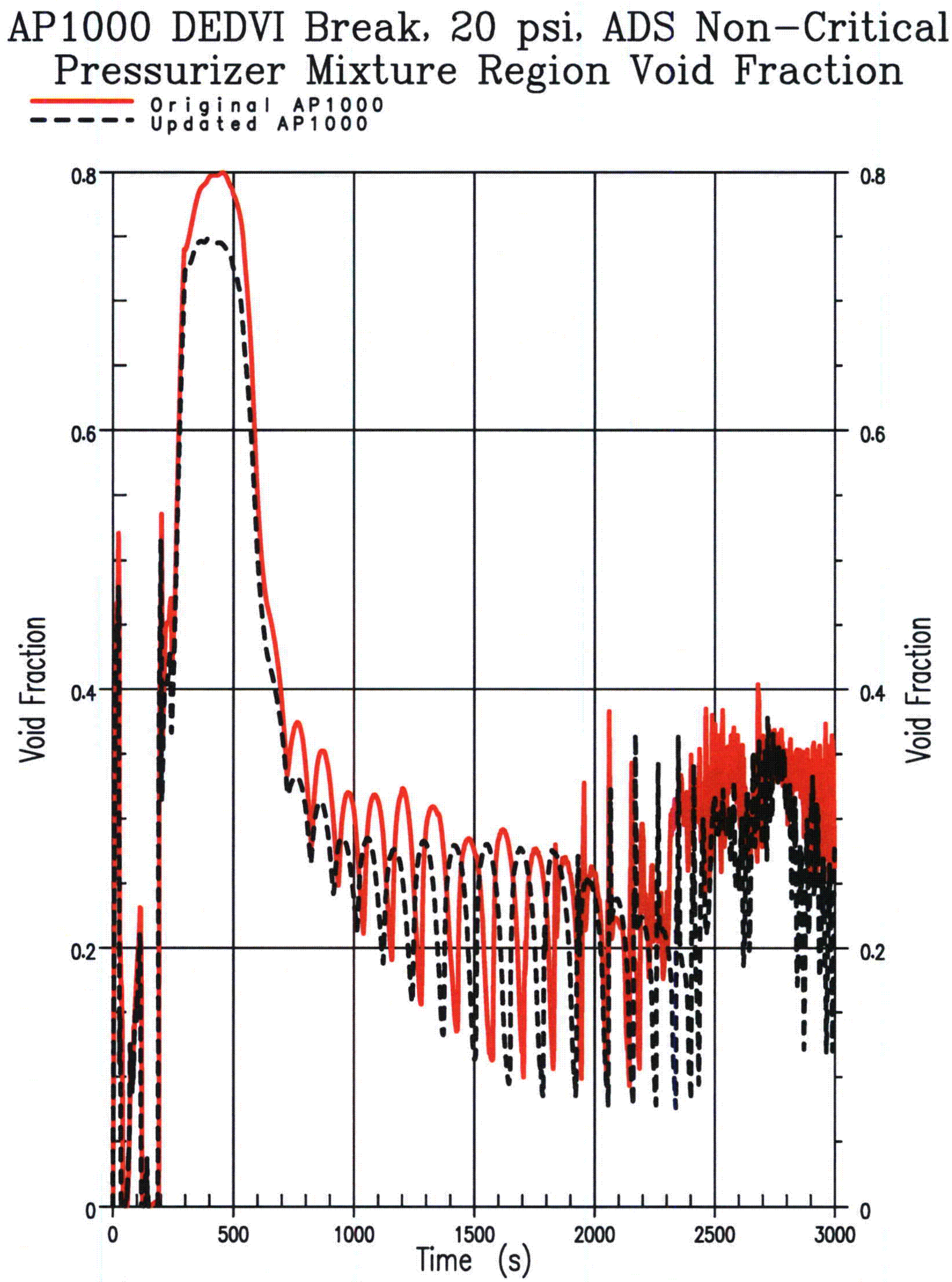


Figure 6-32
DEDVI, Pressurizer Mixture Region Void Fraction

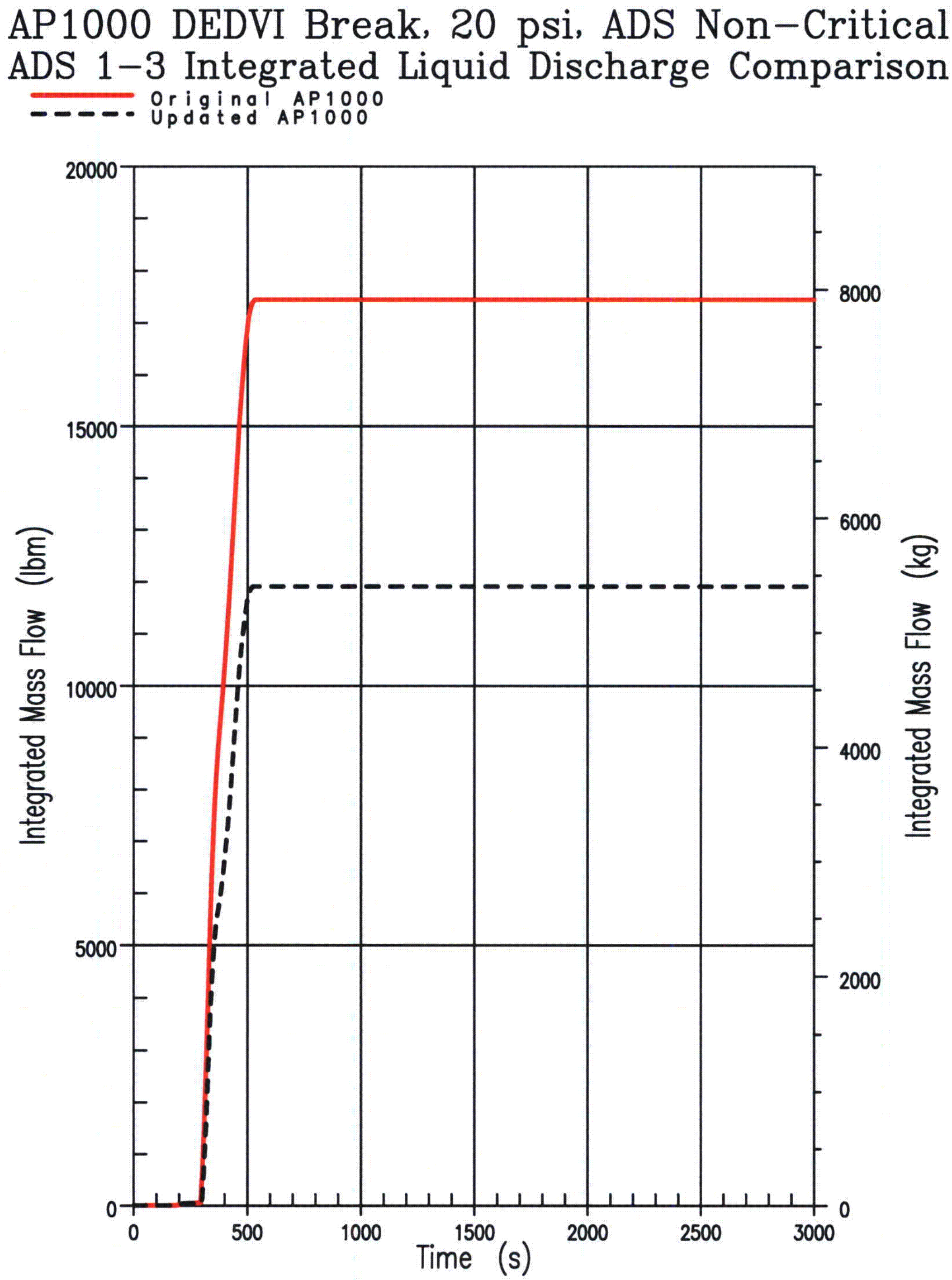


Figure 6-33
DEDVI, ADS 1-3 Integrated Liquid Discharge

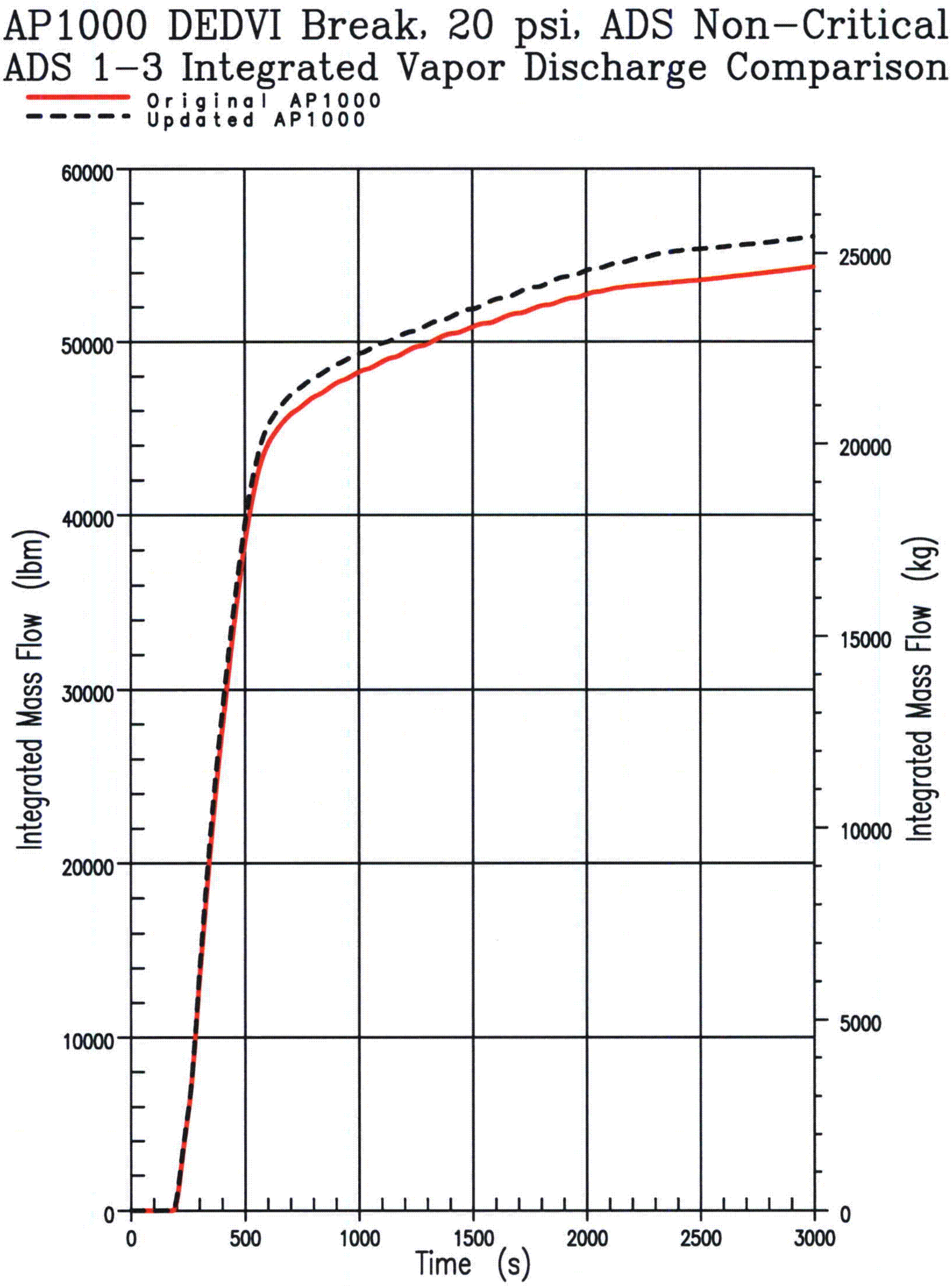


Figure 6-34
DEDVI, ADS 1-3 Integrated Vapor Discharge

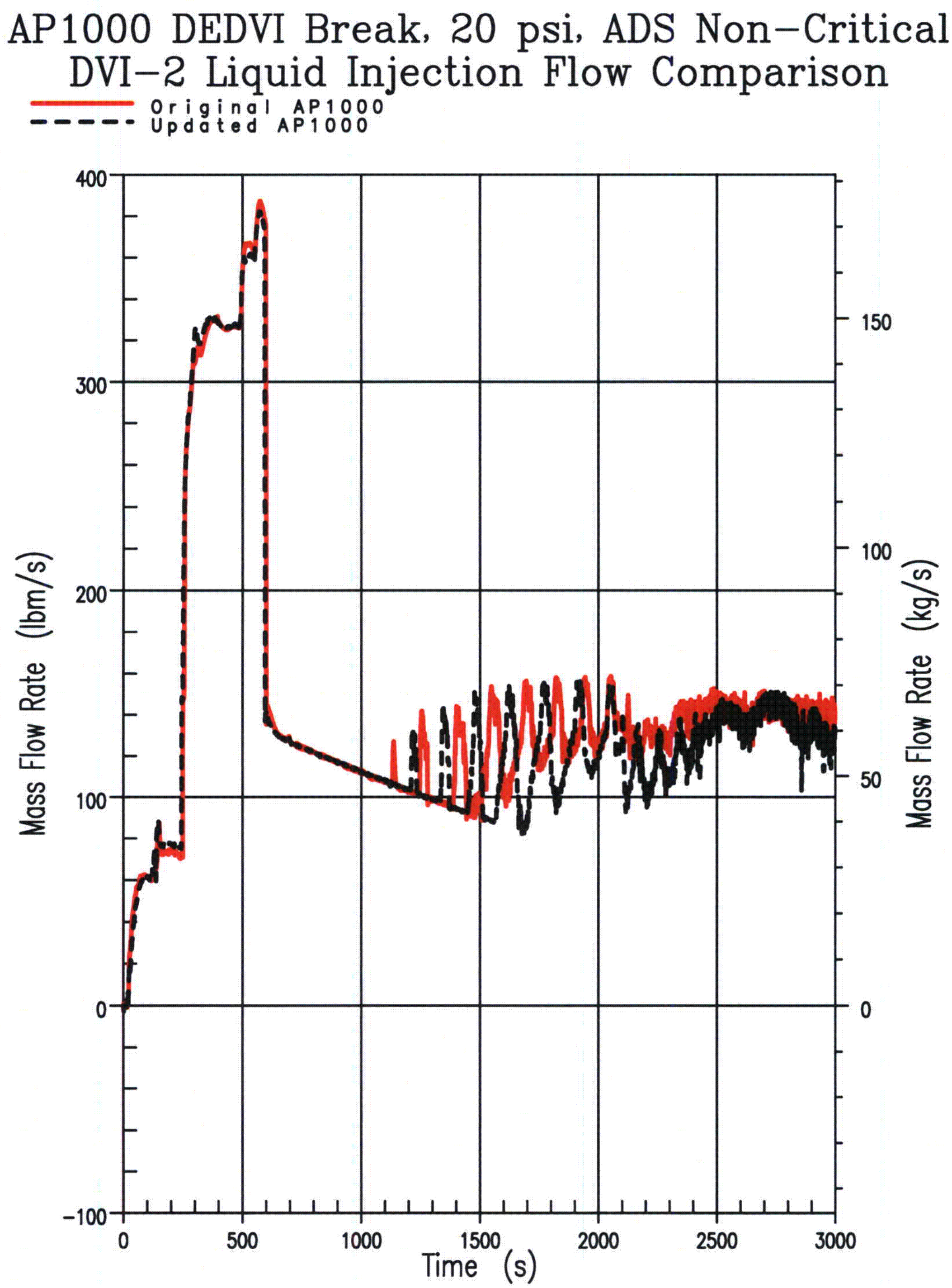


Figure 6-35
DEDVI, Intact DVI Injection Flow

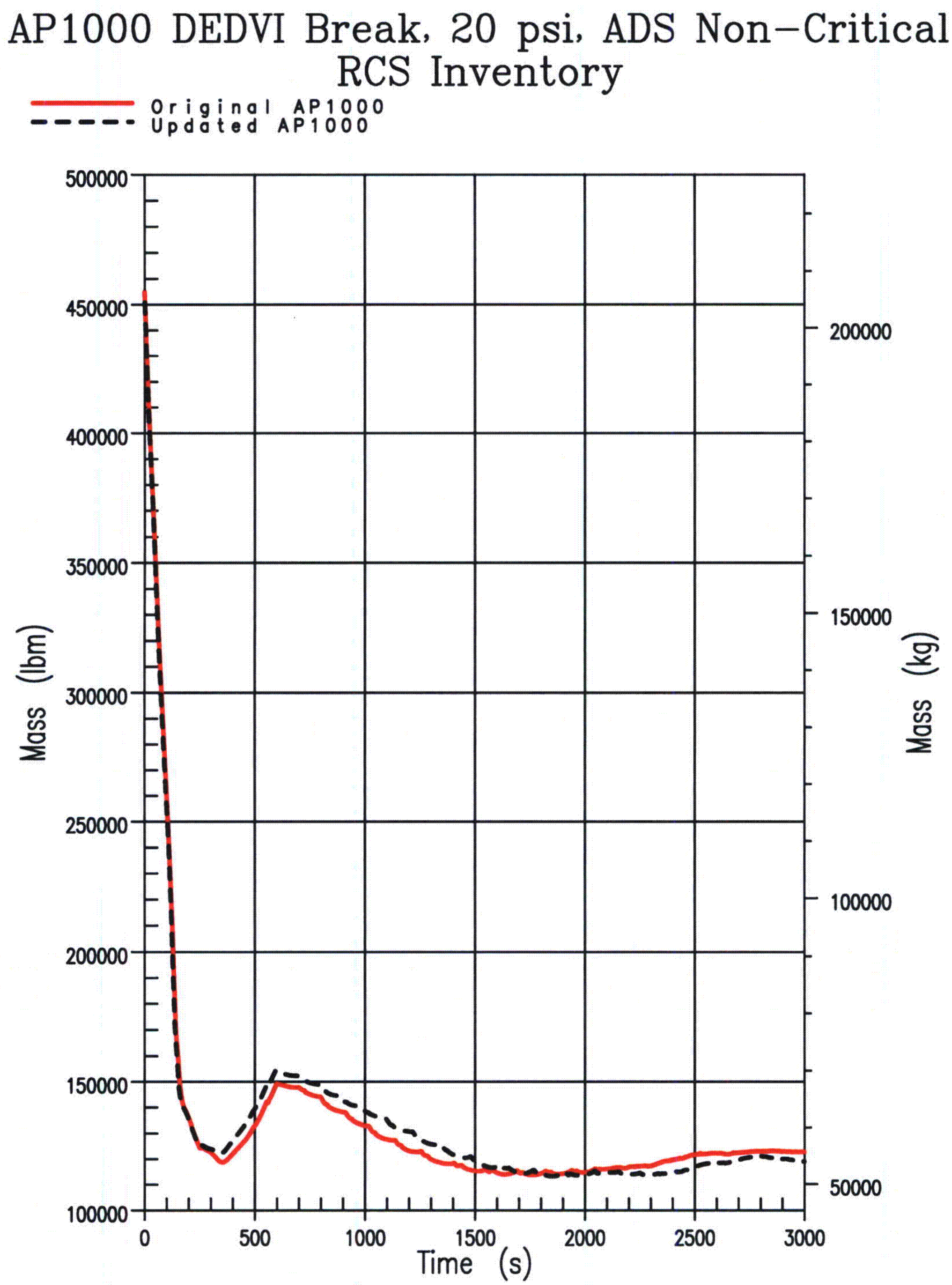


Figure 6-36
DEDVI, RCS Inventory

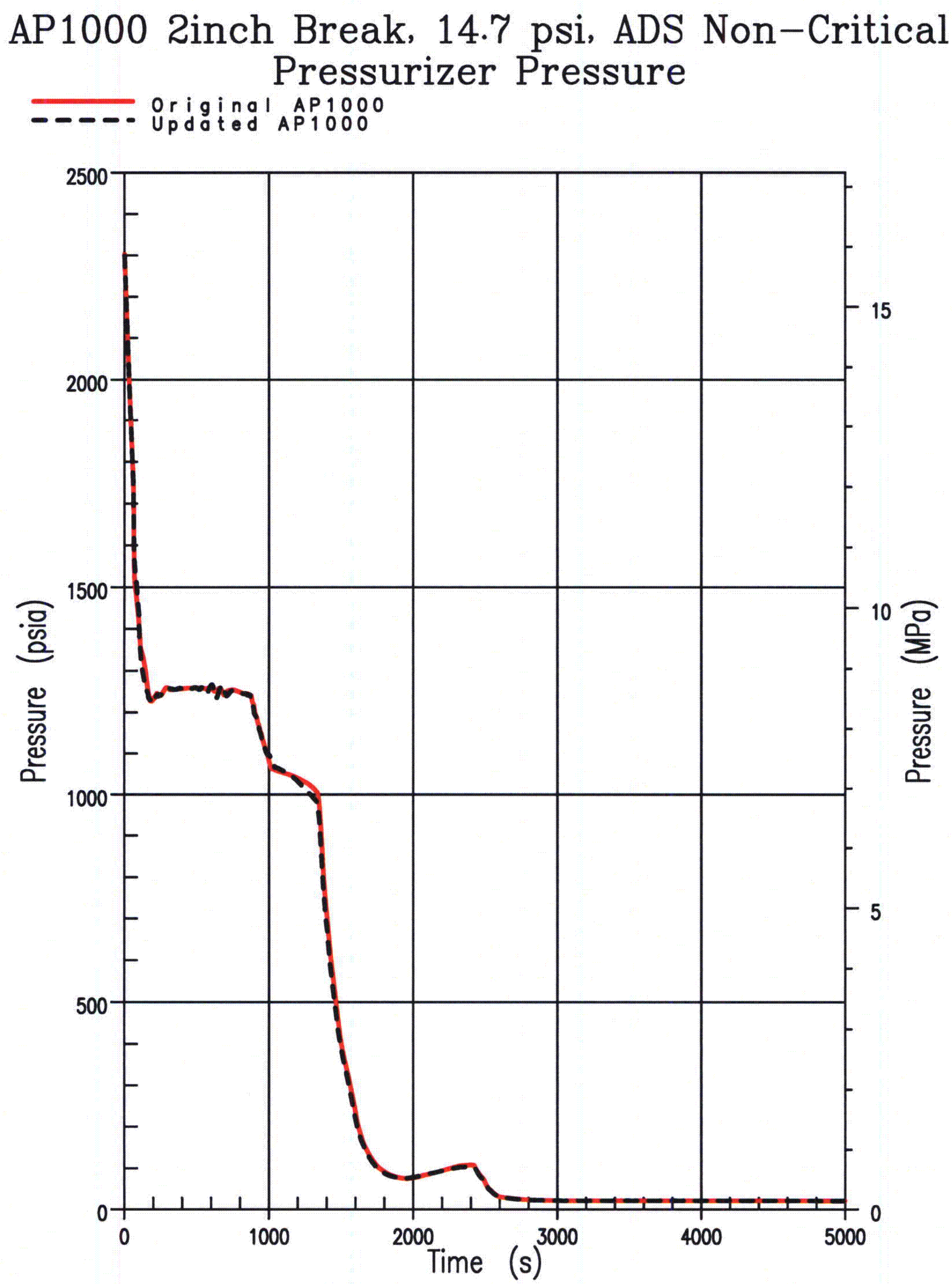


Figure 6-37
2-Inch Cold Leg Break, RCS Pressure

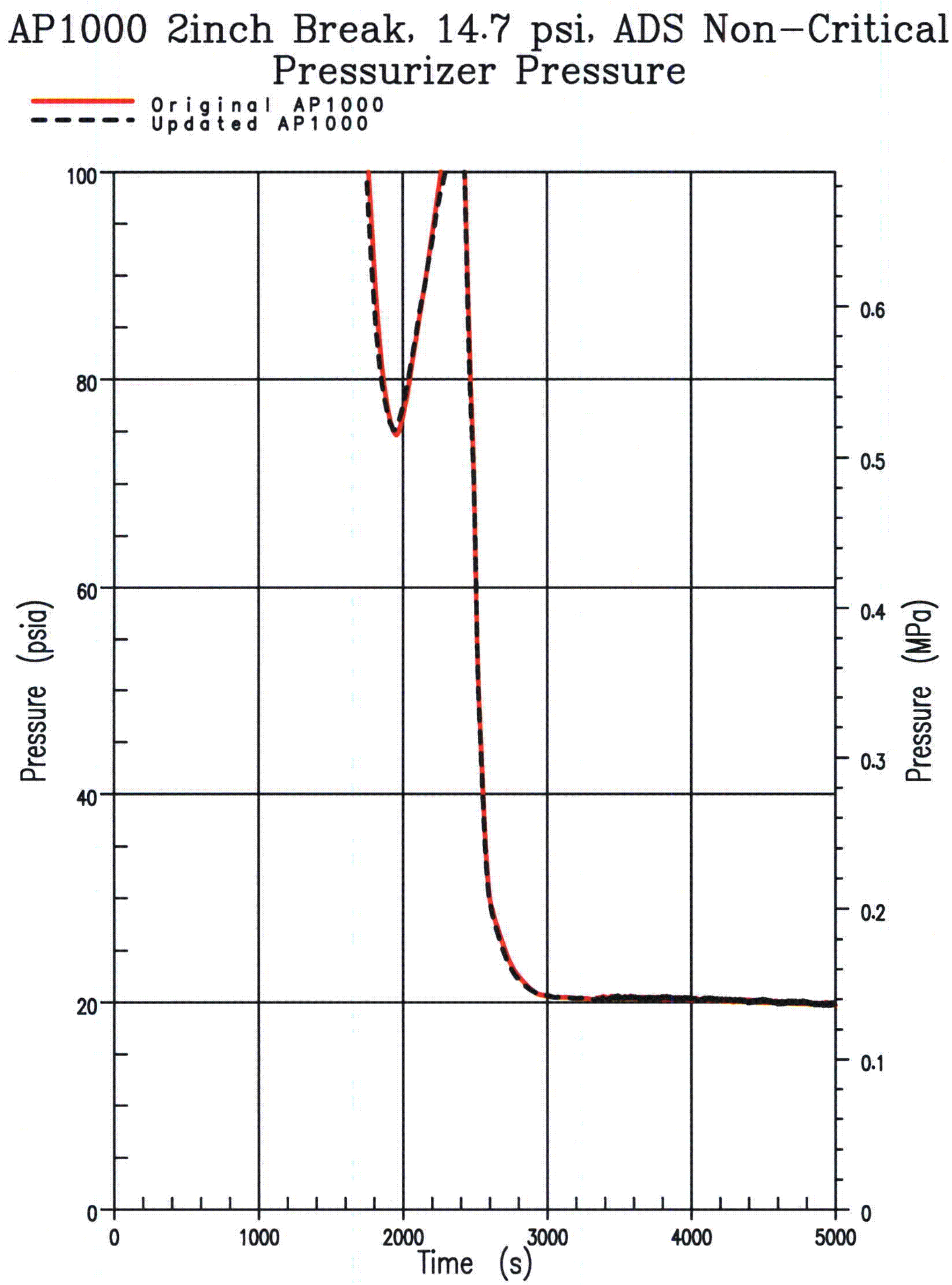


Figure 6-38
2-Inch Cold Leg Break, RCS Pressure, Low Pressure Region

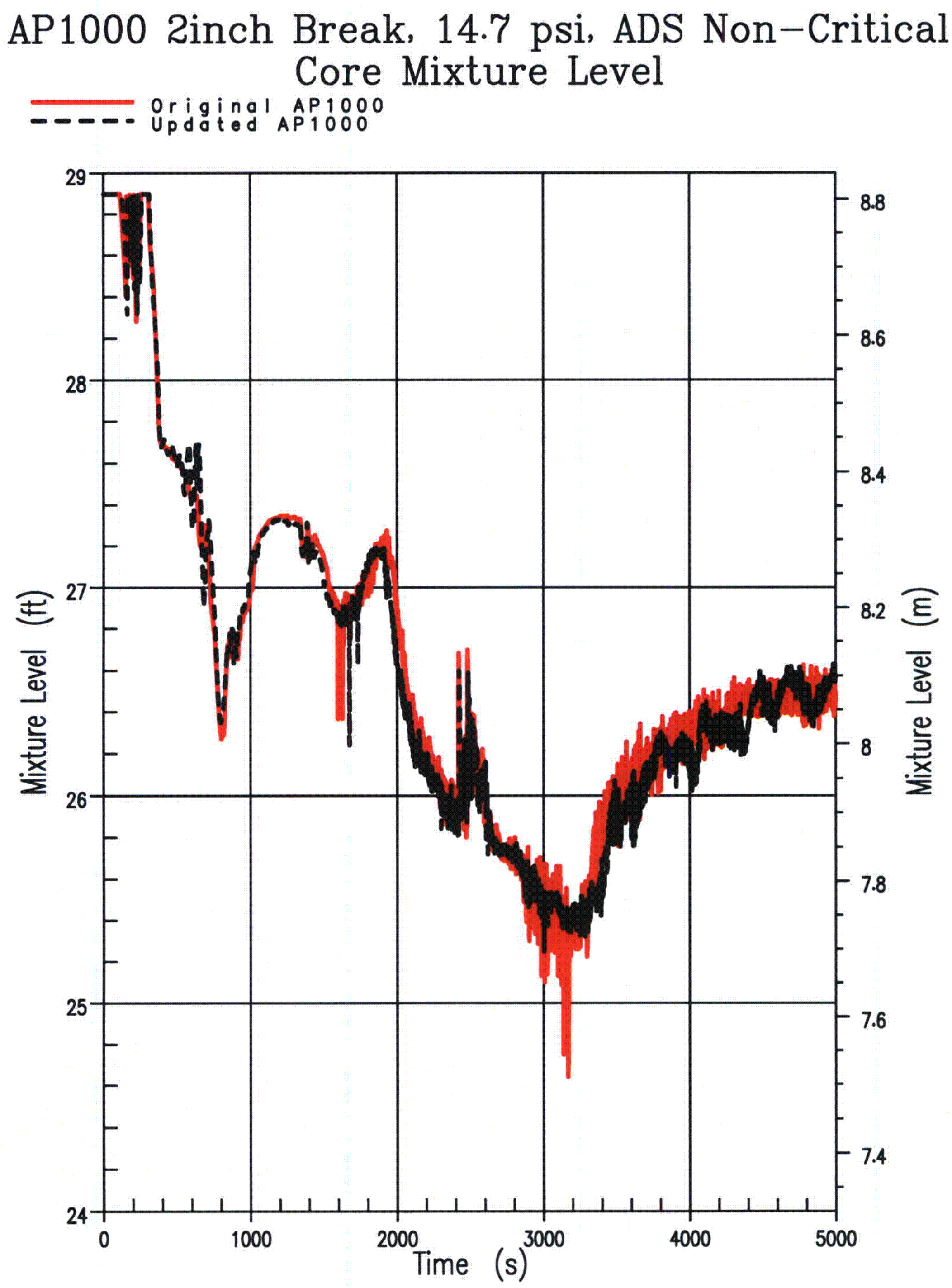


Figure 6-39
2-Inch Cold Leg Break, Core/Upper Plenum Mixture Level

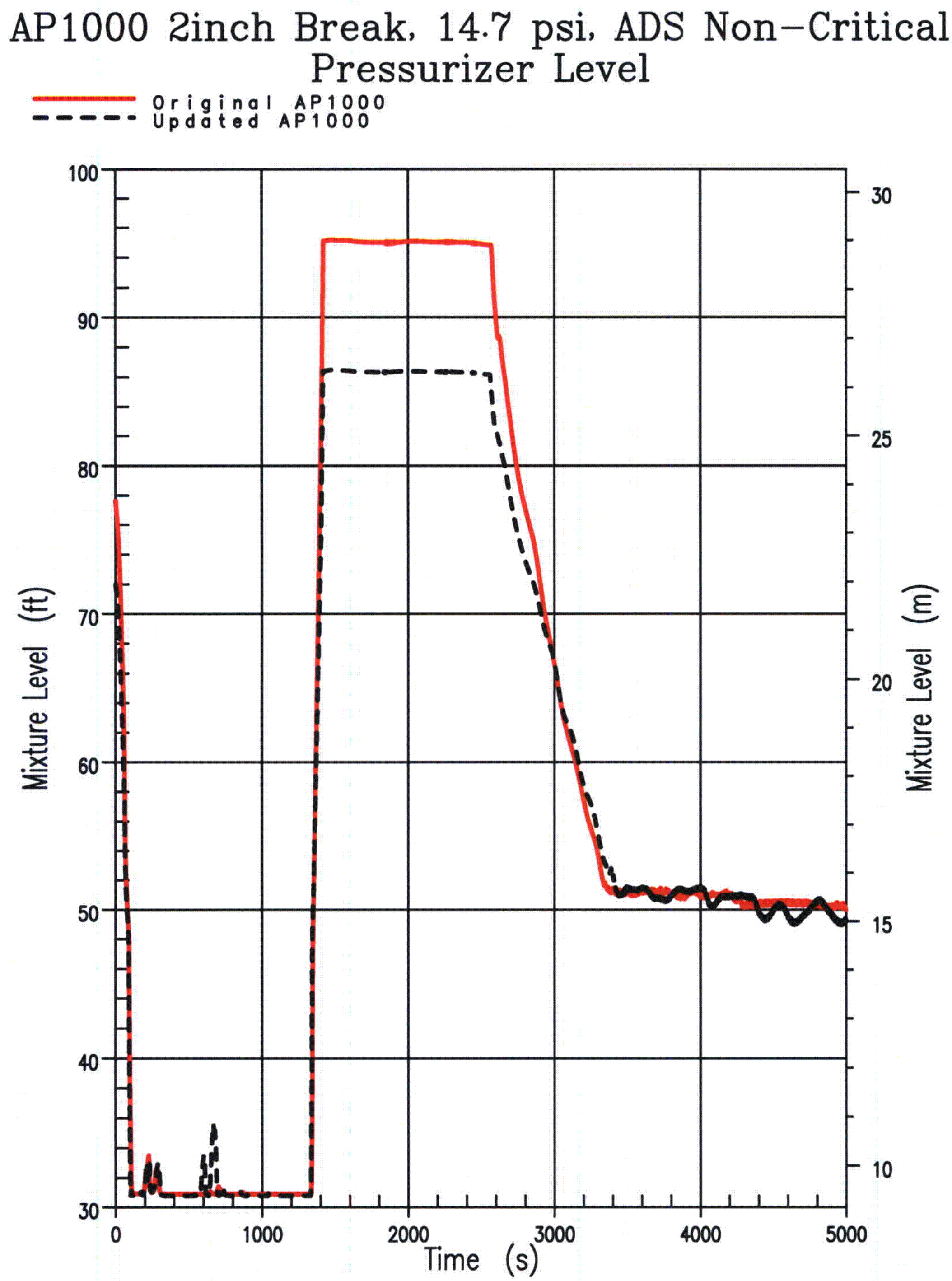


Figure 6-40
2-Inch Cold Leg Break, Pressurizer Mixture Level

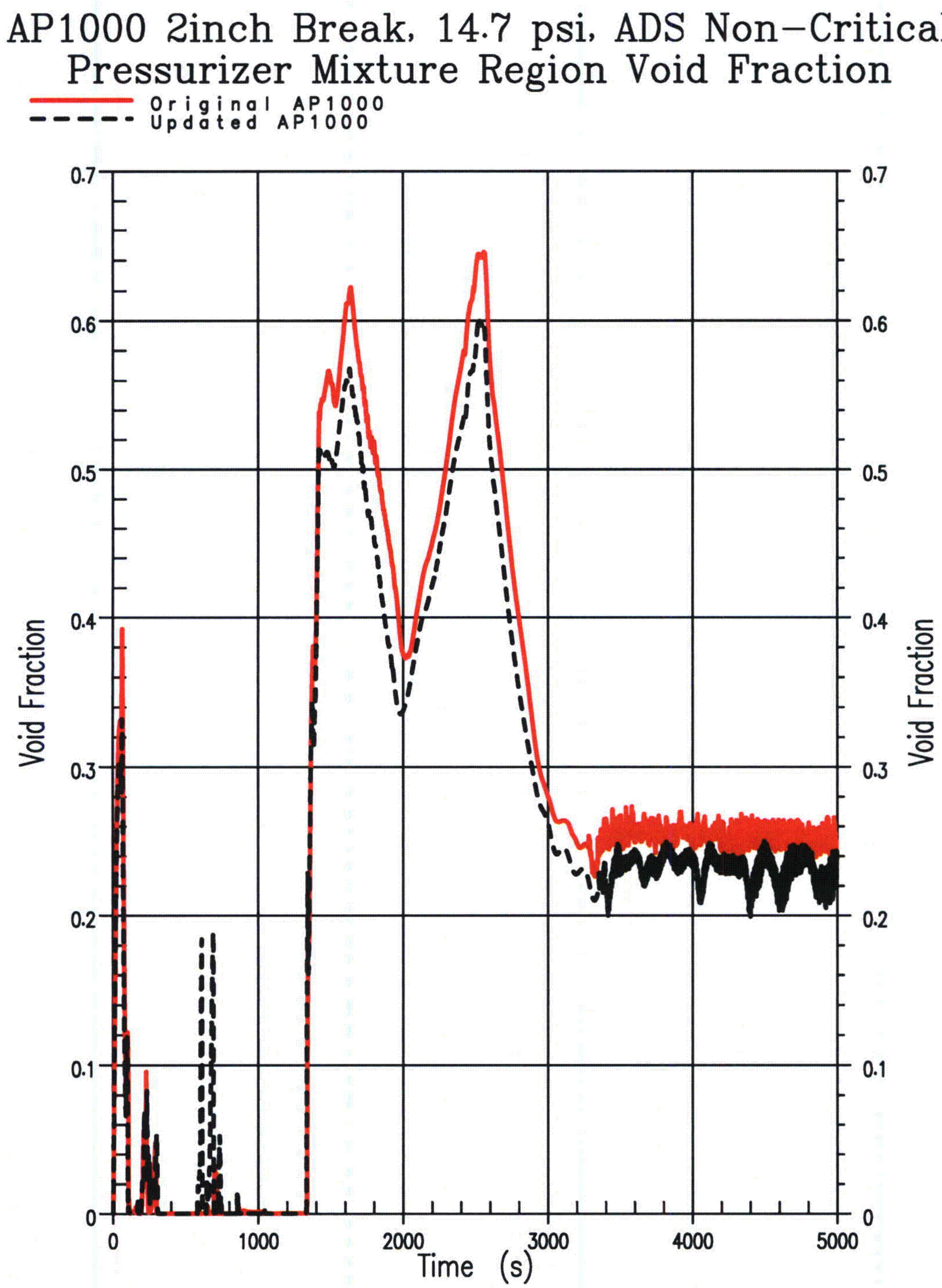


Figure 6-41
2-Inch Cold Leg Break, Pressurizer Mixture Region Void Fraction

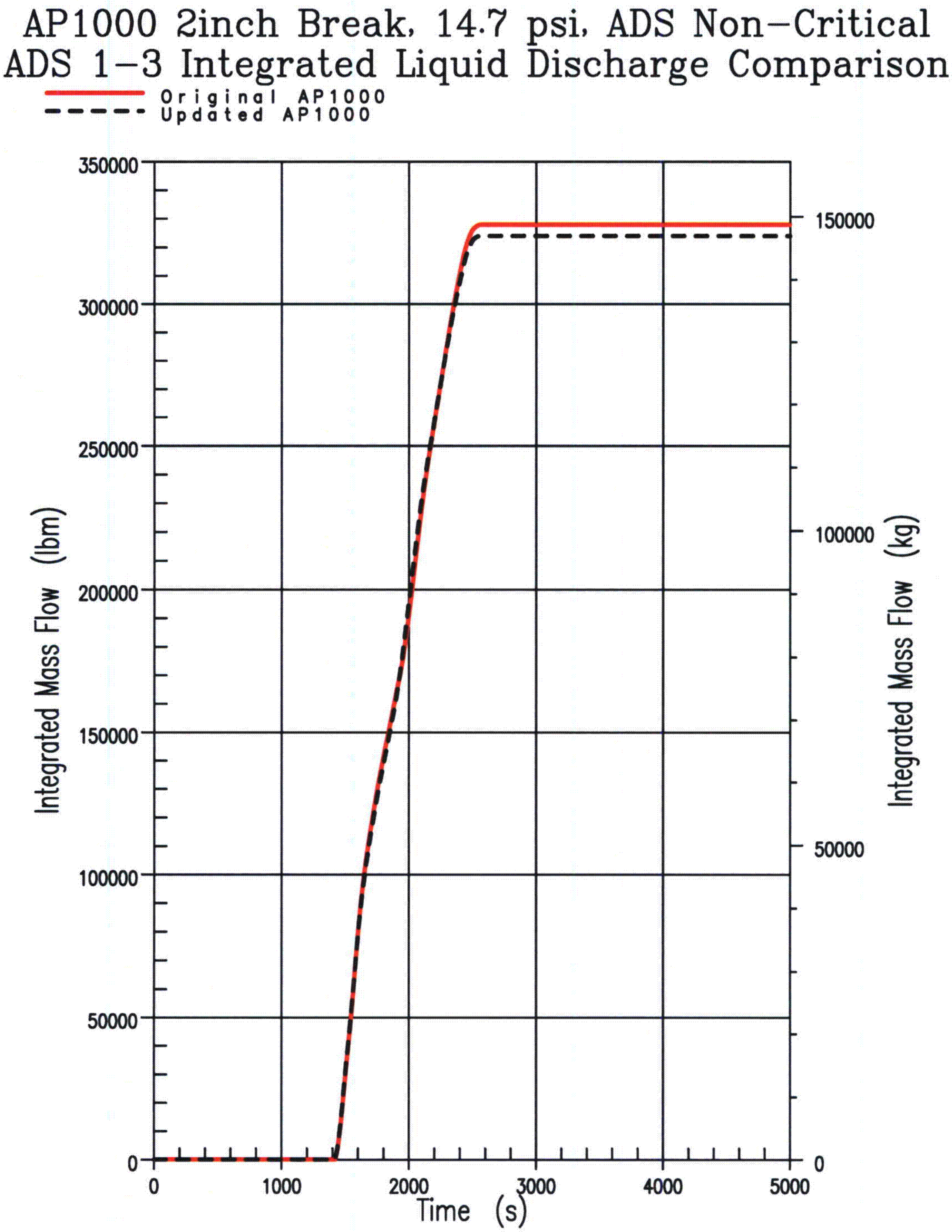


Figure 6-42
2-Inch Cold Leg Break, ADS 1-3 Integrated Liquid Discharge

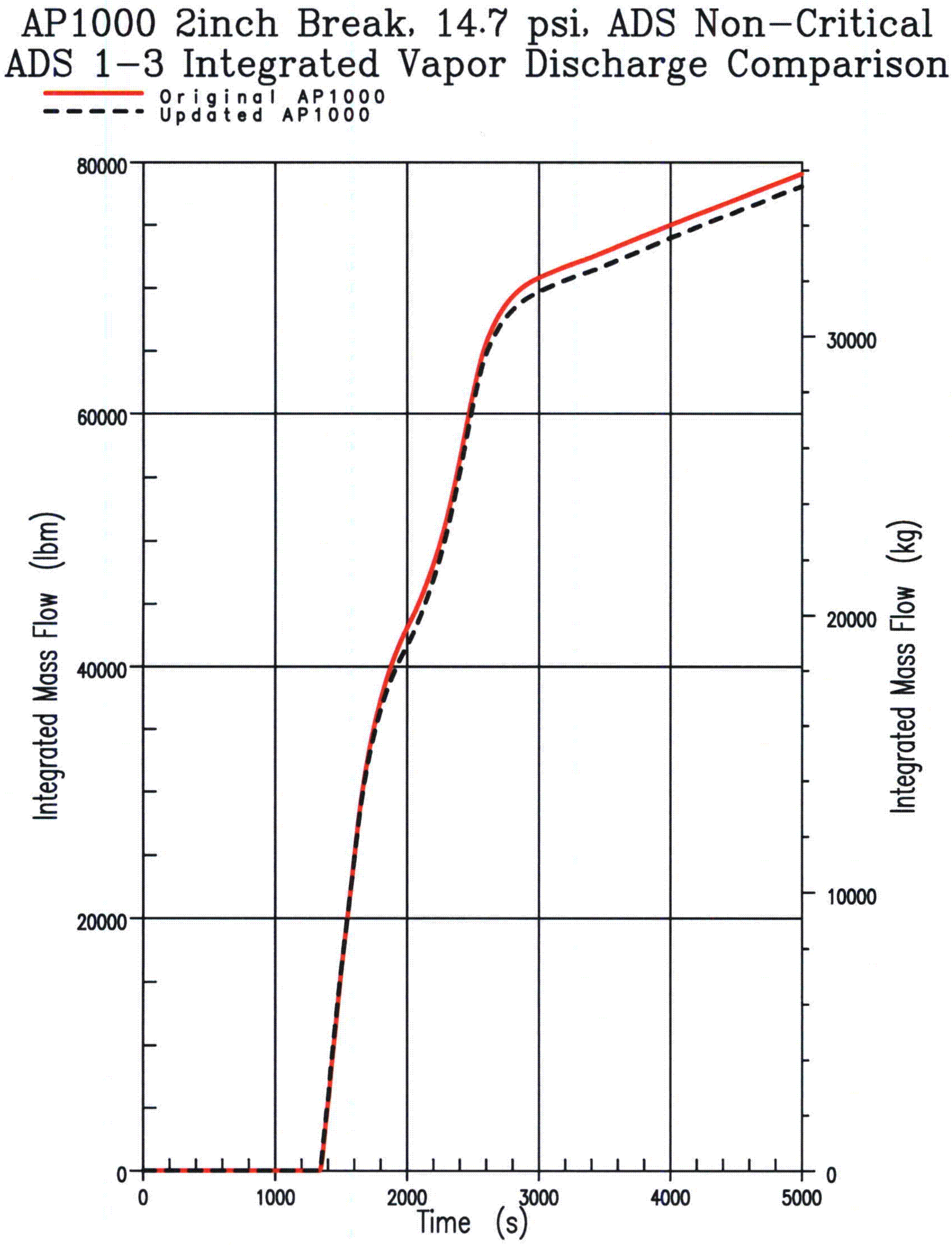


Figure 6-43
2-Inch Cold Leg Break, ADS 1-3 Integrated Vapor Discharge

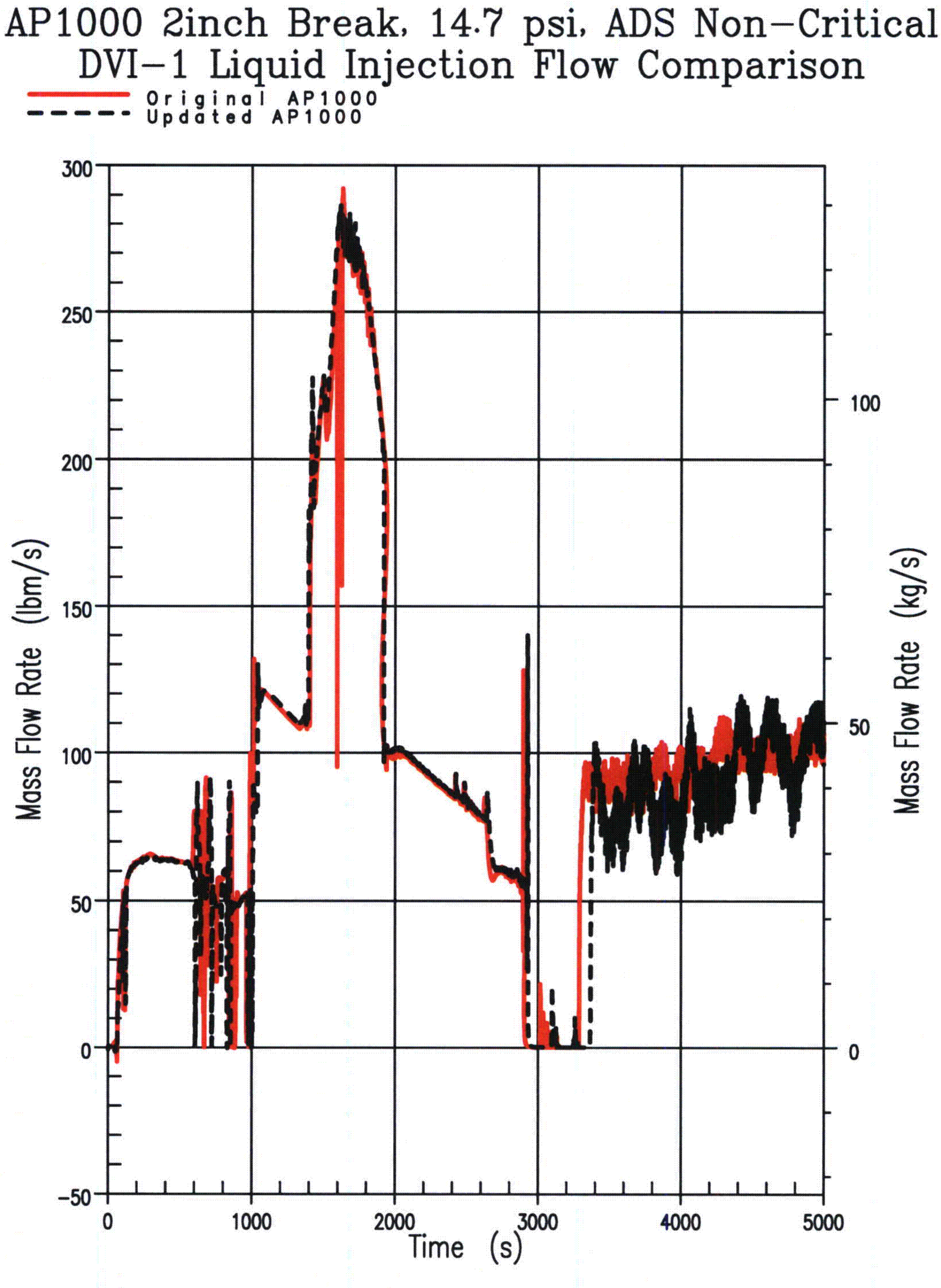


Figure 6-44
2-Inch Cold Leg Break, DVI-1 Injection Flow

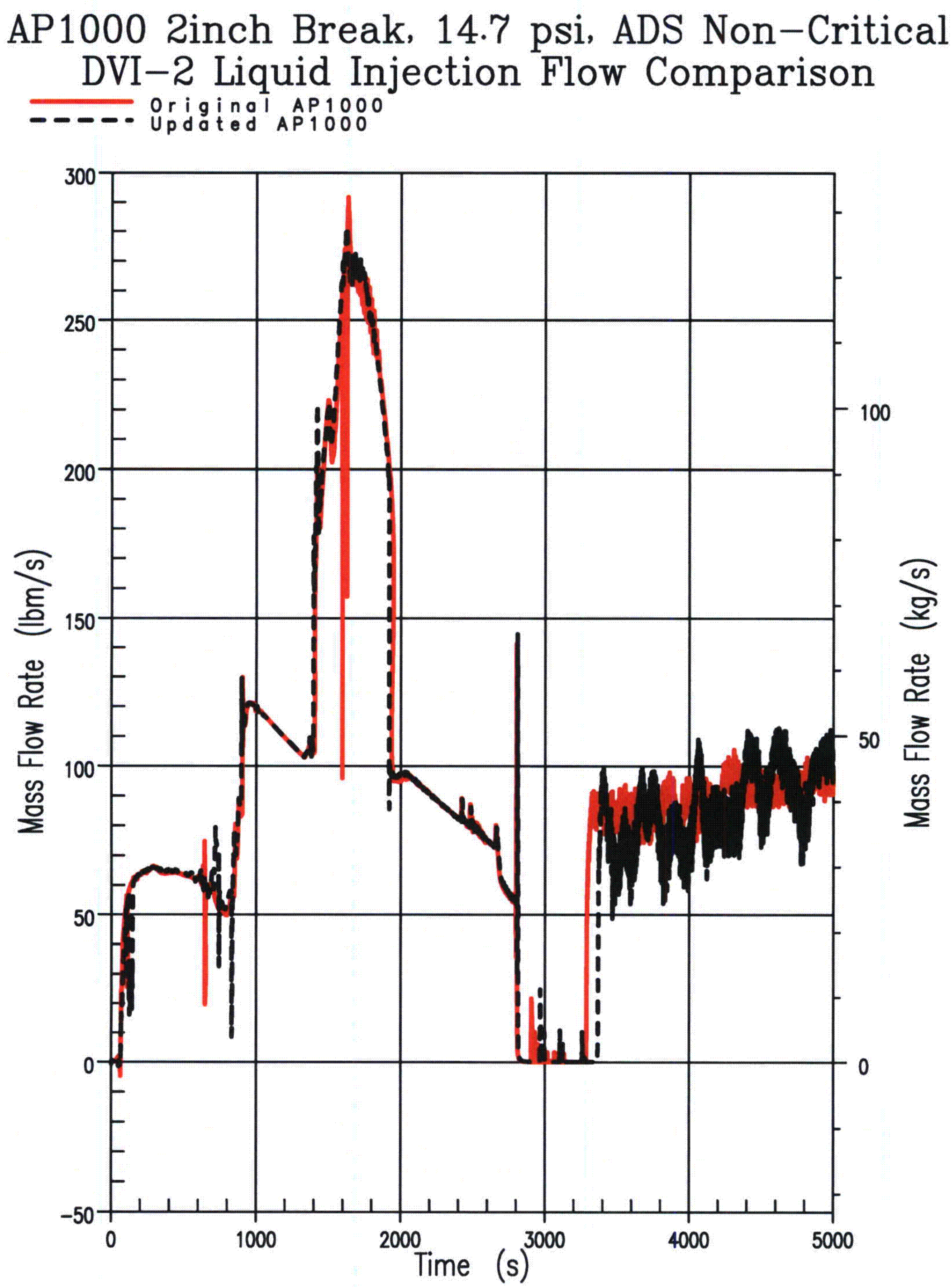


Figure 6-45
2-Inch Cold Leg Break, DVI-2 Injection Flow

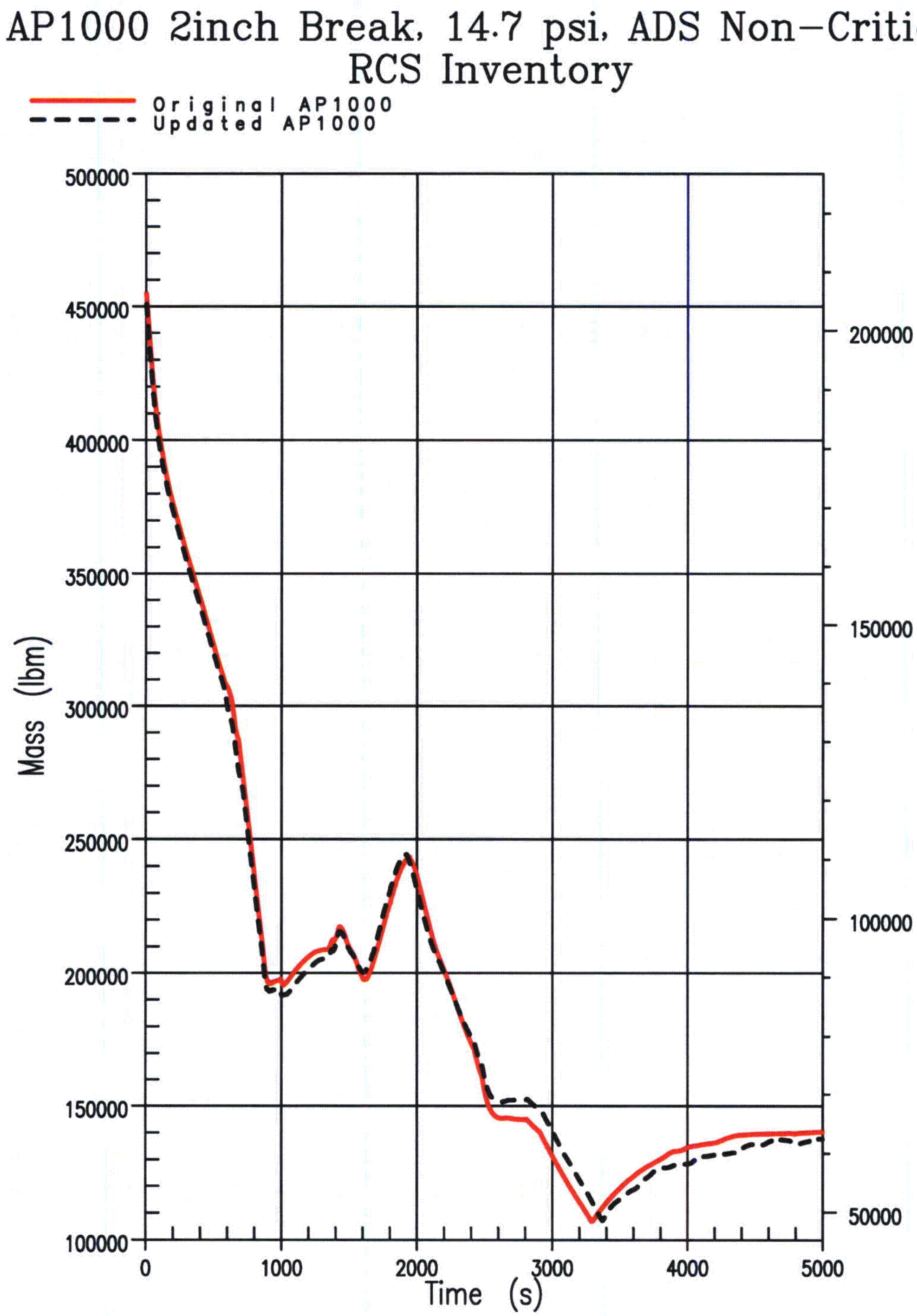


Figure 6-46
2-Inch Cold Leg Break, RCS Inventory

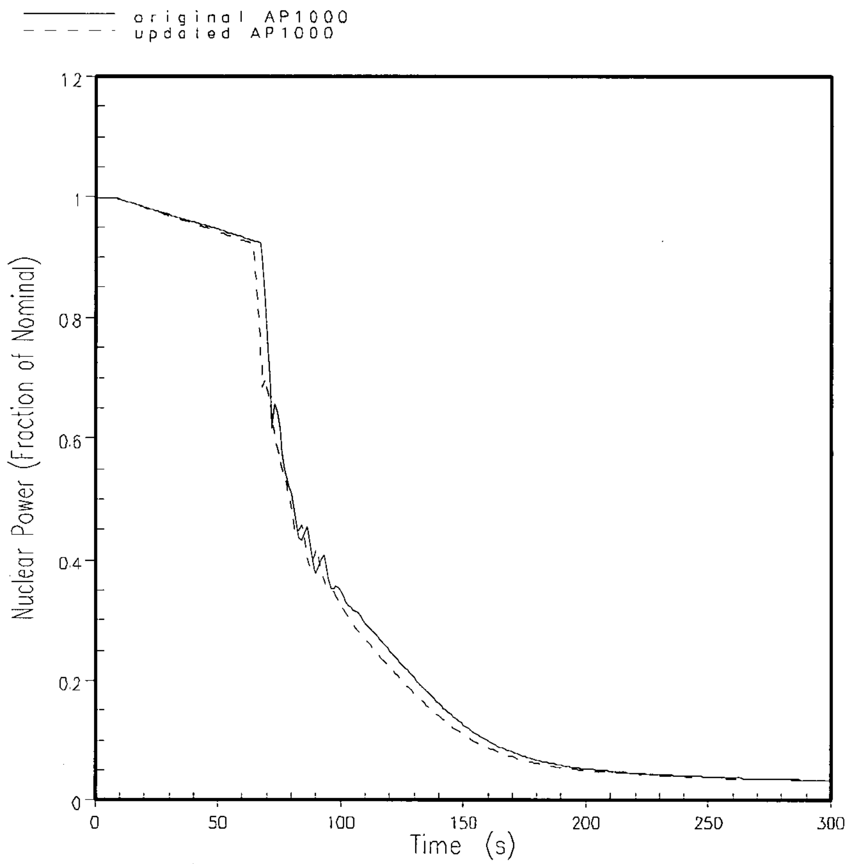


Figure 8-1
ATWS, Nuclear Power

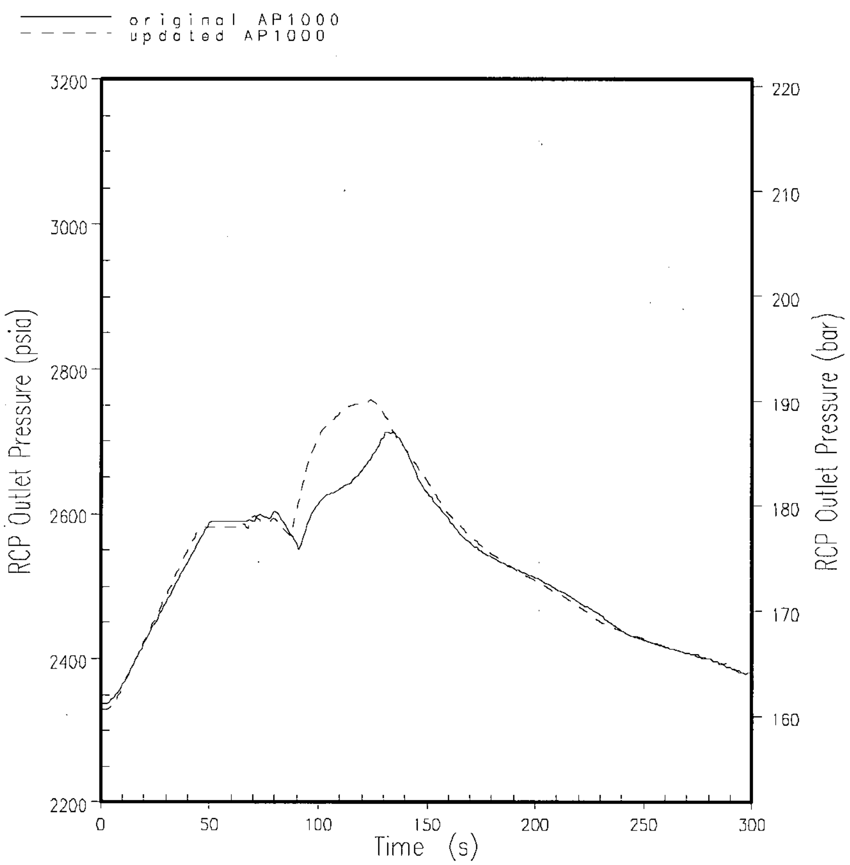


Figure 8-2
ATWS, RCP Outlet Pressure

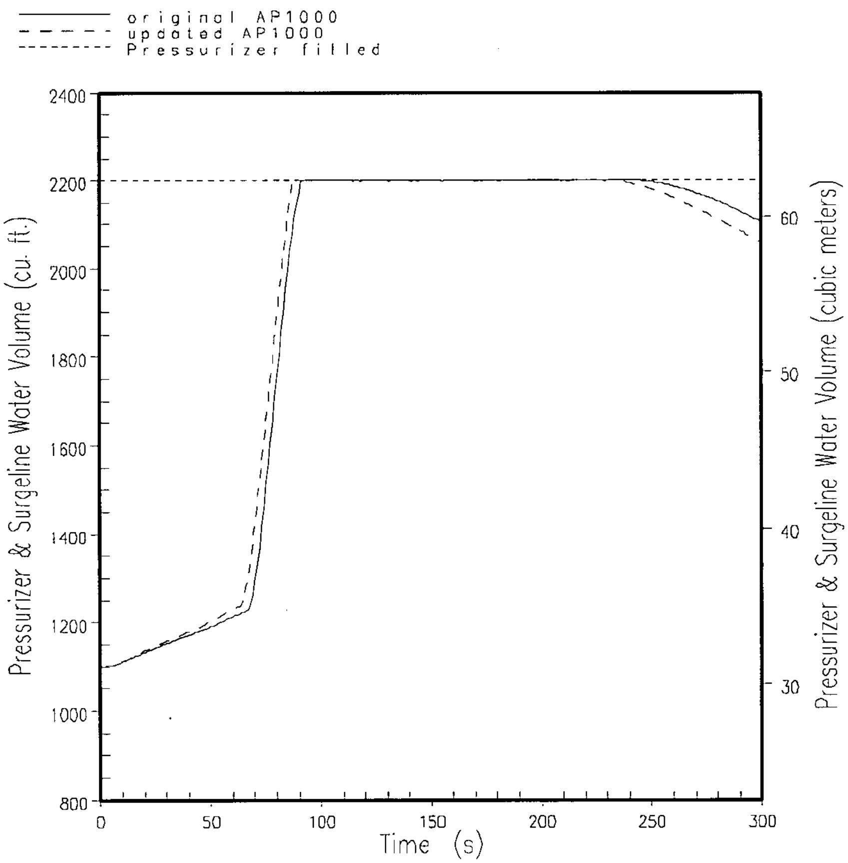


Figure 8-3
ATWS, Pressurizer and Surgeline Water Volume

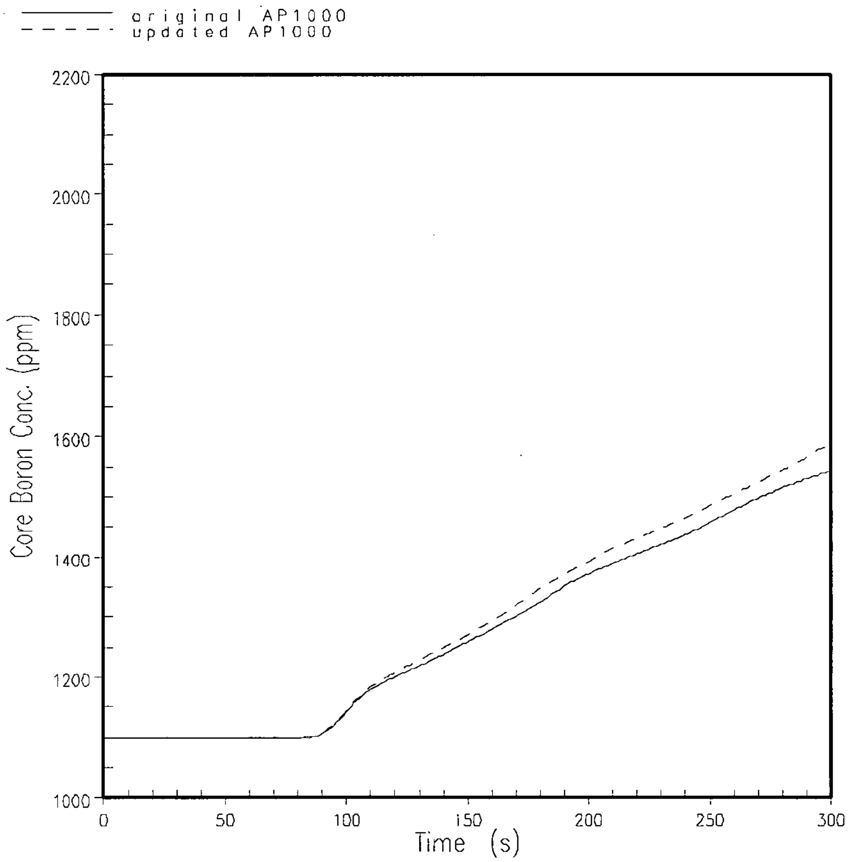


Figure 8-4
ATWS, Core Boron Concentration

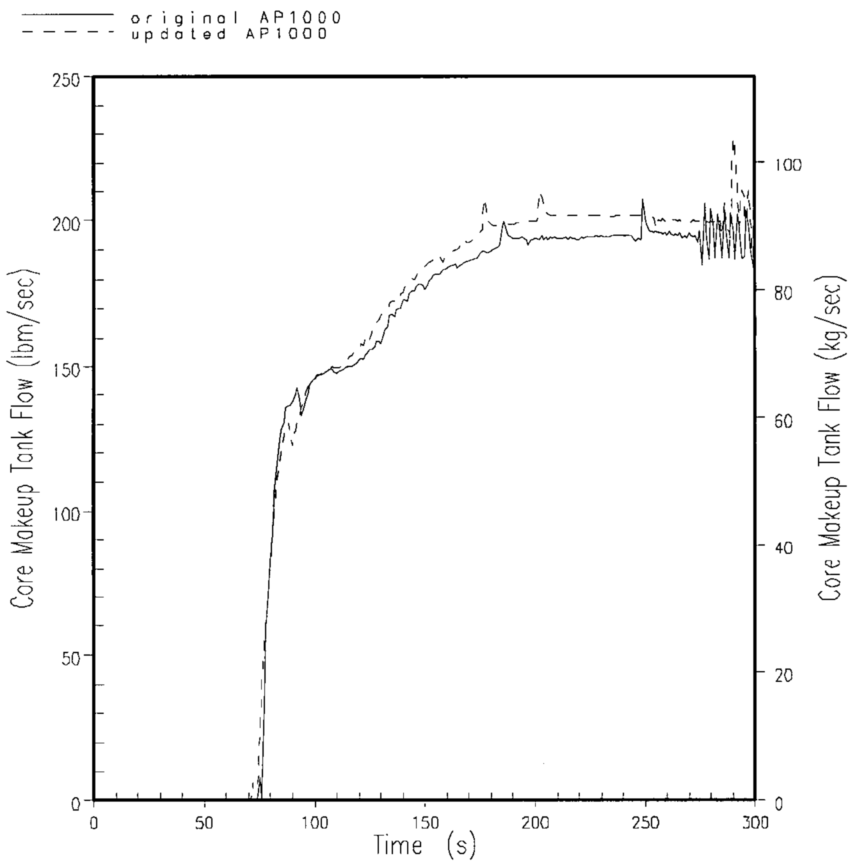


Figure 8-5
ATWS, Core Makeup Tank Flow

Design Control Document (DCD) Revision:
None

PRA Revision:
None

Technical Report (TR) Revision:
None