Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

July 15, 2009
U7-C-STP-NRC-090074

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852-2738

South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Response to Request for Additional Information

Attached are responses to NRC staff questions included in Request for Additional Information (RAI) letter numbers 76 and 89 related to Combined License Application (COLA) Part 2, Tier 2, Section 6.2 and Appendix 3B.

Attachments 1, 2, and 3 are responses to the RAI questions listed below. The responses to these RAI questions are supplements to previous responses. These supplemental responses provide all of the information requested in the RAIs.

RAI 06.02.01.01.C-1
RAI 06.02.01.01.C-2
RAI 06.02.01.01.C-8

There are no commitments in this letter.

If you have any questions regarding these responses, please contact me at (361) 972-7206, or Bill Mookhoek at (361) 972-7274.

D091
NRO

STI 32503104

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 7/15/2009

M A McBurnett

Mark A.McBurnett
Vice President, Oversight and Regulatory Affairs
South Texas Project Units 3 & 4

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Attachments:

1. Question 06.02.01.01.C-1
2. Question 06.02.01.01.C-2
3. Question 06.02.01.01.C-8

cc: w/o attachment except*
(paper copy)

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RAI 06.02.01.01.C-1:

QUESTION:

Section 6.2.1.1.3: The staff found the containment analyses in support of the certified ABWR design to be acceptable based on the use of the GESSAR methodology and confirmatory calculations by the staff. It is the staff's understanding that the applicant plans to replace GESSAR with the GOTHIC computer program. It is also the staff's understanding that the GOTHIC code was adapted to employ models and assumptions outlined in the NEDO-20533 reports. Please, provide:

- GOTHIC input deck/description for the STP ABWR DBA containment analyses,

- detailed description of how the models and assumptions presented in the NEDO-20533 reports were incorporated into the GOTHIC model, and

- reference for qualification and/or benchmarking of GOTHIC to be used as an acceptable tool for performing the STP ABWR DBA containment analysis.

RESPONSE:

1st Bullet Item:
In response to the request of the first bullet in this RAI, the input parameters for the GOTHIC pressure/temperature containment model are provided in RAI 06.02.01.01.C-1 Table 1, which was previously transmitted to NRC in STPNOC Letter U7-C-STP-NRC-090014 dated February 19, 2009. The non-proprietary version of this response was transmitted to the NRC in STPNOC Letter U7-C-STP-NRC-090010 also dated February 19, 2009.

2nd Bullet Item:
Westinghouse (WEC) has prepared a containment Pressure/Temperature (P/T) report that has been submitted to the NRC in STPNOC Letter U7-STP-NRC-090067 dated June 30, 2009. This report, WCAP-17058, describes the WEC approach for adapting the GOTHIC code to employ models and assumptions outlined in the ABWR DCD and NEDO-20533. The WCAP provides a detailed comparison of the DCD approach using NEDO-20533 and the WEC method, and evaluates the impact on the analysis results of the few unavoidable modeling differences due to certain features in the GOTHIC code. The WEC method of analysis is benchmarked against the DCD analysis. The report addresses the modeling updates as described in Part 7 STD DEP 6.2-2 of Rev 2 of the STP 3 & 4 COLA.

The analysis for calculation of pool swell, including pool swell height, velocity, bubble pressure and wetwell airspace pressure which is also affected by the modeling updates described above, will be addressed in a separate departure and will be removed from the STD DEP 6.2-2 scope. Details of this departure and analysis are described in the response to RAI 06.02.01.01.C-6 in Attachment 2 to STPNOC Letter U7-C-STP-NRC-090033. This response will be supplemented on July 31, 2009.

Consistent with this approach, COLA Rev 2 will be revised as follows : (1) Part 7 STD DEP 6.2-2 will be revised to describe the updated containment analysis, reference WCAP-17058, and refer pool swell changes to a new departure for Appendix 3B. Part 7 tables will also be updated to reflect this change; (2) Technical Specification 3.6.1.1, 3.6.1.2 and 3.6.1.4 Bases will be revised in both Part 2 Chapter 16 and in Part 4 to reflect the revised peak containment pressure, and (3) Part 2 Tier 2 Section 6.2 text, Table 6.2-1 and Figures 6.2-3, 6.2-4, 6.2-6, 6.2-7, 6.2-8, 6.2-12, 6.2-13, 6.2-22, 6.2-23, 6.2-24, and 6.2-25 will be revised or deleted as necessary to reflect updated containment temperatures and pressures.

COLA changes described above are provided in the markups in this response. These changes will also be provided in COLA Rev 3. Changed portions of the COLA Rev 2 are shown with gray highlighting. Note that the markups provided in this response supersede those provided in STPNOC Letter U7-C-STP-NRC-090033 submitted on April 20, 2009.

3rd Bullet Item:
The qualification and benchmarking of GOTHIC for the ABWR containment P/T analysis is provided in WCAP-17058, as described above. The benchmarking performed shows close agreement between the WEC results and the DCD results. In addition, the GOTHIC program is used to calculate pressure and temperature in the containment for the Feedwater Line Break (FWLB) and Main Steam Line Break (MSLB) using the DCD modeling assumptions with the updates identified in STD DEP 6.2-2. These results are in close agreement with the results from NEDO-33372 which incorporated the analysis methodology and assumptions from the DCD with the updates incorporated.

6.2.1.1.2.1 Drywell

STD DEP 6.2-2

The maximum drywell temperature occurs in the *case of a steamline break (~~169.7°C~~~~161~~ 173.2°C).* Although this exceeds the ABWR drywell design temperature (171.1°C), it only exceeds it by 2.1°C and only for about 2 seconds. Due to thermal inertia, components in the drywell would not have sufficient time to reach the design limit temperature.

The maximum drywell pressure occurs in the *case of a feedwater line break (~~268.7~~~~240~~ 281.8 kPaG). The design pressure for the drywell (309.9 kPaG) includes ~~16%~~* approximately ~~22~~ 10% *margin.*

6.2.1.1.2.2 Wetwell

STD DEP 6.2-2

*The wetwell chamber design pressure is 309.9 kPaG and design temperature is ~~103.9°C~~*104°C.

Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 35°C or less. ~~Under blowdown conditions following an isolation event or LOCA, the initial pool water temperature may rise to a maximum of 76.7 [ ]°C.~~ The continued release of decay heat after the initial blowdown following an isolation event or LOCA *may result in suppression pool temperatures as high as ~~97.2~~* 99.5°C. *The Residual Heat Removal (RHR) System is available in the Suppression Pool Cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the Reactor Building Cooling Water (RCW) System and finally to the Reactor Service Water (RSW) System. The RHR System is described in Subsection 5.4.7.*

6.2.1.1.3.3 Accident Response Analysis

STD DEP 6.2-2

The containment design pressure and temperature were established based on enveloping the results of this range of analyses ~~plus providing NRC prescribed margins~~.

For the ABWR pressure suppression containment system, the peak containment pressure following a LOCA is ~~very~~ relatively *insensitive to variations in the size of the assumed primary system rupture. This is because the peak occurs late in the blowdown and is determined in very large part by the transfer of the noncondensible* *gases from the drywell to the wetwell airspace.* ~~*This process is not significantly influenced by the size of the break.*~~ *In addition, there is* ~~*a 15%*~~ an approximately ~~22~~ 10 % *margin between the peak calculated value and the containment design pressure that will easily accomodate small variations in the calculated maximum value.*

Tolerances associated with fabrication ~~*and installation may result in the as-built size of the postulated break areas being 5% greater than the values presented in this chapter Based on the above, these as-built variations would not invalidate the plant safety analysis presented in this chapter and Chapter 15*~~ of the RPV nozzles have been taken into account in this analysis.

6.2.1.1.3.3.1.1 Assumptions for Short-Term Response Analysis

STD DEP 6.2-2

The response of the Reactor Coolant System and the Containment System during the short-term blowdown period of the accident has been analyzed using the following assumptions:

(1) The initial conditions for the FWLB accident ~~*are such that system energy is maximized and the system mass is minimized*~~ maximize the containment pressure response. That is:

(a) The reactor is operating at 102% of the rated thermal power, which maximizes the post-accident decay heat.

(b) The initial suppression pool mass is at the ~~*low*~~~~nominal~~ high *water level.*

(c) The initial wetwell air space volume is at the high water level.

(d) The suppression pool temperature is the operating maximum ~~*temperature*~~ value.

(4) ~~*The main steam isolation valves (MSIVs) start closing at 0.5 s after the accident*~~. *The main steam isolation valves (MSIVs) start closing at 0.5 s after the accident.* ~~*They are fully closed in the shortest possible time (at 3.5 s) following closure initiation.*~~. The turbine stop valves are closed in 0.2 seconds after reactor trip/turbine trip (RT/TT). *By assuming rapid closure of these valves, the RPV is maintained at a high pressure, which maximizes the calculated discharge of high energy water into the drywell.*

(5) ~~*The vessel depressurization flow rates are calculated using Moody's homogeneous equilibrium model (HEM) for the critical break flow (Reference 6.2-2).*~~ *The vessel depressurization flow rates are calculated using Moody's* ~~*homogeneous equilibrium model (HEM) for the critical break flow*~~ critical flow model *(Reference 6.2-2).* ~~*The break area on the RPV side for this study is shown in Figure 6.2-2. During the inventory depletion period, subcooled blowdown occurs and the effective break area at saturated conditions is much less than the actual area. The detailed calculational method is provided in Reference 6.2-1*~~.

~~*Reactor vessel internal heat transfer is modeled by dividing the vessel and internals into six metal nodes. A seventh node depends on the fluid (saturated or subcooled liquid, saturated steam) covering the node at the time. The assumptions include:*~~

~~*(a) The center of gravity of each node is specified as the elevation of that node.*~~

~~*(b) Mass of water in system piping (except for HPCF and feedwater) is included in initial vessel inventory.*~~

~~*(c) Initial thermal power is 102% of rated power at steady-state conditions with corresponding heat balance parameters which correspond to turbine control valve constant pressure of 6.75 MPaA.*~~

~~*(d) Pump heat, fuel relaxation, and metal-water reaction heat are added to the ANSI/ANS-5.1 decay heat curve plus 20% margin.*~~

~~*(e) Initial vessel pressure is 7.31 MPaA*~~.

(6) ~~*There are two HPCF Systems, one RCIC System, and three RHR Systems in the ABWR. One HPCF System, one RCIC System and two RHR Systems are assumed to be available. HPCF flow cannot begin until 36 seconds after a break, and then the flow rate is a function of the vessel-to-wetwell differential pressure. Rated HPCF flow is 182 m3/h per system at 8.12 MPaD and 727 m3/h, per system at 0.69 MPaD. Rated RHR flow is 954 m3/h at 0.28 MPaD with shutoff head of 1.55 MPaD. Rated RCIC flow is 182 m3/h with reactor pressure between 8.12 MPaG and 1.04 MPaG, and system shuts down at 0.34 MPaG.*~~ Influence of ~~these~~ the ECCS systems is minimal since the time interval analyzed for short-term is approximately the same time as the response time of associated systems injections into the RPV.

(8) ~~*The wetwell airspace temperature is allowed to exceed the suppression pool temperature as determined by a mass and energy balance on the airspace.*~~ *The wetwell airspace temperature is allowed to exceed the suppression pool temperature as determined by a mass and energy balance on the airspace.* ~~Not Used~~

(9) *Wetwell and drywell ~~wall and~~ wall and structure heat transfer ~~are~~*~~is~~ *are ignored.*

(10) *Actuation of SRVs is modeled.*

(11) *Wetwell-to-drywell vacuum breakers ~~are not modeled~~*~~do not open in the short term response analysis.~~ are modeled but do not open.

(12) *Drywell and wetwell sprays and RHR cooling mode are not modeled.*

(13) ~~*The dynamic backpressure model is used.*~~ Not Used

(14) *Initial drywell conditions are ~~0.107 MPa, 57°C~~*~~106.5 kPa,~~ *0.107 MPa, 57°C and 20% relative humidity.*

(15) *Initial wetwell airspace conditions are ~~0.107 MPa~~106.5 kPa 0.107MPa, 35°C and 100% relative humidity.*

(16) ~~*The drywell is modeled as a single node. All break flow into the drywell is homogeneously mixed with the drywell inventory.*~~ *The drywell is modeled as a single node. All break flow into the drywell is homogeneously mixed with the drywell inventory.* ~~Not Used~~

(17) ~~*Because of the unique containment geometry of the ABWR, the inert atmosphere in the lower drywell would not transfer to the wetwell until the peak pressure in the drywell is achieved. Figure 6.2-5 shows the actual case and the model assumption. Because the lower drywell is connected to the drywell connecting vent, no gas can escape from the lower drywell until the peak pressure occurs. This situation can be compared to a bottle whose opening is exposed to an atmosphere with an increasing pressure. The contents of the lower drywell will start transferring to the wetwell as soon as the upper drywell pressure starts decreasing. A conservative credit for transfer of 50% of the lower drywell contents into the wetwell was taken)*~~ Not Used.

6.2.1.1.3.3.1.2 Assumptions for Long-Term Cooling Analysis

STD DEP 6.2-2

Following the blowdown period, the ECCS discussed in Section 6.3 provides water for core flooding, containment spray, and long-term decay heat removal. The containment pressure and temperature response during this period was analyzed using the following assumptions:

(1) ~~*The ECCS pumps are available as specified in Subsection 6.3.1.1.2 (except one low pressure flooder feeding a broken feedwater line, in case of a FWLB).*~~ There are two HPCF Systems, one RCIC System, and three RHR Systems in the ABWR. All motor operated pump systems (HPCF and RHR) are assumed to be available to maximize pump heat into the suppression pool *A single failure of one RHR heat exchanger was assumed for conservatism.*

(2) *The ANSI/ANS-5.1-* 1979 *decay heat* plus 2-sigma uncertainty *is used. Fission energy, fuel relaxation heat, and pump heat are included.*

(3) *The suppression pool* ~~*is the only*~~ modelled as a ~~*heat sink available in the containment system.*~~ volume corresponds to the low water level.

(4) ~~*After 10 minutes, the RHR heat exchangers are activated to remove energy via recirculation cooling of the suppression pool with the RCW System and ultimately to the RSW System. This is a conservative assumption, since the RHR design permits initiation of containment cooling well before a 10 minute period (see response to Question 430.26).*~~ After 30 minutes, one RHR heat exchanger is activated to remove energy via recirculation cooling of the suppression pool and

one RHR heat exchanger is activated to remove energy via drywell sprays with the RCW System and ultimately to the RSW System.

(6) *The lower drywell flooding ~~of 815m3 was assumed to occur 70 seconds after scram. During the blowdown phase, a portion of break flow flows into the lower drywell. This is conservative, since lower drywell flooding will probably occur at approximately 110 to 120 second time period~~* is not modeled. Water which is from the lower drywell is assumed to be mixed with the suppression pool to calculate the bulk average temperature.

(7) *~~At 70 seconds, the feedwater specific enthalpy becomes 418.7 J/g (100°C saturation fluid enthalpy)~~* Structural heat sinks are modeled in the containment system.

6.2.1.1.3.3.1.4 Long-Term Accident Responses

STD DEP 6.2-2

In order to assess the adequacy of the containment system following the initial blowdown transient, an analysis was made of the long-term temperature and pressure response following the accident. The analysis assumptions are those discussed in Subsection 6.2.1.1.3.3.1.2.

~~The short-term pressure peak (268.7 kPaG) of Figure 6.2-6 is the peak pressure for the whole transient.~~ Figure 6.2-8 shows temperature time histories for the suppression pool, wetwell, and drywell temperatures. The peak pool temperature (~~96.9~~ 99.5*°C) is reached at ~~15,350~~* 6600 *seconds (~~4.264~~* 1.833 *hours.* This is less than the suppression pool temperature value of 100°C which is used in the net positive suction head available (NPSHA) calculations.

6.2.1.1.3.3.2 Main Steamline Break

STD DEP 6.2-2

Flow from the condenser side of the break continues for 0.5 seconds, at which time the MSIVs begin to close on high flow signal. A valve stroke time of ~~5~~ 4.5 seconds is used for the MSIV closure. Flow from the condenser side of the break is ~~linearly~~ ramped down to zero between 0.5 and ~~5.5~~ 5 seconds. *~~The effective break area used for the MSL is shown in Figure 6.2-10. More detailed descriptions of the MSL break model are provided in the following:~~*

6.2.1.1.3.3.2.1 Assumptions for Short-Term Response Analysis

STD DEP 6.2-2

The response of the reactor coolant system and the containment system during the short-term blowdown period of the MSLB accident is analyzed using the assumptions listed in the above subsection and Subsection 6.2.1.1.3.3.1.1 for the feedwater line *break,* ~~*with the following exceptions:*~~ except feedwater mass flow rate for a MSL break was assumed to be 130% NBR for the case where no operator action is assumed to control water level. Additional cases were run with feedwater mass flow rate regulated to control RPV water level or with no feedwater flow based on an assumed loss of offsite power.

6.2.1.1.3.3.2.3 Short-Term Accident Response

The maximum drywell temperature (~~161~~ 173.2°C) is predicted to occur for the steamline break. *The MSLB with two-phase blowdown starting when the RPV* ~~*collapsed*~~ water *level is at* or below *the main steamline nozzle provides the highest peak drywell temperature. The peak drywell* air *temperature is* ~~*169.7*~~~~161~~ 173.2*°C,* ~~*below the*~~ which is above the *design value of 171.1°C, and is the limiting one as compared to the FWLB peak temperature.* As noted in Section 6.2.1.1.2.1, this peak calculated drywell temperature exceeds the design limit for only 2 seconds. *The peak drywell pressure for the MSLB remains below that for the FWLB, which becomes the most limiting.*

6.2.2.3.1 System Operation and Sequence of Events

(4) ~~*Containment cooling is initiated after 10 minutes (see Response to Question 430-26).*~~ Containment cooling is initiated after 30 minutes.

6.2.8 References

STD DEP 6.2-2

6.2-5 "Implementation of GE NEDO-20533 Methodology with GOTHIC for ABWR Containment Design Analyses," WCAP-17058, Westinghouse Electric Company, LLC, June 2009.

Table 6.2-1 Containment Parameters

Design Parameter	**Design Value**	**Calculated Value**[1]
1. Drywell pressure	*309.9 kPaG*	~~268.7 kPaG~~ ~~240~~ **281.8 kPaG**
2. Drywell temperature	*171.1°C*	~~170°C~~ ~~161~~ **173.2°C**[2]
3. Wetwell pressure	*309.9 kPaG*	~~179.5~~ ~~210.2~~ **219.3 kPaG**
4. Wetwell temperature		
• Gas Space	~~103.9 °C~~ **104°C**	~~98.9~~ 98.6°C
• Suppression pool	~~97.2~~ 100°C	~~96.9~~ 99.5°C

[1] Calculated values from Ref 6.2-5

[2] Calculated drywell maximum temperature exceeds design temperature for only 2 seconds. See discussion in Section 6.2.1.1.2.1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

STD DEP 6.2-2

The maximum allowable leakage rate for the primary containment (La) is 0.5% by weight of the containment air per 24 hours at the ~~maximum~~ calculated *peak containment pressure (Pa) of ~~0.269 MPaG~~* ~~279.6240~~ 281.8) kPaG *or* ~~[0.259~~ 0.257*% by weight of the containment air per 24 hours at the reduced pressure of Pt of* ~~[124.1~~ 144.8 *~~MPaG~~*kPaG *(Ref. 1).*

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

STD DEP 16.3-44

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions. Failure to meet air lock leakage testing (SR 3.6.1.2.1), resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.~~7~~6~~),] main steam isolation valve leakage (SR 3.6.1.3.13~~), or hydrostatically tested valve leakage (SR 3.6.1.3.~~12~~11) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J. The Frequency is required by 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

STD DEP 16.3-45
STD DEP 6.2-2

REFERENCES

1. ~~DCD Tier 2, Section 6.2.~~ WCAP-17058, June 2009.

*2. DCD Tier 2, Section ~~15.1~~*15.6.

3. 10 CFR 50, Appendix J.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Locks

STD DEP 6.2-2

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (La) of 0.5% (excluding MSIV leakage) by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (Pa) of ~~0.269 MPaG~~ ~~240~~ 281.8 kPaG *(Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.*

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

The information in this section of the reference ABWR DCD, including all subsections, is incorporated by reference with the following departure.

STD DEP 6.2-2

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among *the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of $5.20x10^{-3}$ MPaG. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of 0.310 MPaG.*

The maximum calculated drywell pressure occurs during ~~the reactor blowdown phase of the DBA, which is determined to be~~ a feedwater line break. The calculated peak drywell pressure for this limiting event is ~~0.269 MPaG~~ ~~240~~ 281.8 kPaG *(Ref. 1).*

Part 7, Section 2.2 Departures from the Generic Technical Specifications

STD DEP 6.2-2, Containment Analysis

Description

This departure updates the containment analysis for the ABWR DCD in ~~two~~ three areas: (1) the modeling of flow and enthalpy into the drywell for the feedwater following a FWLB, ~~and~~ (2) the modeling of the drywell connecting vents for the FWLB and MSLB, and (3) the modeling of decay heat. A more detailed description is shown below.

This departure also makes the following changes: (1) it updates the suppression pool temperature limit from the DCD specified value of 97.2°C to a value of 100°C, and (2) it revises the assumed elapsed time between the start of the LOCA and the initiation of suppression pool cooling and containment sprays from 10 minutes to 30 minutes.

In the ABWR DCD for the FWLB, the maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR) flow, based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the Feedwater Control System would respond to the decreasing reactor pressure vessel (RPV) water level by demanding increased feedwater flow, and there was no FWLB logic/mitigation in the certified ABWR design, this maximum feedwater flow was assumed to continue for 120 seconds. This was based on the following assumptions:

(1) All feedwater system flow is assumed to go directly to the drywell.

(2) Flashing in the broken feedwater line was ignored.

(3) Initial feedwater flow was assumed to be 105% NBR.

(4) The feedwater pump discharge flow will coast down as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100 meters was assumed on the feedwater system side.

Subsequent to certification, analysis for plant-specific ABWRs revealed that these assumptions were non-conservative.

For the containment analysis, the feedwater system side of the FWLB has been changed using a revised time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy have been determined from the predicted characteristics of a typical feedwater system. The conservatism of the assumed mass flow and enthalpies will be confirmed after detailed condensate and feedwater designs are complete. In addition, to provide added assurance of acceptable results, safety related FWLB mitigation has been added to the STP 3 & 4 ABWR design which adds safety related instrumentation to sense and confirm a FWLB based on high differential pressure between feedwater lines coincident with high drywell pressure to trip the condensate pumps (Ref. STD DEP T1 2.4-2). This automated condensate pump trip is not credited in the containment analysis.

The analysis is further revised to reflect the characteristics of the horizontal vents configuration that had not been modeled in the DCD. The certified DCD model did not properly simulate the horizontal vent portion of the vent system and incorrectly modeled the vent clearing time. The revised STP 3 & 4 ABWR containment analysis has been performed using the drywell connecting vent (DCV) loss coefficients and considering the horizontal vents. The total DCV loss coefficient is based on a summation of losses.

Further analysis done based on ANSI/ANS-5.1 (1994), including the 2-sigma uncertainty, has determined that the decay heat curves used in the DCD based on best estimate ANSI/ANS-5.1 (1979) were non-conservative for long-term analysis. To address this, the decay heat curves used in the revised containment analysis were revised to reflect the ANSI/ANS-5.1 (1979) with 2-sigma uncertainty included.

The revised containment analysis uses the GOTHIC code and is documented in WCAP-17058. The analysis uses the same assumptions and inputs that were used in the DCD with consideration of the revised modeling as noted above. The report describes all input assumptions, baselining of the GOTHIC code results to those used in the DCD and all containment time-dependent pressure and temperature results. The report also evaluates the impact on the analysis results of the few unavoidable modeling differences due to certain features in the GOTHIC code.

The impact of the revised pressure and temperature results on pool swell velocity and height described in Appendix 3B is evaluated in a new departure which is STD DEP 3B-2.

Technical Specification 3.6.1.1, 3.6.1.2, and 3.6.1.4 Bases (Applicable Safety Analyses) are changed based upon the containment analysis. These changes show the peak containment pressure (Pa) from the containment analysis.

Evaluation Summary

This departure which updates the containment analysis for STP 3 & 4 does not affect Tier 1, Tier 2*, or any operational requirements. However, it does affect the Bases for Technical Specifications 3.6.1.1, 3.6.1.2, and 3.6.1.4 ~~and 3.6.1.6~~ and therefore requires NRC approval.

There is no impact on environmental qualification of equipment due to the higher predicted drywell temperatures and pressures. The qualification of equipment is based on the containment design pressures and temperatures. The calculated containment pressure and temperature for both the FWLB and MSLB remain below the design values except for a two second period when the drywell temperature exceeds the design temperature by 2.1°C for the MSLB. Due to thermal inertia, components in the drywell would not have sufficient time to reach the design limit temperature in such a short time.

The change in the design suppression pool temperature limit is to align the limit with the NPSH calculation assumptions to determine that adequate NPSH exists for the ECCS pumps. These calculations represent the limiting condition for determining maximum allowable suppression pool temperature. These calculations use a suppression pool temperature of 100°C. Therefore, the allowable design limit for the peak suppression pool temperature is being changed to 100°C.

The change in the assumed elapsed time between start of LOCA and initiation of suppression pool cooling and containment sprays from 10 minutes to 30 minutes is conservative relative to the DCD as heat removal from these systems is not credited until later in the accident sequence. This also brings the assumption for operator action into alignment with current safety analysis practices.

This departure was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

(1) This exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. The design change and revised containment analysis represents an improvement and therefore will not present an undue risk to the public health and safety. The change does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstance (iv) applies in that this represents a benefit in public health and safety. The ~~more advanced and complete analysis methods~~ incorporation of these modeling changes as well as the use of an analysis method

which has been baselined to the certified DCD analysis method provide a more accurate prediction of peak containment conditions post-accident. These results show that the peak containment pressure ABWR and temperature conditions calculated following an accident based on these improved analyses are ~~below the design limits~~ acceptable. The FWLB mitigation, while not specifically credited in the containment analysis, will provide added assurance that the revised containment analysis results will remain conservative when detailed feedwater and condensate system design and procurement work is completed.

As discussed above, the change satisfies the exemption criteria per the requirements in 10 CFR 52 Appendix A Section VIII.C.4.

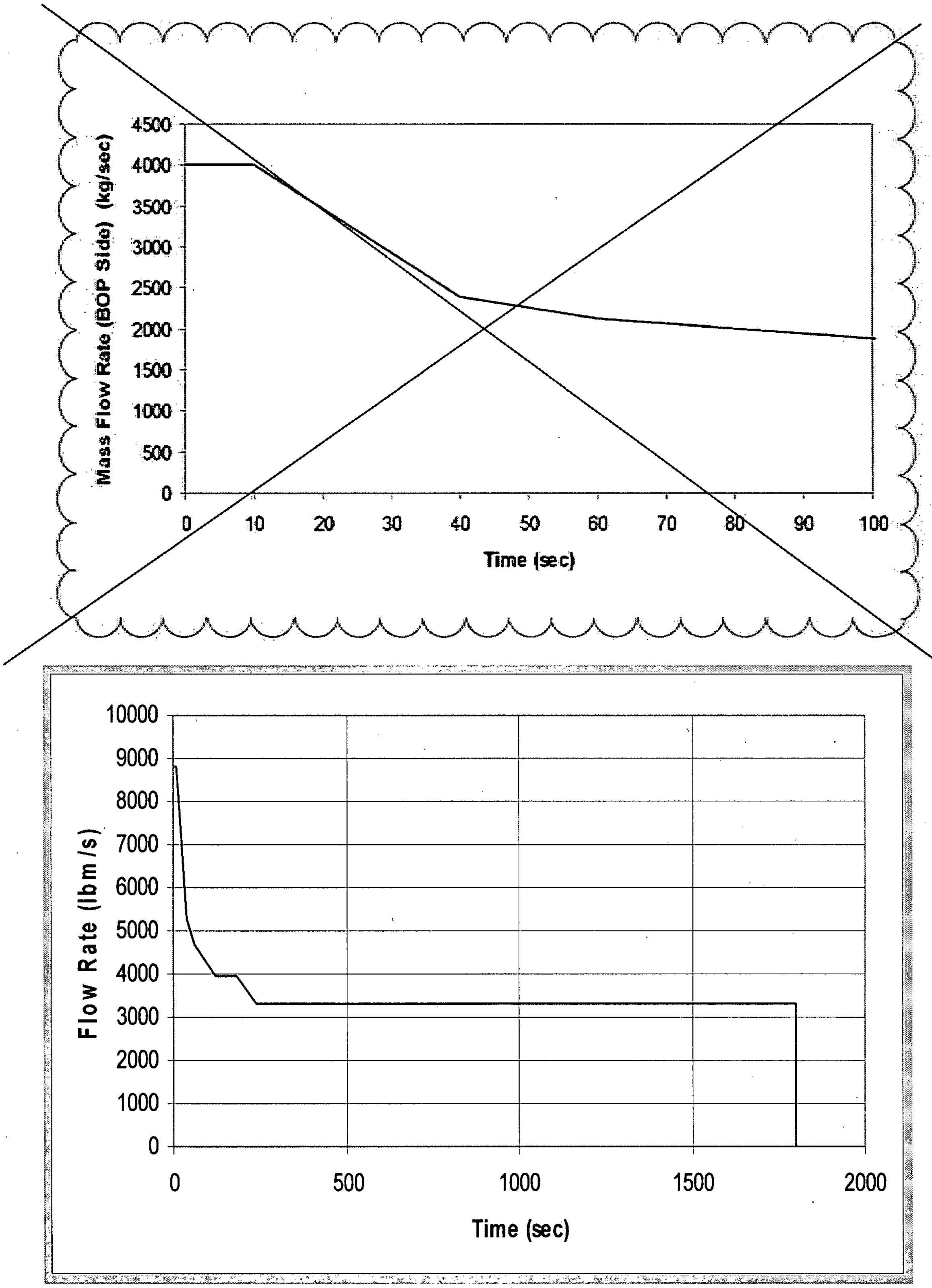
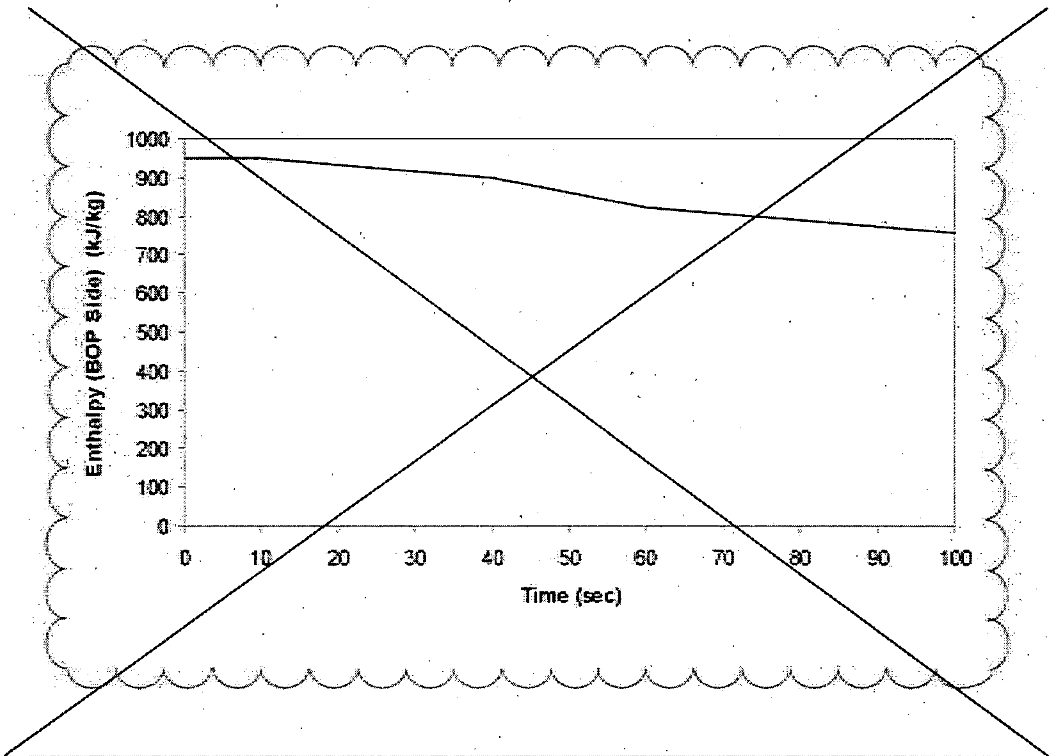


Figure 6.2-3 Feedwater Line Break Flow – Feedwater System Side of Break



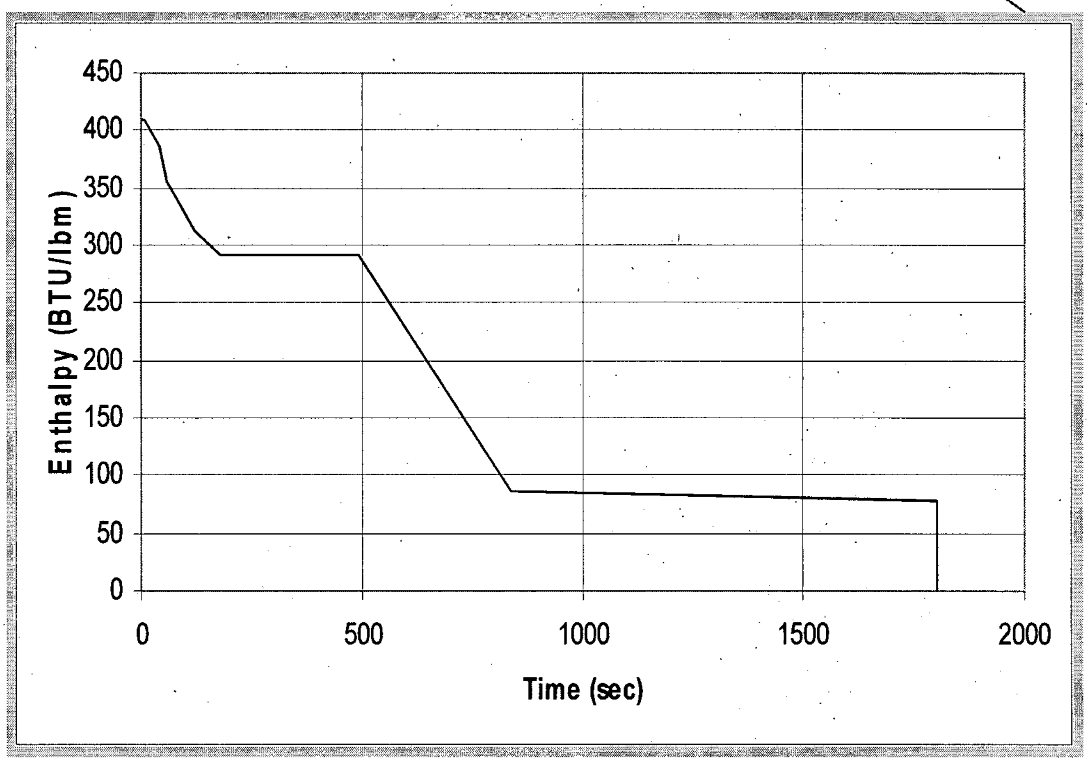
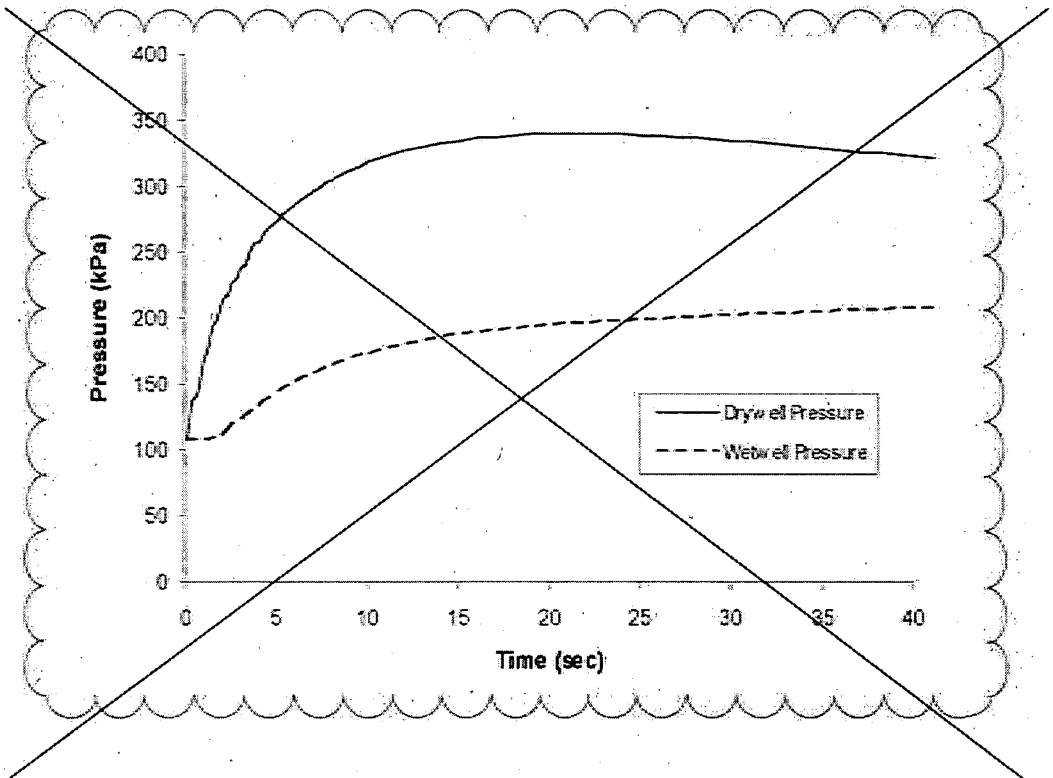


Figure 6.2-4 Feedwater Line Break Flow Enthalpy – Feedwater System Side of Break



~~Figure 6.2-6 Pressure Response of Primary Containment for Feedwater Line Break~~

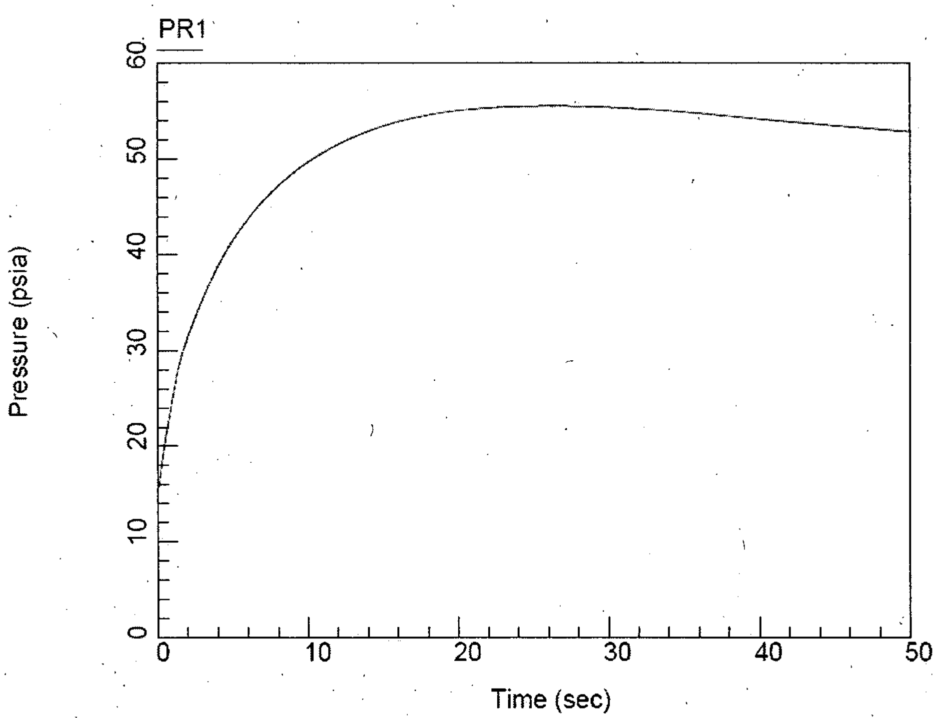


Figure 6.2-6a Drywell Pressure Response for Feedwater Line Break

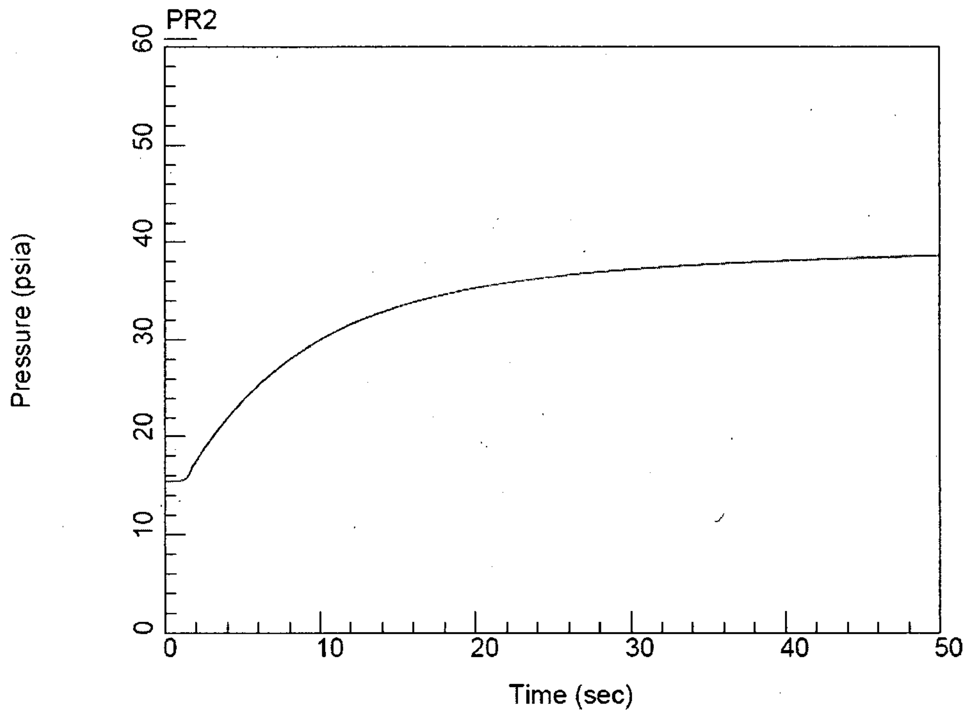
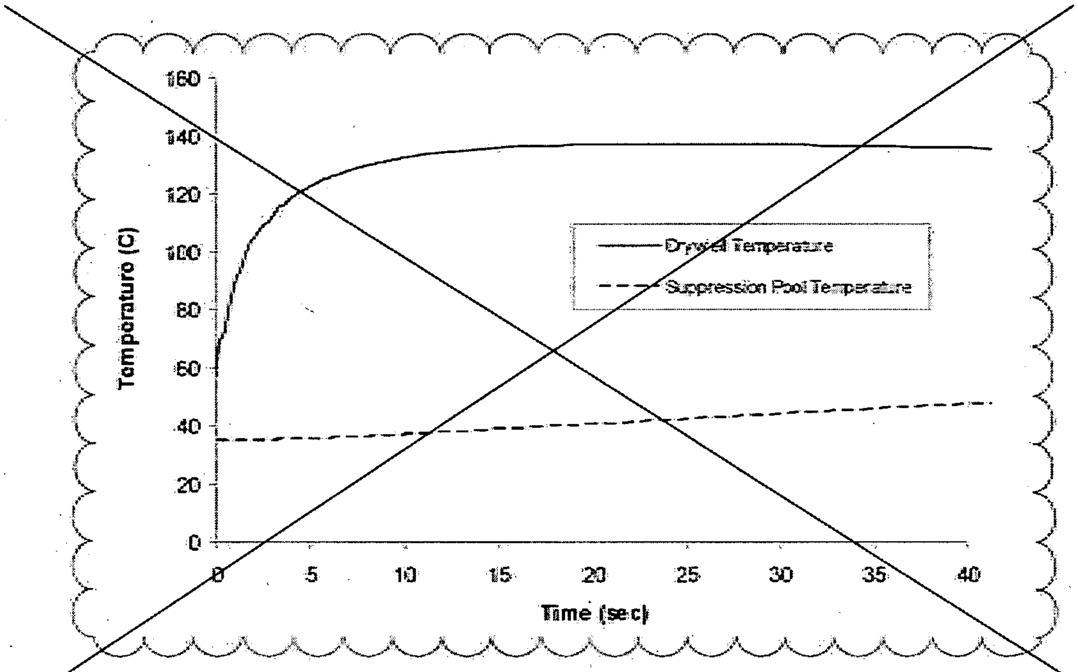


Figure 6.2-6b Wetwell Pressure Response for Feedwater Line Break



~~Figure 6.2-7 Temperature Response of Primary Containment for Feedwater Line Break~~

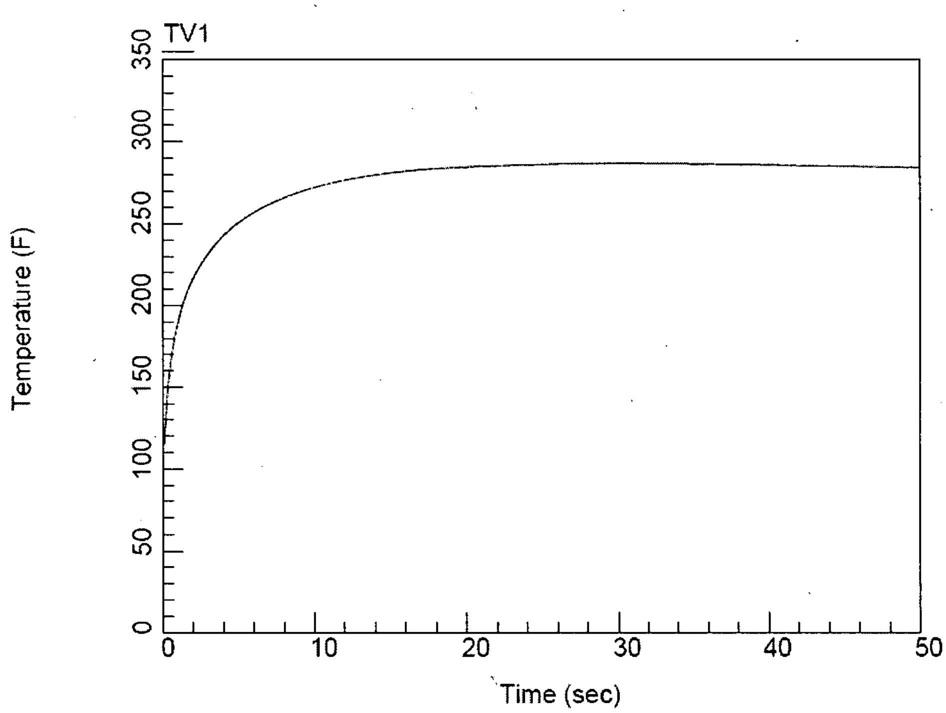


Figure 6.2-7a Temperature Response of Drywell for Feedwater Line Break

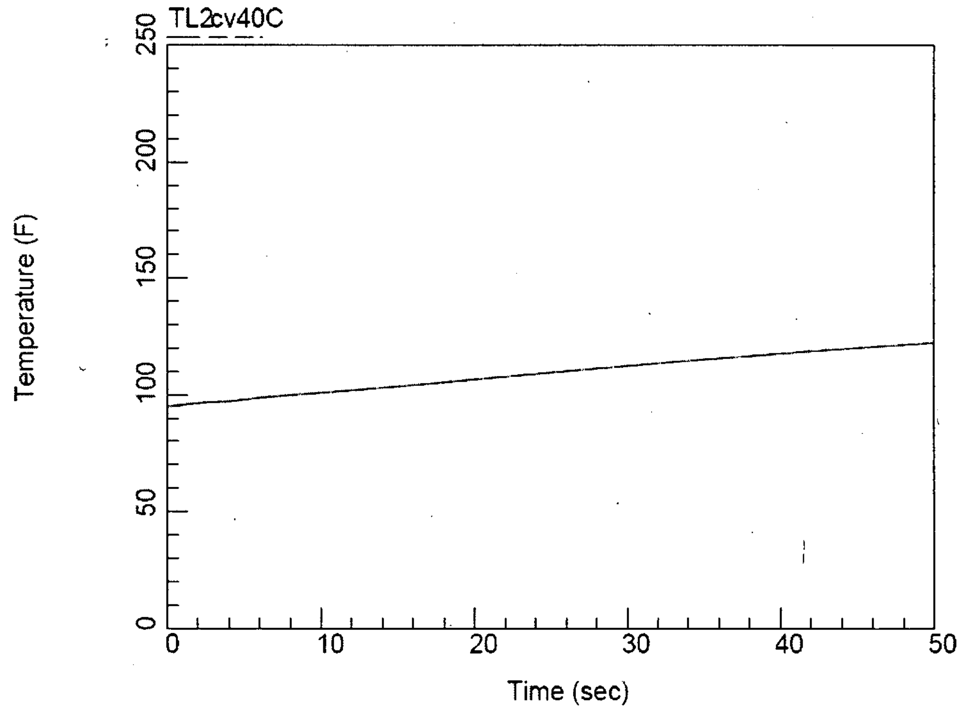
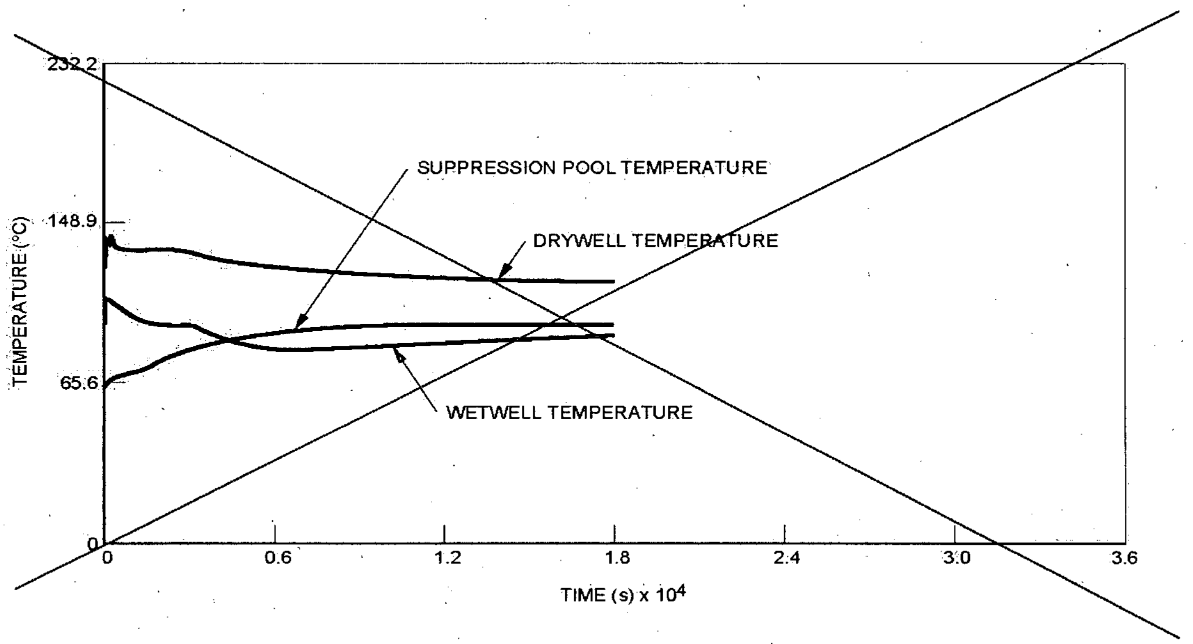


Figure 6.2-7b Temperature Response of Wetwell for Feedwater Line Break



~~Figure 6.2-8 Temperature Time History After a Feedwater Line Break~~

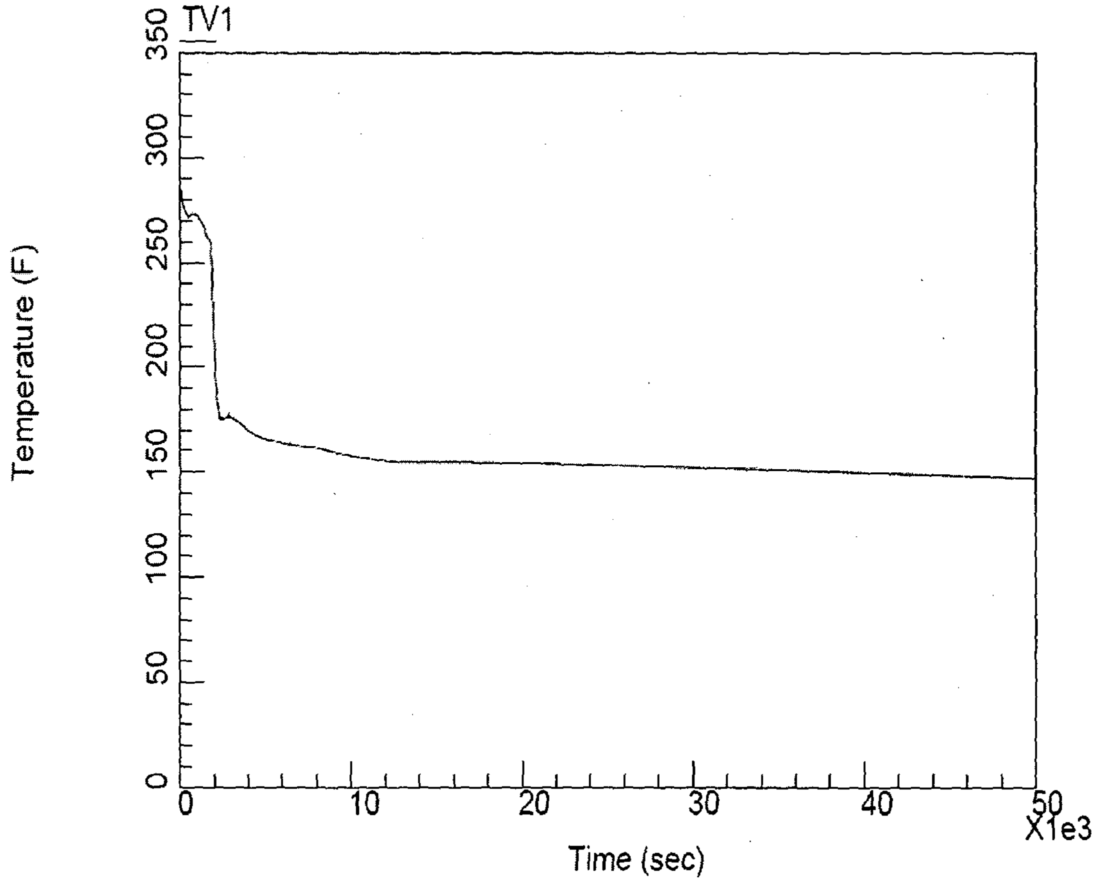


Figure 6.2-8a Drywell Temperature Time History After a Feedwater Line Break

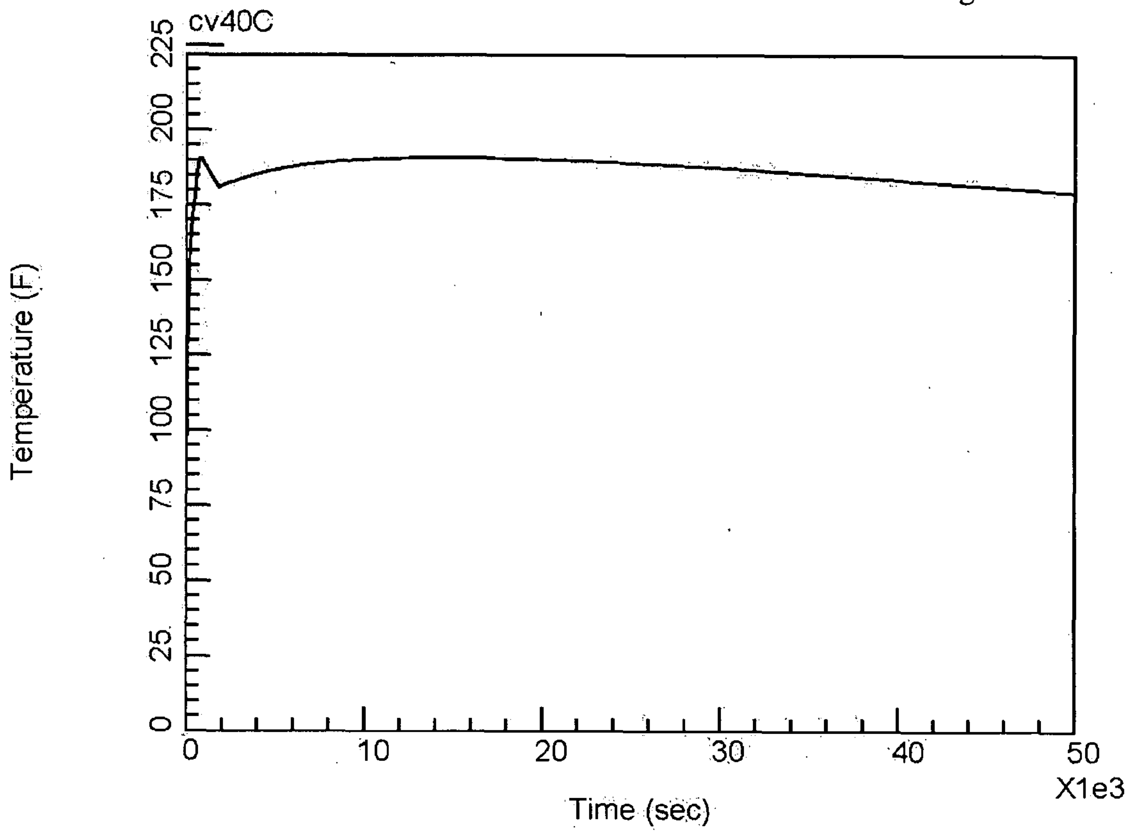


Figure 6.2-8b Suppression Pool Temperature Time History After a Feedwater Line Break

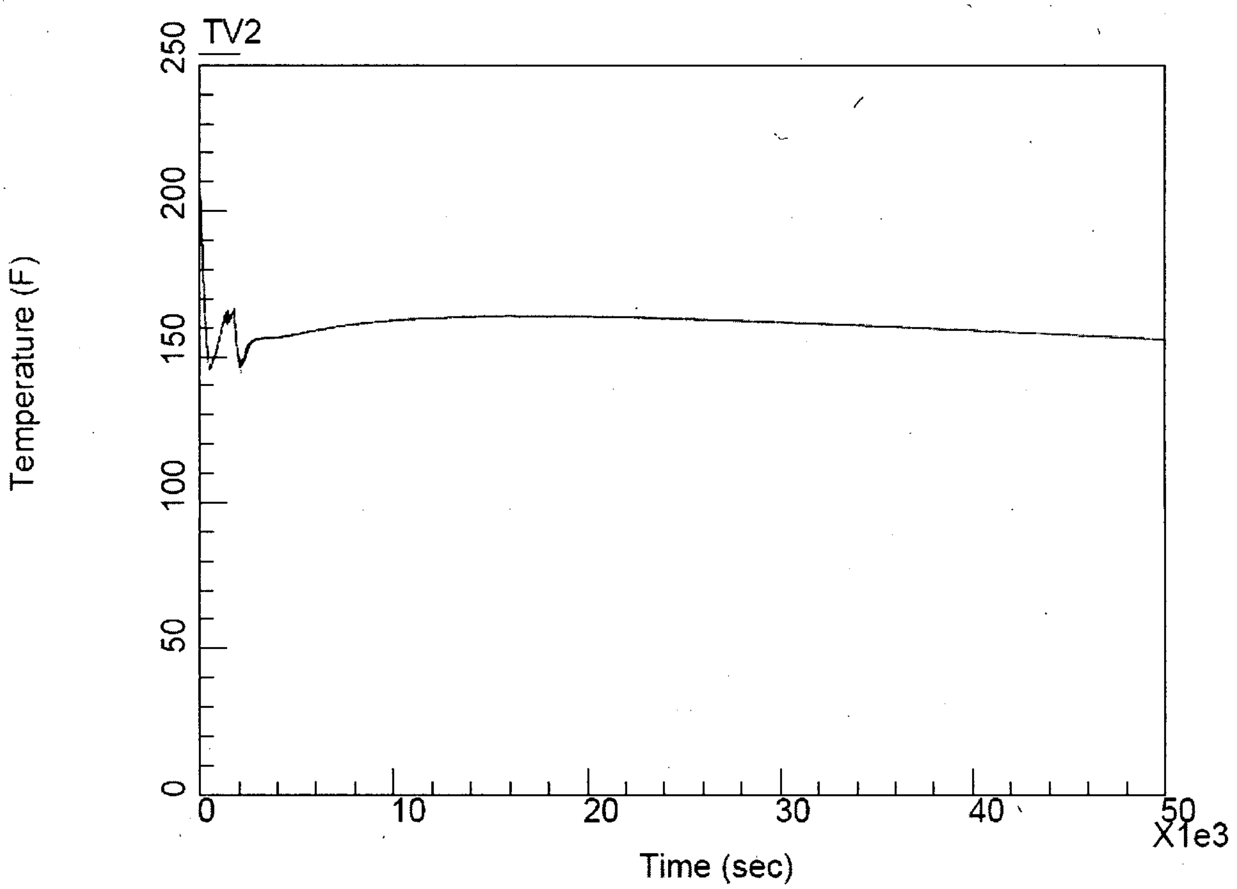
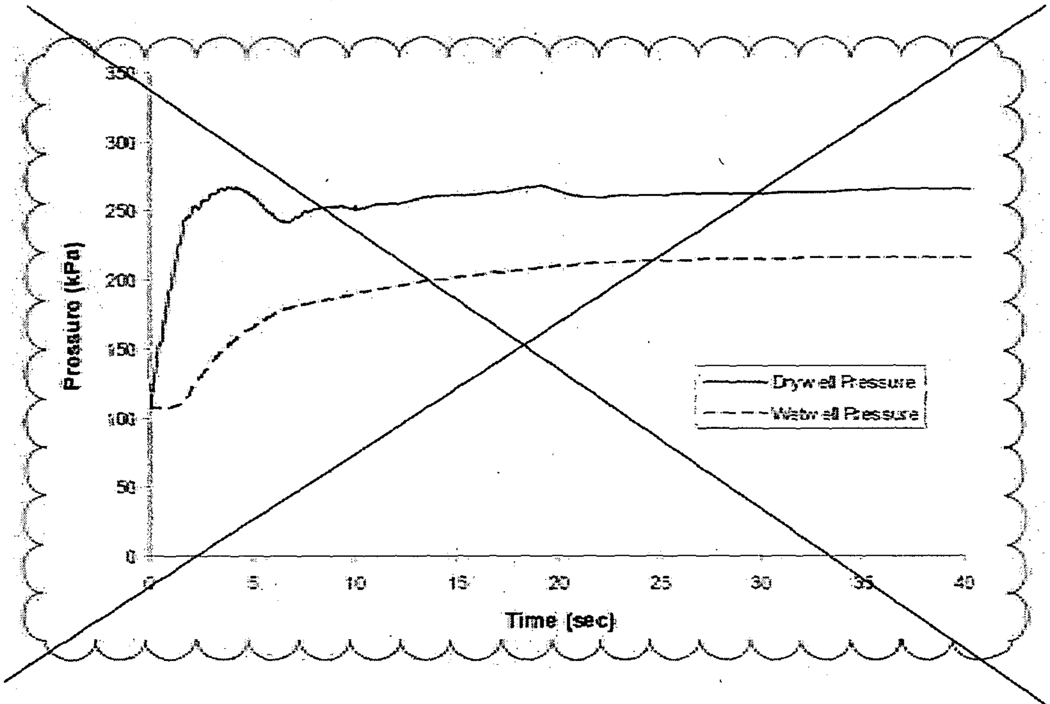


Figure 6.2-8c Wetwell Temperature Time History After a Feedwater Line Break



~~Figure 6.2-12 Pressure Time History for MSLB~~

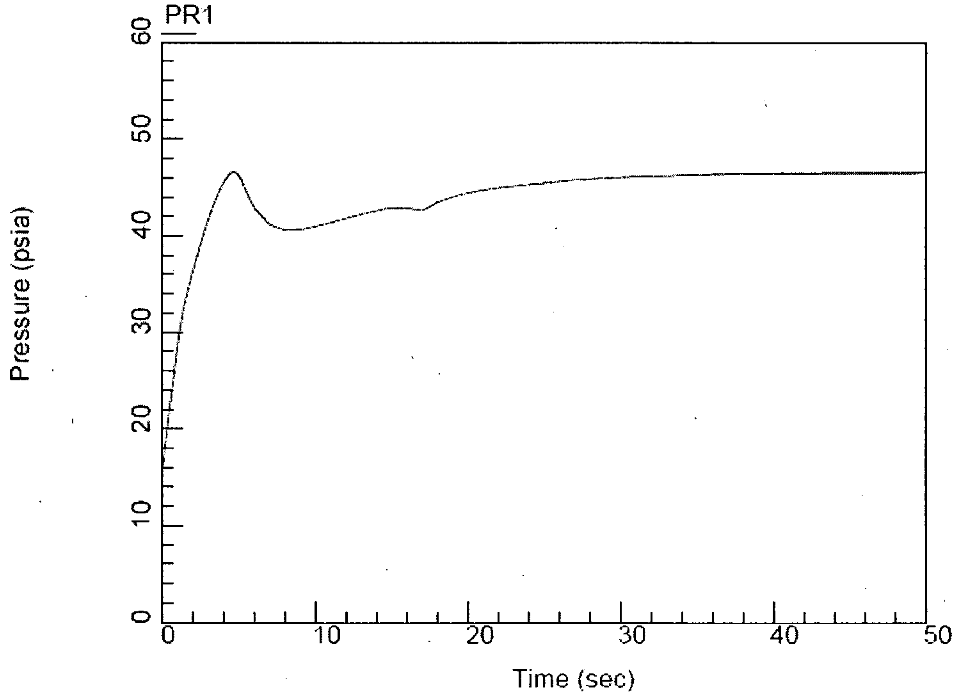


Figure 6.2-12a Drywell Pressure Time History for MSLB

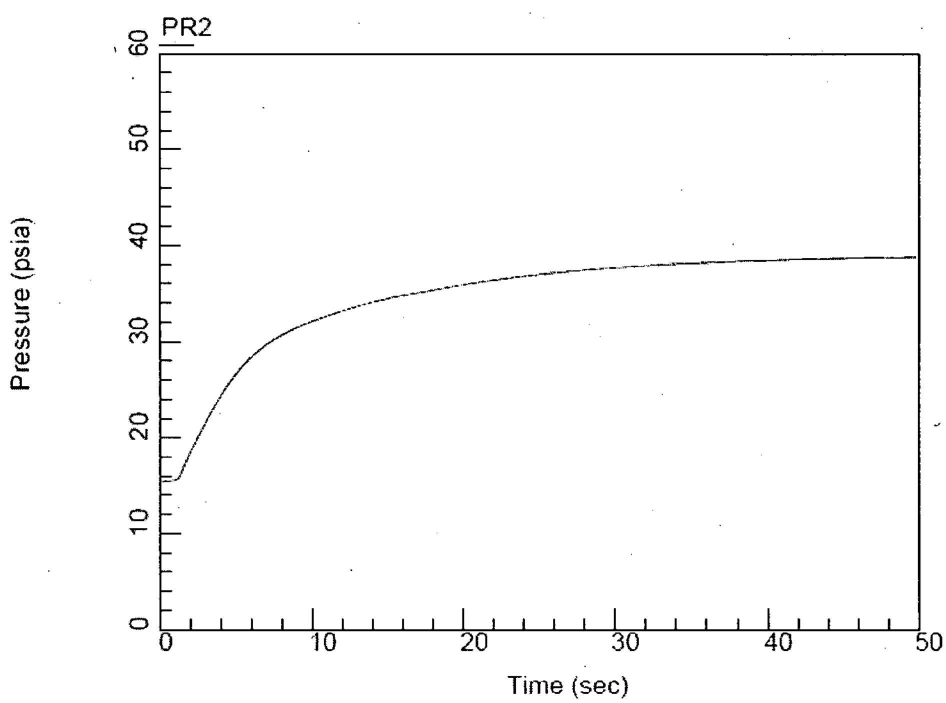
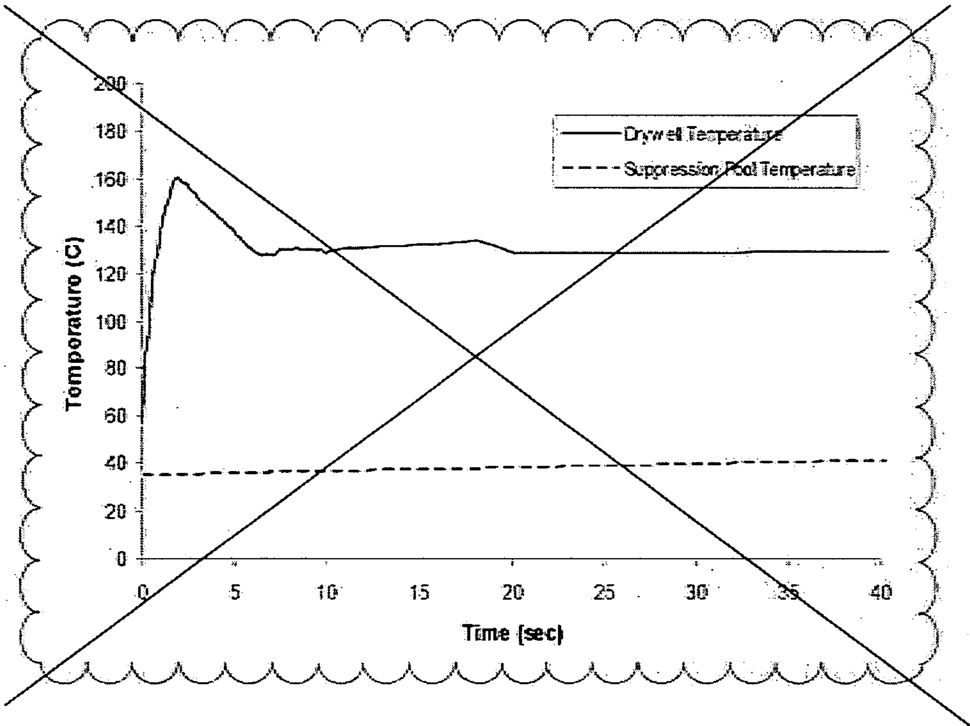


Figure 6.2-12b Wetwell Pressure Time History for MSLB



~~Figure 6.2-13 Temperature Time History for MSLB~~

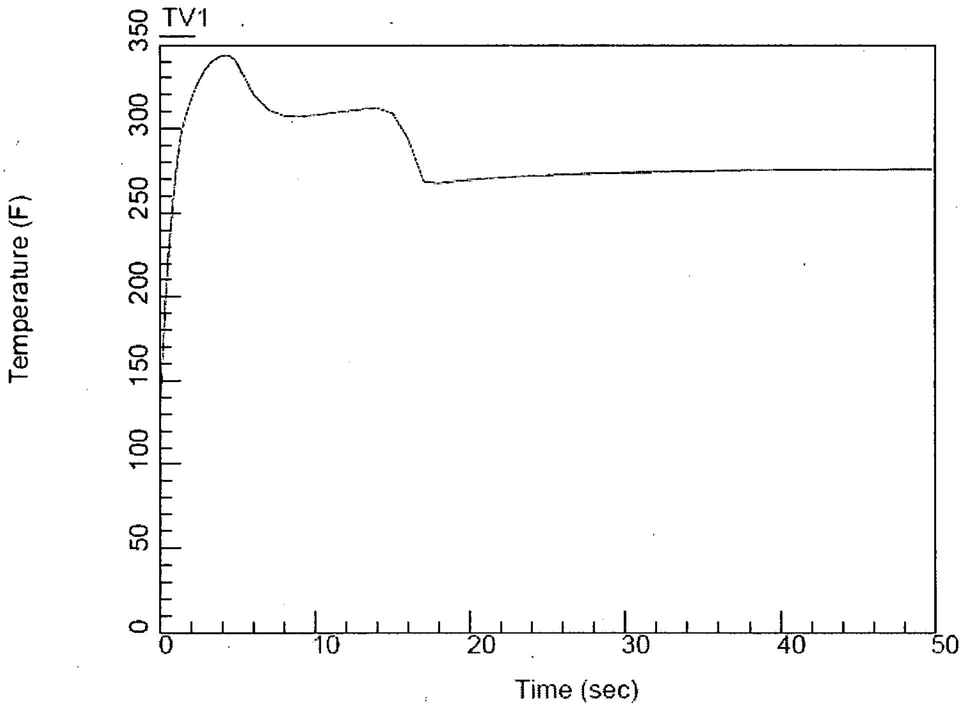


Figure 6.2-13a Drywell Temperature Time History for MSLB

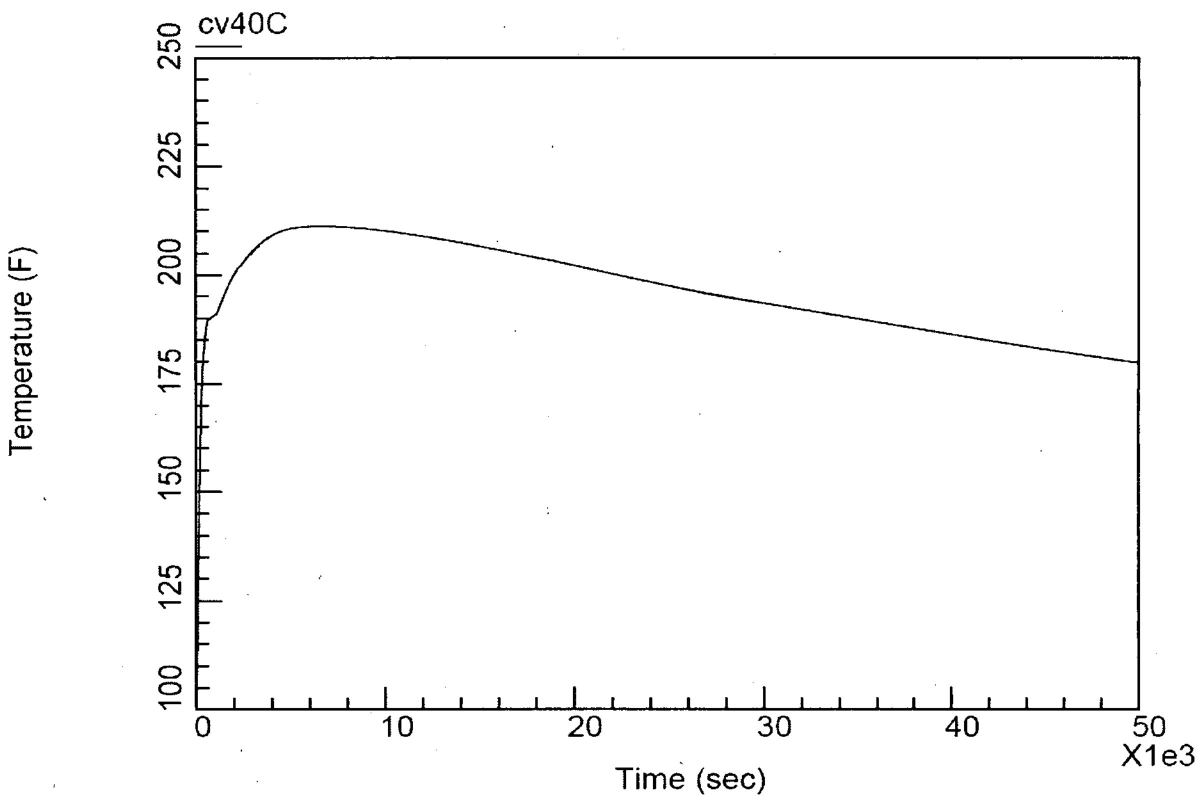


Figure 6.2-13b Suppression Pool Temperature Time History for MSLB

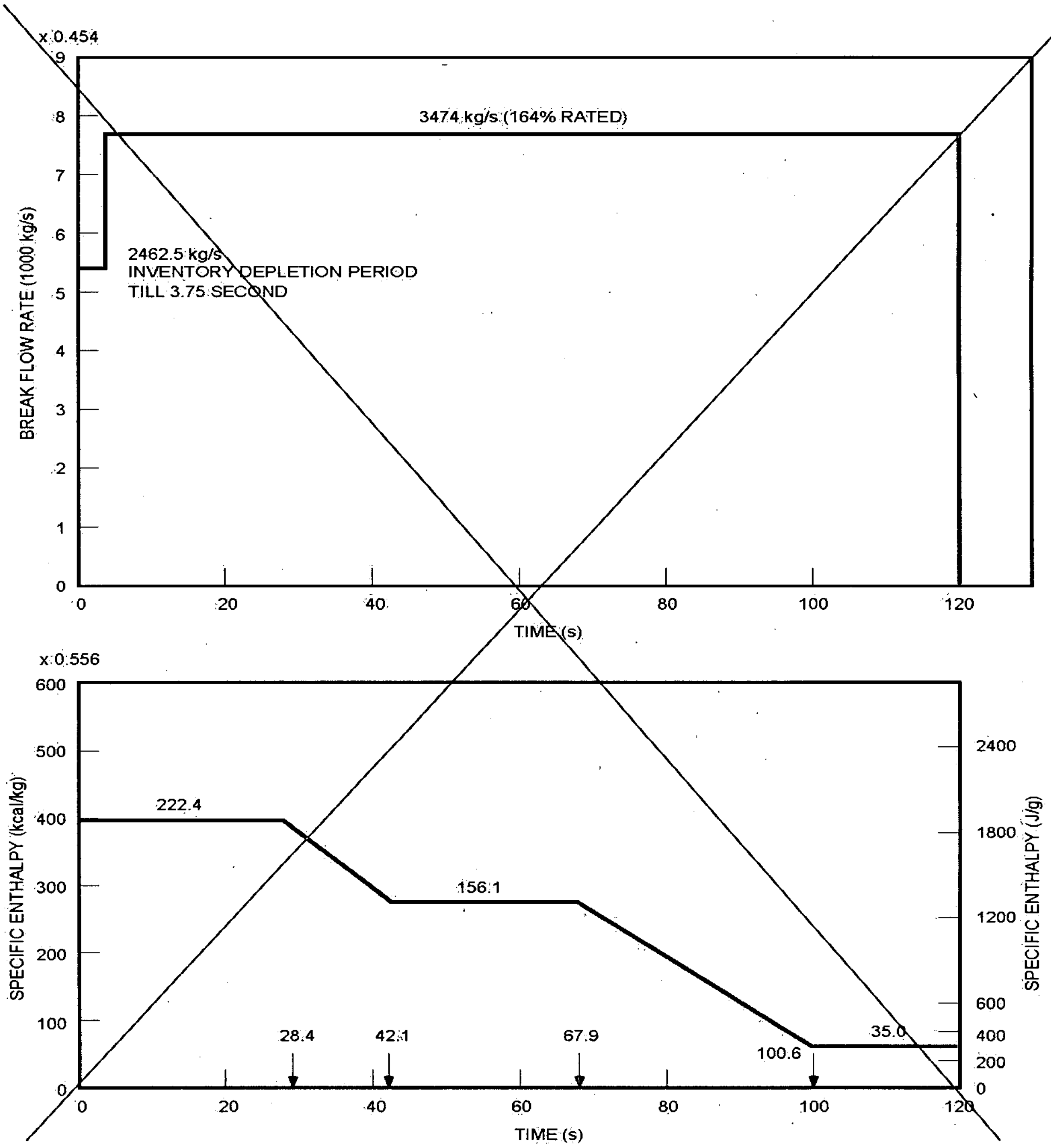
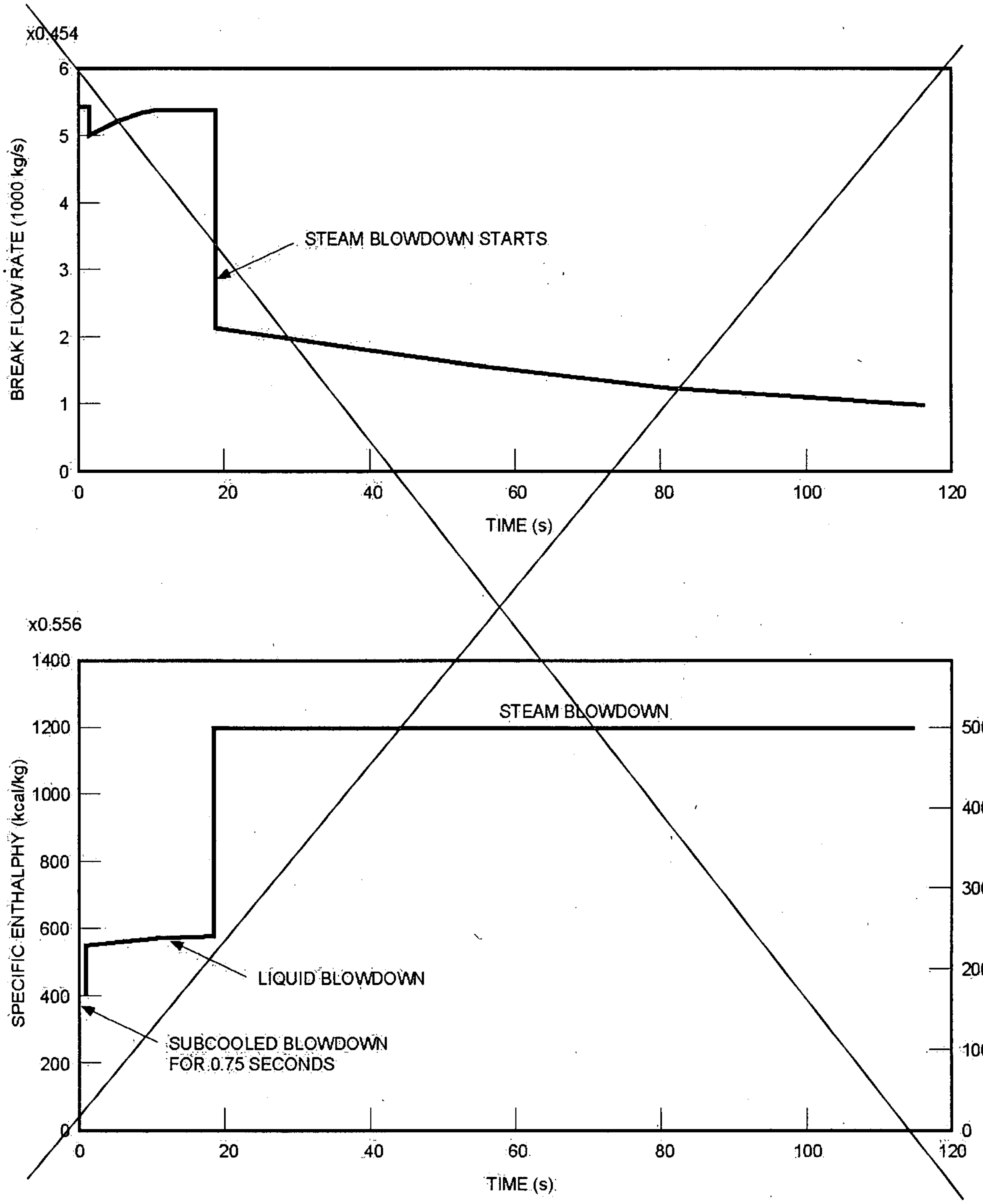


Figure 6.2-22 ~~**Break Flow Rate and Specific Enthalpy for the Feedwater Line Break Flow Coming from the Feedwater System Side**~~ **Not Used**



~~Figure 6.2-23 Break Flow Rate and Specific Entahalpy for the Feedwater Line Break Flow coming from the RPV Side~~

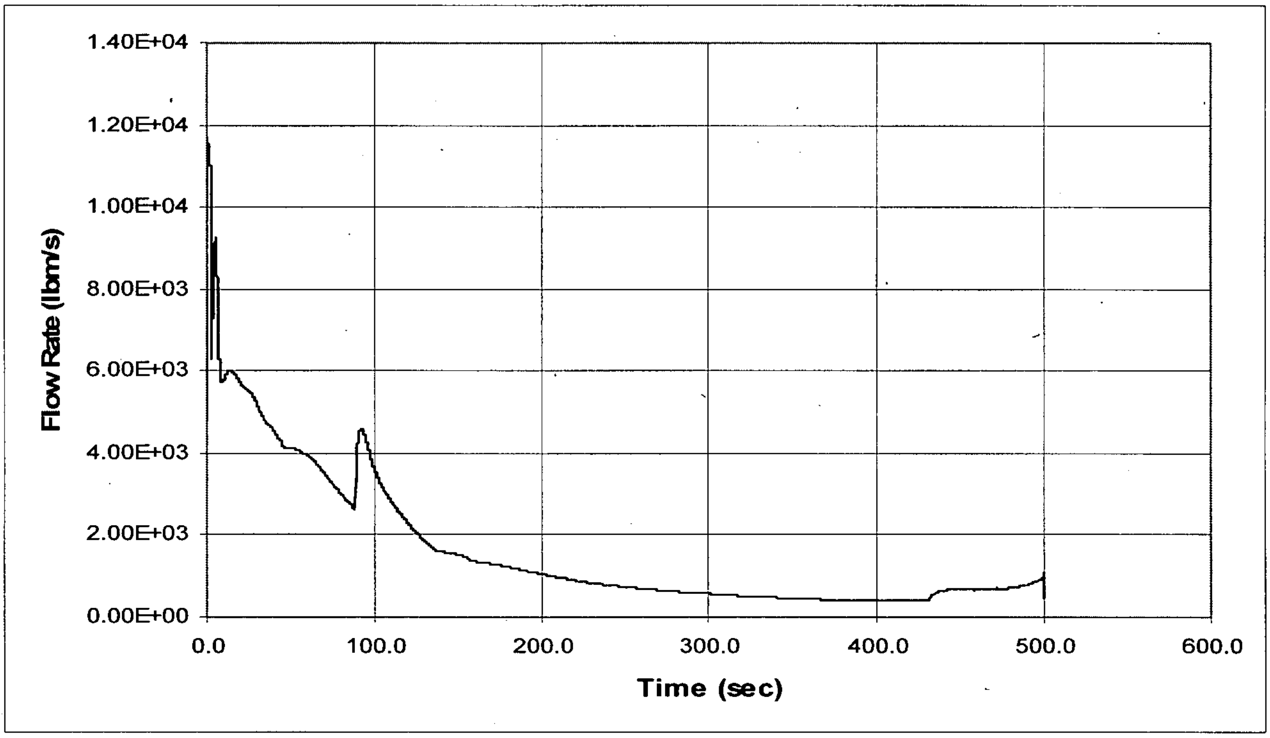


Figure 6.2-23a Break Flow Rate for the Feedwater Line Break Flow coming from the RPV Side

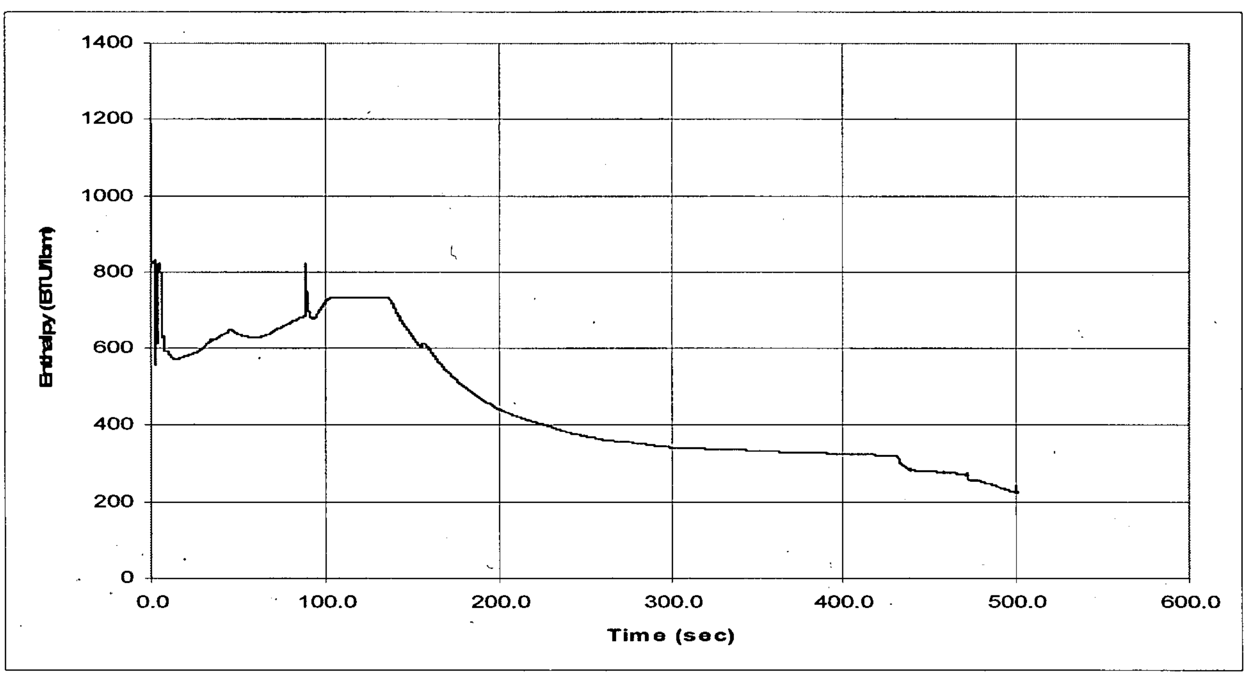


Figure 6.2-23b Break Flow Specific Enthalpy for the Feedwater Line Break Flow coming from the RPV Side

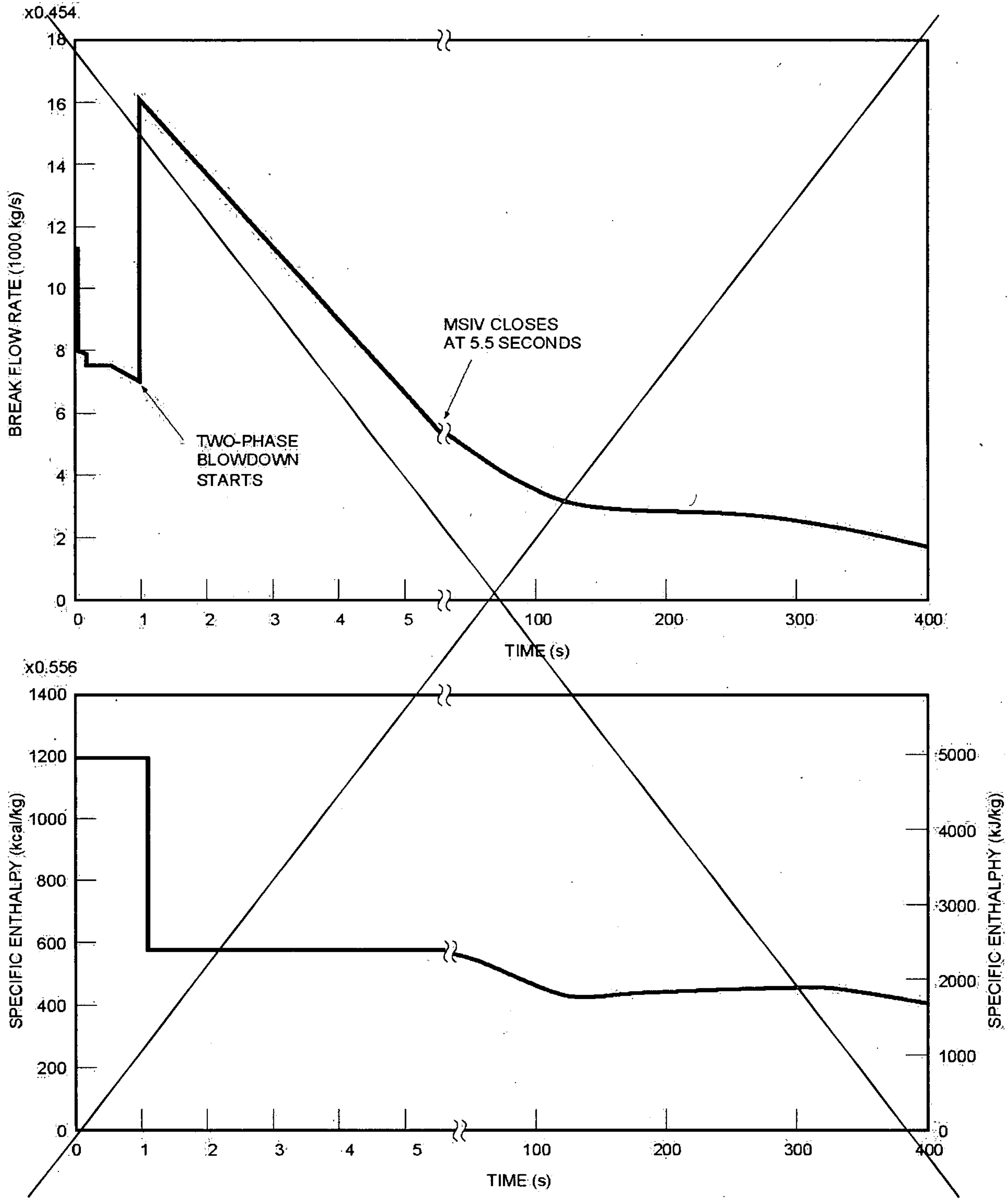
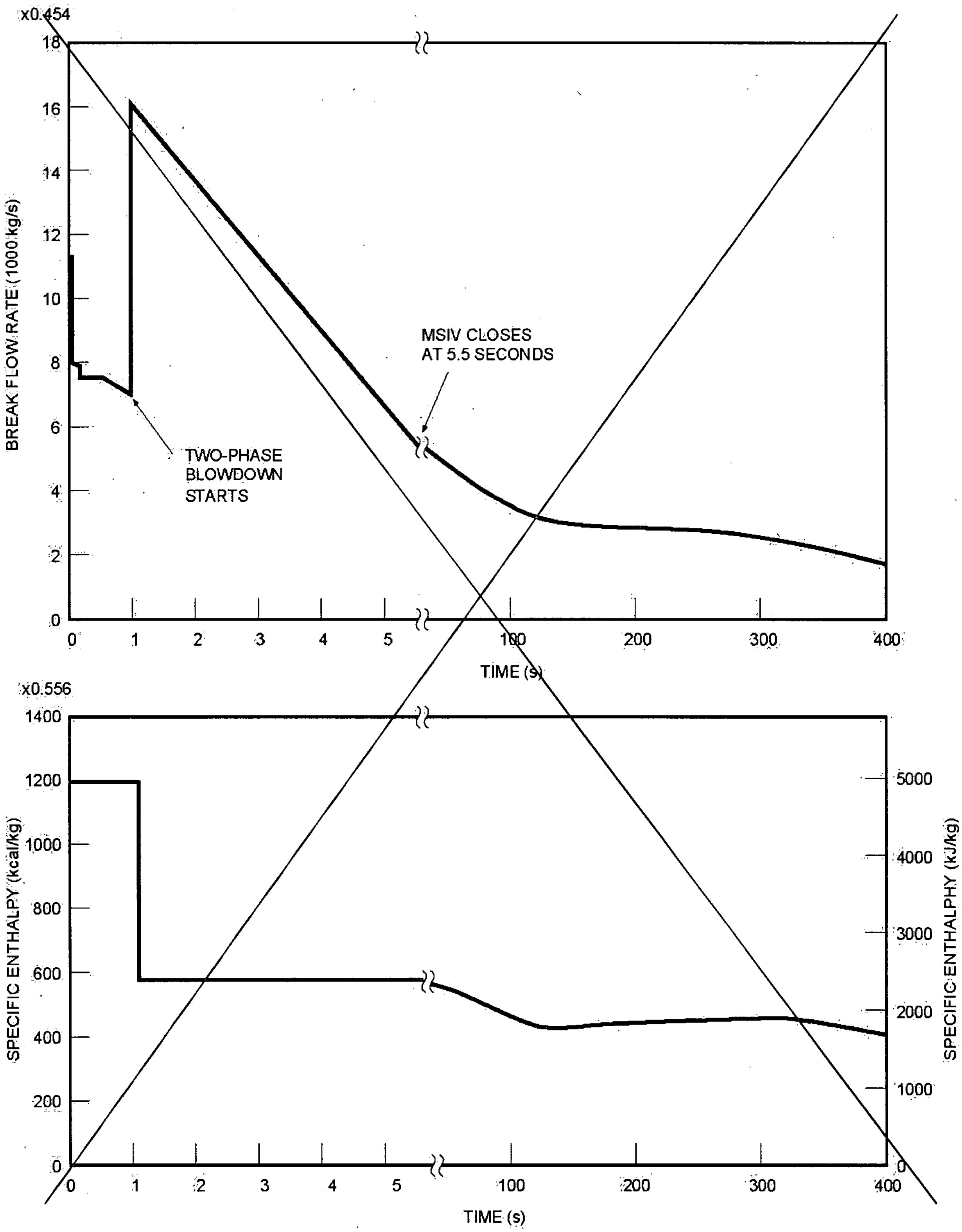


Figure 6.2-24 ~~Break Flow rate and Specific Enthalpy for the Main Steamline Break with Two-Phase Blowdown Starting Whjen the Collapsed Water Level Reaches the Steam Nozzle~~ Not Used



~~Figure 6.2-25 Break Flow Rate and specific Enthalpy for the Main Steamline break with Two-Phase Blowdown Starting at One Second~~

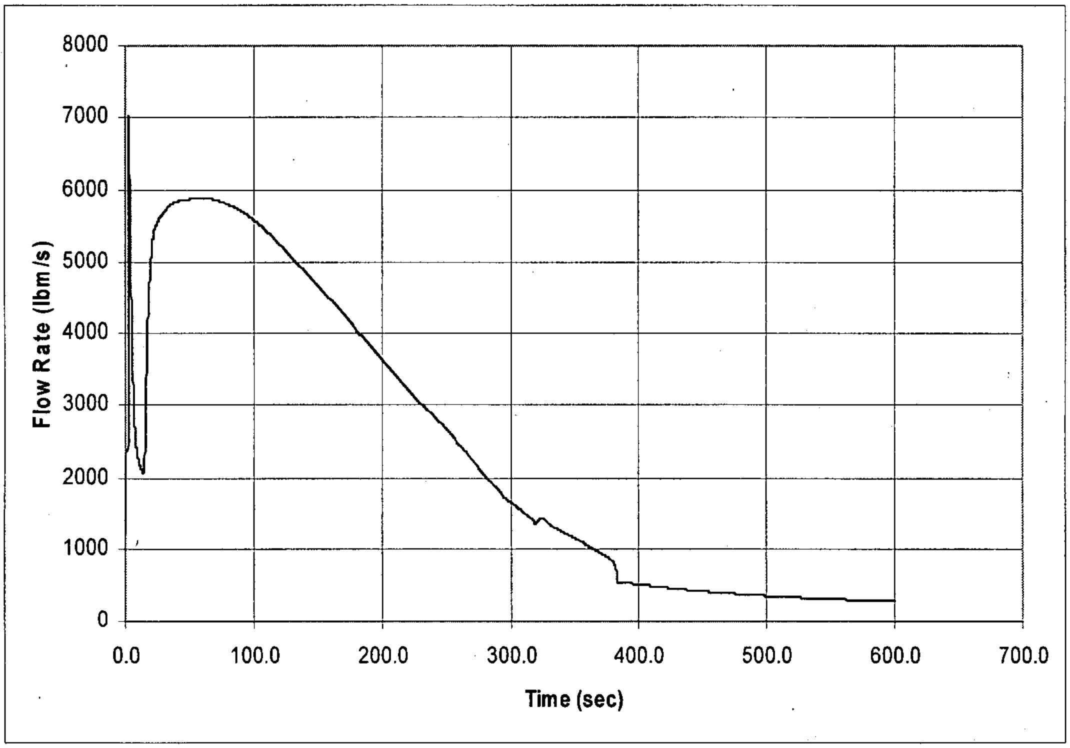


Figure 6.2-25a MSLB Short Term Break Flow (RPV Side)

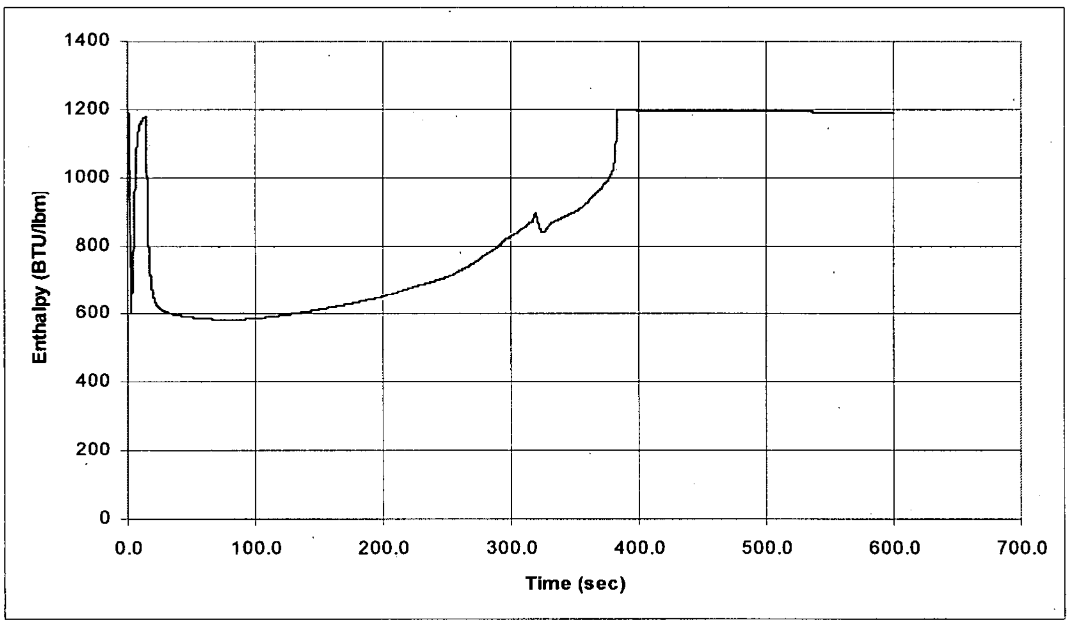


Figure 6.2-25b MSLB Short Term Break Flow Enthalpy (RPV Side)

RAI 06.02.01.01.C-2:

QUESTION:

Section 6.2.1.1.3: The staff is preparing an STP ABWR MELCOR model in support of performing independent confirmatory analysis. The following information is needed for development of the MELCOR input file:
- reactor vessel: water flow loss coefficient for the following junctions: downcomer to lower plenum, lower plenum to core channel, lower plenum to core bypass, core channel to steam separator, steam separators to downcomer,
- reactor vessel: elevation of the main feed water spargers,
- setpoint value of the Condensate Storage Tank (CST) level at which High Pressure Core Flooder (HPCF) and Reactor core Isolation Cooling (RCIC) systems suction transfer from the CST to the Suppression Pool (SP),
- setpoint value of the SP level at which HPCF and RCIC systems suction transfer from the CST to the SP,
- setpoint value of the reactor vessel pressure at which the low pressure permissive signal is generated to open the Low Pressure Core Flooder (LPCF) injection valve,
- ADS valves opening sequence after receiving the ADS initiation signal,
- a figure showing the feedwater line break flow from the feedwater system side of break (i.e., Figure 6.2-3 in STP COLA, Rev. 2 with the time axis varying from 0.0 to 5 hrs),
- a figure showing the feedwater line break flow enthalpy from the feedwater system side of break (i.e., Figure 6.2-4 in STP COLA, Rev. 2 with time axis varying from 0.0 to 5 hrs),
- a figure showing the feedwater line break flow from the RPV side of break (i.e., Figure 6.2-23 in ABWR DCD with time axis varying from 0.0 to 5 hrs),
- a figure showing the feedwater line break flow enthalpy from the RPV side of break (i.e., Figure 6.2-23 in ABWR DCD with time axis varying from 0.0 to 5 hrs),
- a figure showing the main steam line break flow from the RPV side of break (i.e., Figure 6.2-24 in ABWR DCD with time axis varying from 0.0 to 5 hrs),
- a figure showing the main steam line break flow enthalpy from the RPV side of break (i.e., Figure 6.2-24 in ABWR DCD with time axis varying from 0.0 to 5 hrs),
- a figure showing the main steam line break flow from the piping side of break (0 to 5 hrs),
- a figure showing the main steam line break flow enthalpy from the piping side of break (0 to 5 hrs),
- a figure showing the feedwater flow rate and enthalpy assumed for the MSLB accident analysis as described in section 6.2.1 of STP COLA, Rev. 2.

RESPONSE:

The requested input parameter information (bullets 1-6) was provided to NRC in STP Letter U7-C-STP-NRC-090014 dated February 19, 2009.

The requested figures are provided with this response as follows:

- Figure 1 shows the feedwater line break flow from the feedwater system side of the break. As shown in the figure, the feedwater flow terminates at 30 minutes, thus the flow from 30 minutes to the requested 5 hour time is zero.
- Figure 2 shows the feedwater line break flow enthalpy from the feedwater system side of the break. As shown in Figure 1, the feedwater flow terminates at 30 minutes, thus the flow enthalpy from 30 minutes to the requested 5 hour time is zero.
- Figure 3a shows the short-term feedwater line break flow from the RPV side of the break from 0 to 500 seconds.
- Figure 3b shows the long-term feedwater line break flow from the RPV side of the break from 500 seconds to 5 hours.
- Figure 4a shows the short-term feedwater line break flow enthalpy from the RPV side of the break from 0 to 500 seconds.
- Figure 4b shows the long-term feedwater line break flow enthalpy from the RPV side of the break from 500 seconds to 5 hours.
- Figure 5a shows the short-term main steam line break flow from the RPV side of the break from 0 to 600 seconds.
- Figure 5b shows the long-term main steam line break flow from the RPV side of the break from 600 seconds to 5 hours.
- Figure 6a shows the short-term main steam line break flow enthalpy from the RPV side from 0 to 600 seconds.
- Figure 6b shows the long-term main steam line break flow enthalpy from the RPV side from 600 seconds to 5 hours.
- Figure 7 shows the main steam line break flow from the piping side of the break from 0 to 5.6 sec, at which time the flow is zero.
- Figure 8 shows the main steam line break flow enthalpy from the piping side of the break from 0 to 10 sec, at which time the flow is zero.
- Figure 9 is the feedwater line flow and enthalpy from 0 to 600 sec assumed for the MSLB accident from 0 to 600 sec.

No COLA revision is required as a result of this RAI response.

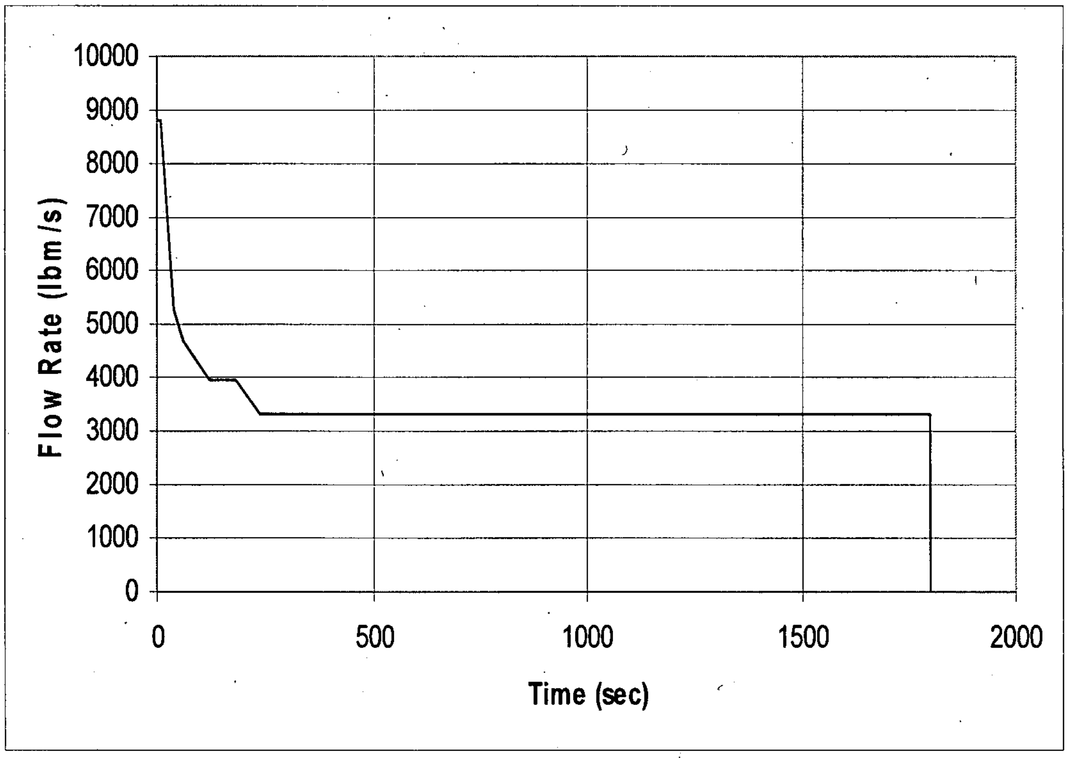


Figure 1 – FWLB Break Flow Rate (Feedwater Side)

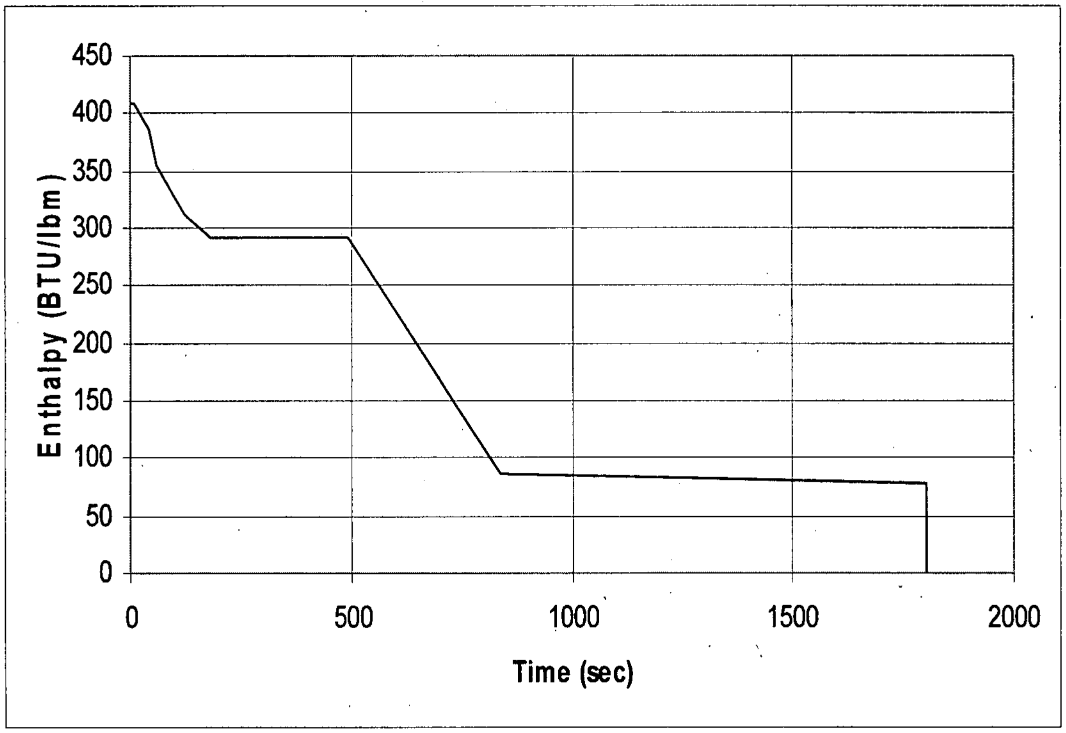


Figure 2 FWLB Enthalpy (Feedwater Side)

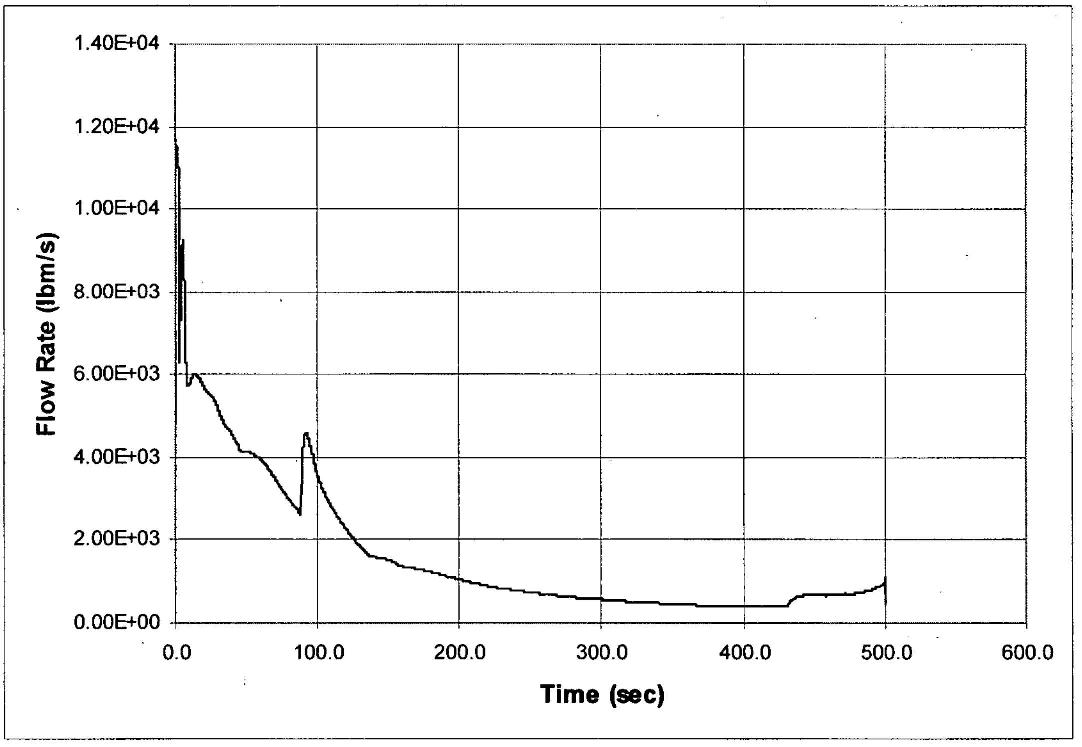


Figure 3a – FWLB Short Term Break Flow Rate (RPV Side)

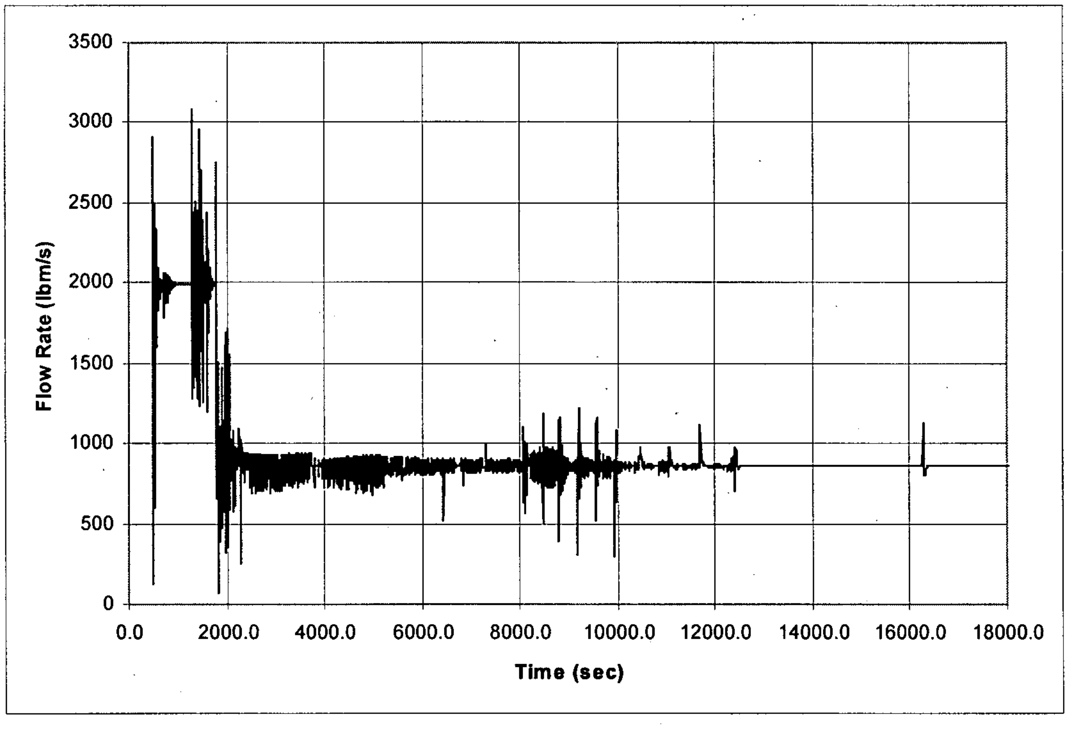


Figure 3b FWLB Long Term Break Flow Rate (RPV Side)

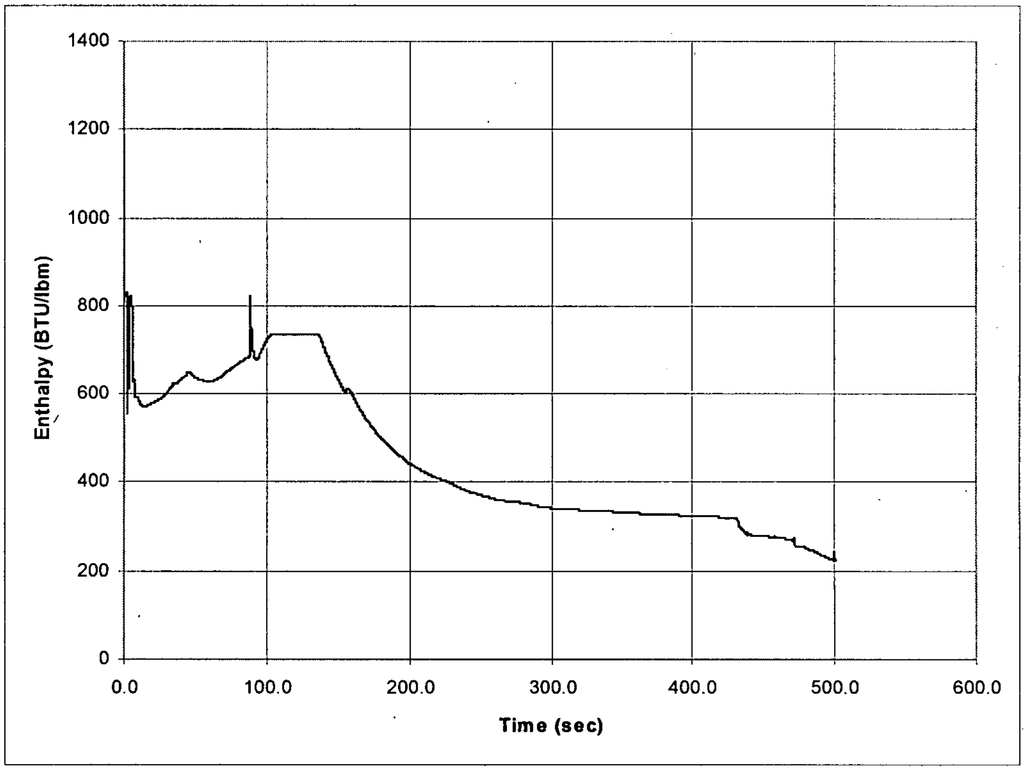


Figure 4a – FWLB Short Term Enthalpy (RPV Side)

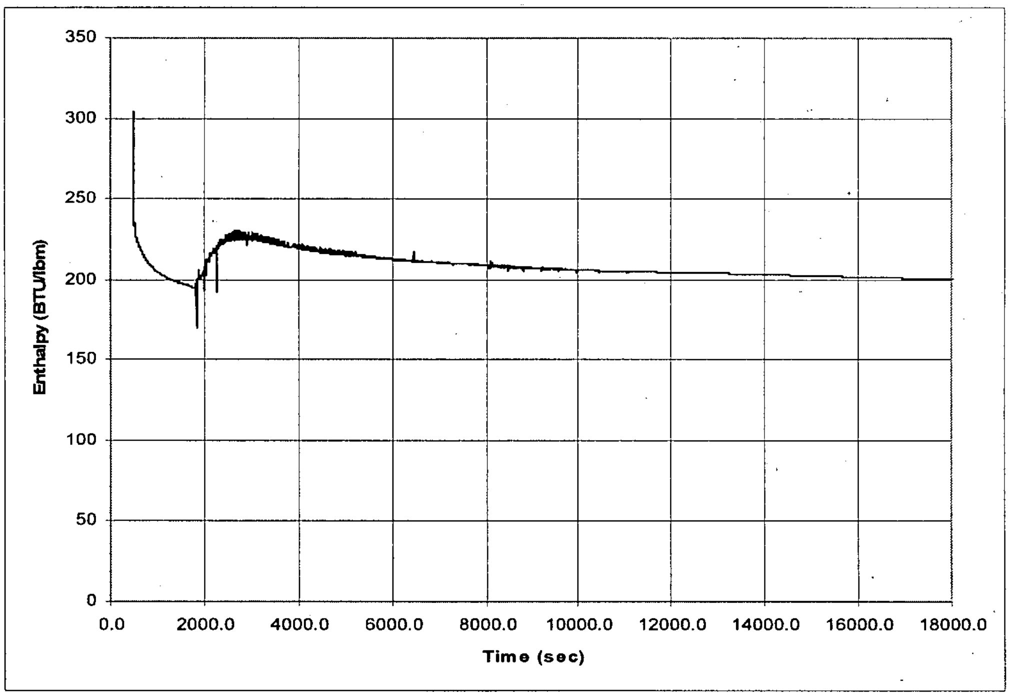


Figure 4b FWLB Long Term Enthalpy (RPV Side)

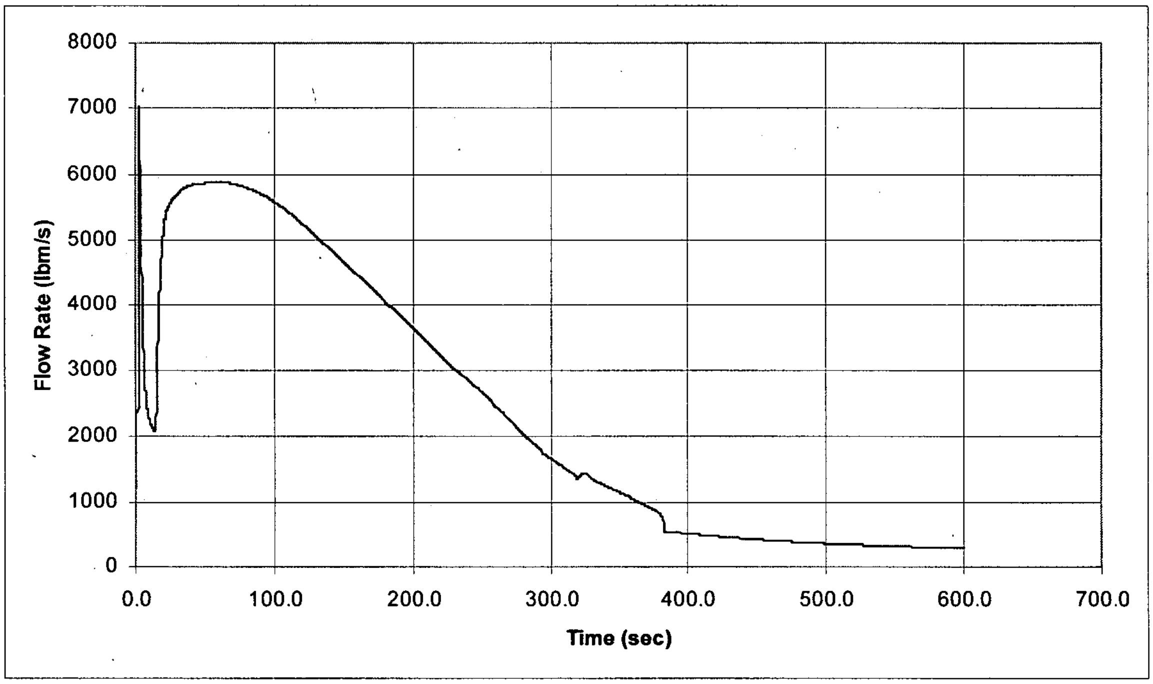


Figure 5a – MSLB Short Term Break Flow Rate (RPV side)

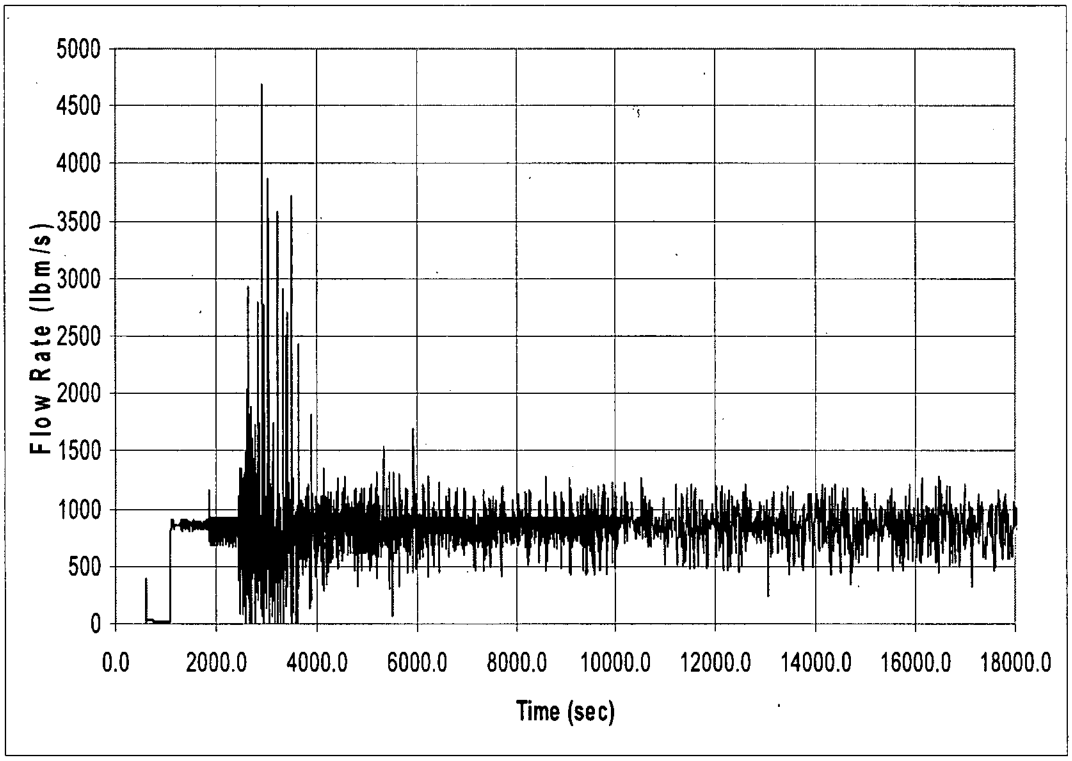


Figure 5b MSLB Long Term Break Flow Rate (RPV Side)

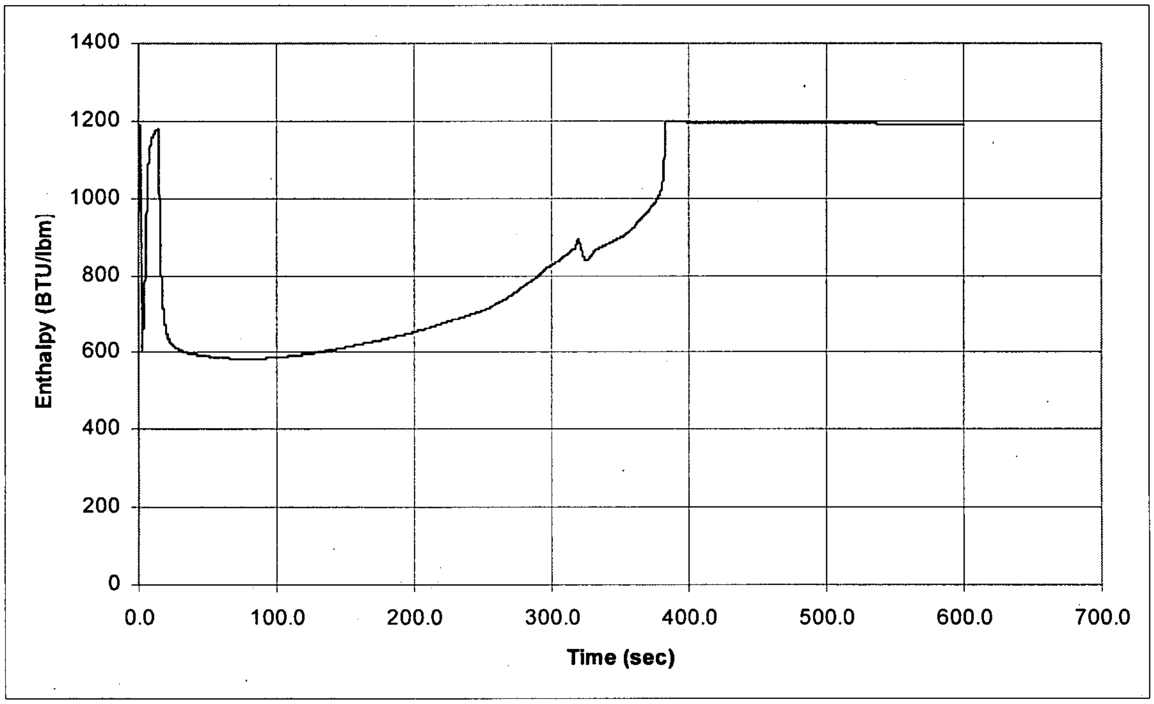


Figure 6a – MSLB Short Term Enthalpy (RPV side)

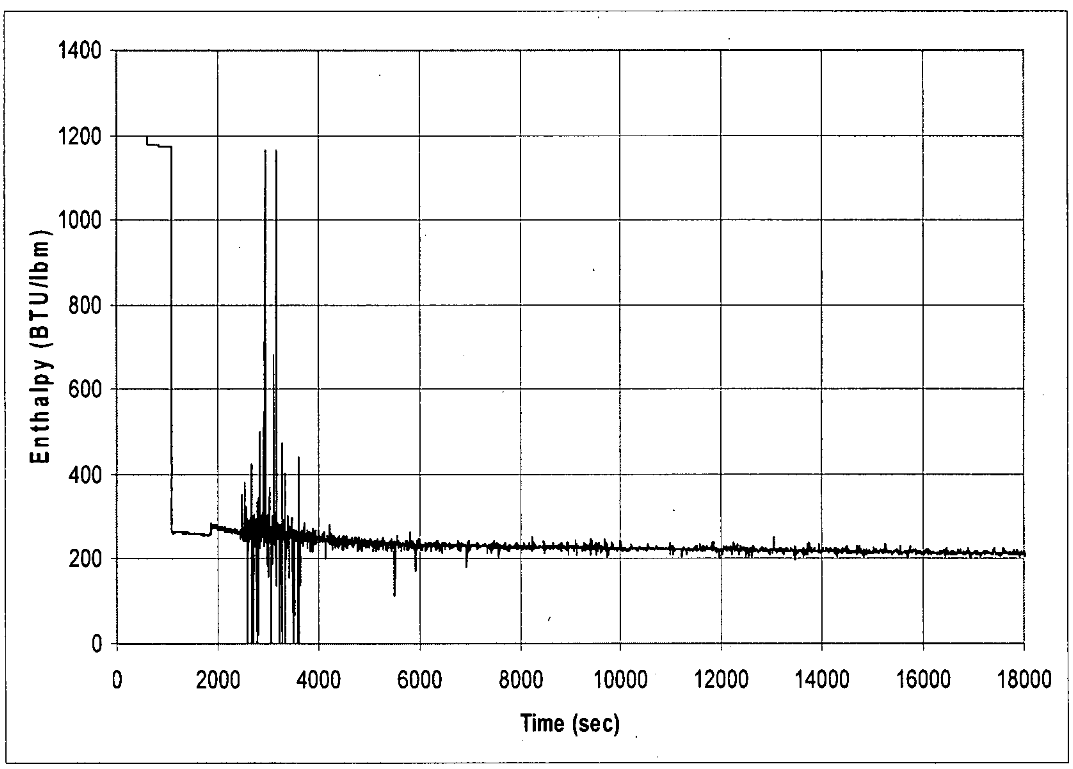


Figure 6b MSLB Long Term Enthalpy (RPV Side)

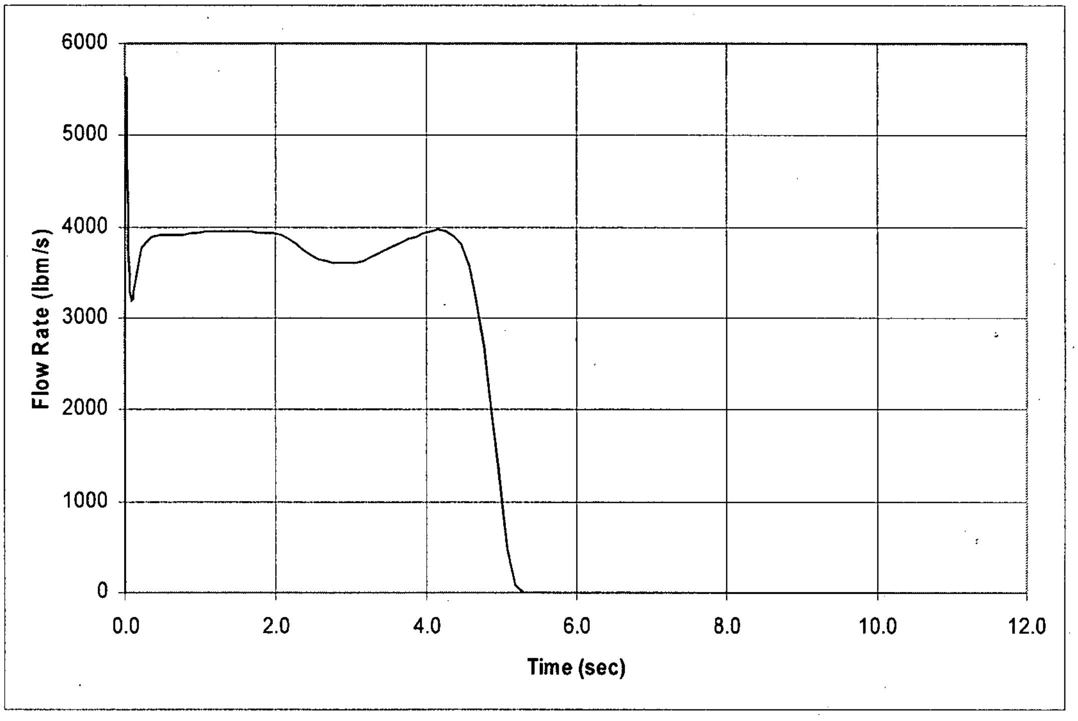


Figure 7 MSLB Break Flow Rate (Piping Side)

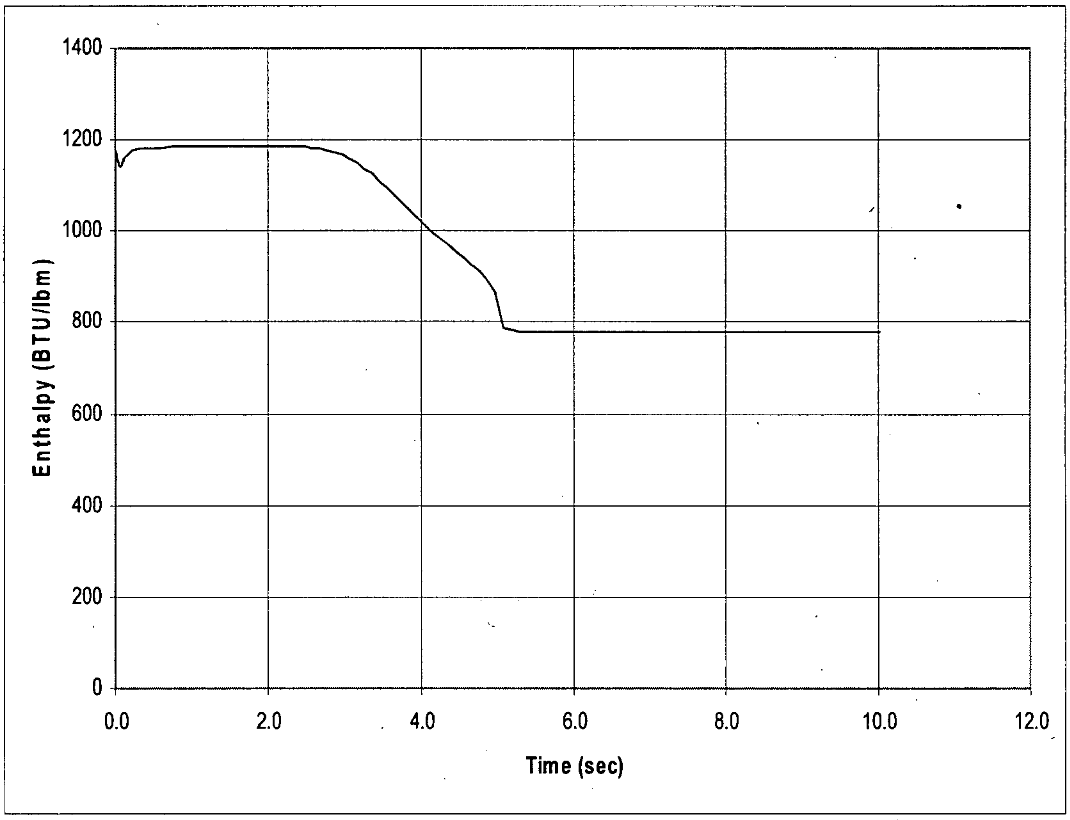


Figure 8 MSLB Enthalpy (Piping Side)

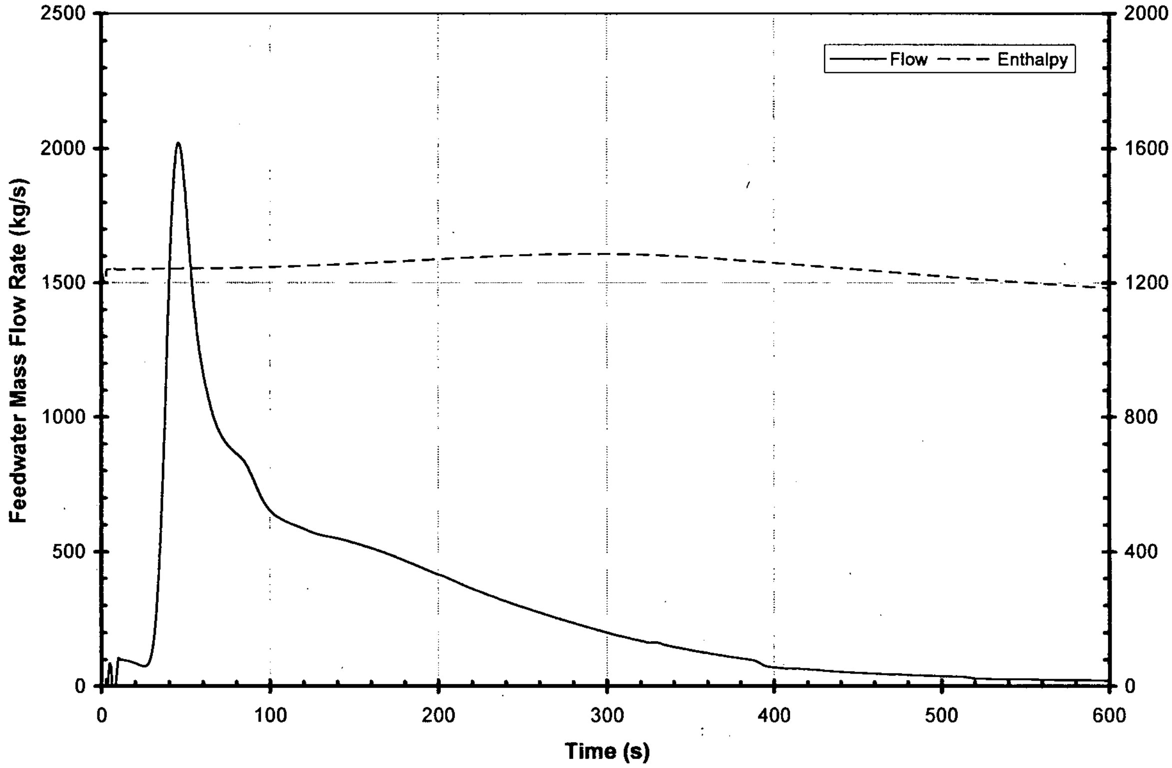


Figure 9 MSLB Feedwater Flow Rate and Enthalpy

RAI 06.02.01.01.C-8

QUESTION:

6.2.1.1.1 Design Basis - **Supplement to RAI 06.02.01.01.C-2 :** In support of performing independent confirmatory analysis, the following are requested additional information:

(a) Reactor Pressure Vessel
- Core channel flow rate during full power operation
- Core bypass flow rate during full power operation
- Loss coefficients for steam/water flow through the feedwater sparger and feedwater nozzle
- Approximate elevation of the Reactor Internal Pump (RIP) suction
- Design details of the core support plate (weight, thickness, diameter and distribution of holes in the core plate)
- Design details of the orificed and peripheral fuel supports (diameter of orifices and weight and height of the fuel supports)
- Length and inside diameter of control rod guide tubes
- Dimensions of the control rod (lengths of SS sheathed blades and absorber tubes, thickness of the blades, diameter of absorber tubes, and number of absorber tubes)
- Weights of SS and B4C in each control rod
- Weights of Zircaloy-4 and Zircaloy-2 in each fuel assembly
- Outside diameter of control rod housing
- Design details of the Top Guide (weight, thickness, diameter and distribution of holes in the core plate)
- Dimensions of the main steam line flow restricting nozzle
- Loss coefficient for steam/water flow through the main steam line flow restricting nozzle
- Discharge coefficient for steam/water flow through the main steam line flow restricting nozzle

(b) Fuel
- Weight of UO2 per assembly
- Pitch of the fuel assemblies (or spacing between the fuel assemblies)
- Length of the fuel channel (Zircaloy-4 canister)
- Fuel channel inside dimensions and wall thickness
- Bottom elevation of the fuel channel
- Length and material of the fuel assembly nose piece
- Length of the active fuel
- Elevation of the bottom of active fuel (BAF)
- Diametrical gap between fuel pellet and cladding
- Length of gas plenum
- Fuel rod cladding thickness
- Fuel rod outside diameter
- Pitch of the fuel rods
- Fuel pellet density
- Fuel pellet diameter

- Fuel pellet length
- Flow area of fully open Main Steam Isolation Valve (MSIV)
- Flow resistance of open MSIV
- Discharge coefficient of open MSIV

(c) Engineered Safety Features
- Flow area of fully open ADS valve
- Loss coefficient and discharge coefficient for the fully open ADS valve
- Setpoint value of the drywell pressure at which reactor trip occurs
- Setpoint value of the main steam line steam flow rate at which reactor trip occurs
- Setpoints for the closure of MSIV
- Elevations and radial positions of the HPCF, LPCF and RCIC systems suction strainer in SP
- Elevations and radial positions of the SRV line quenchers in the SP
- Elevation and radial position of the exit of RCIC turbine steam exhaust line the SP

(d) Feedwater Line Break (FWLB):
- A figure showing the containment pressure and temperature response (i.e., Figures 6.2-6 and 6.2-7 in [reference 2] STP COLA with the time axis varying from 0.0 to 30 min)
- A decay power curve in Fig. 6.3-11 of [reference 2] STP COLA is normalized with respect to which power; operating power or 102 % of the operating the operating power?

(e) Main Steam Line Break (MSLB):
- A figure showing the containment pressure and temperature response (i.e., Figures 6.2-12 and 6.2-13 in [reference 2] STP COLA with the time axis varying from 0.0 to 30 min)

RESPONSE:

The response to items (a) through (c) were previously transmitted to NRC in STPNOC Letter U7-STP-NRC-090038 on April 29, 2009. Table 1 and Figure 1 were provided as part of that response and are not included here. The response to items (d) and (e) are provided herein.

(d) For the first bullet item, the requested FWLB figures and curves are provided in Figures 2 through 9. The figures provide results for both the short term (0-50 sec) and the long term (0-13.9 hours) analyses.
For the second bullet item, the decay power curve in Fig. 6.3-11 is not used in the containment analysis but rather is used for the ECCS analysis. Consequently, the normalized power level for that Figure would not have any bearing on the containment analysis results. The decay power curve used for the containment analysis is based on 102% power.

(e) The requested MSLB figures are provided in Figures 10 through 17. These figures provide results for both the short term and the long term analyses.

There are no revisions to the COLA as a result of this response.

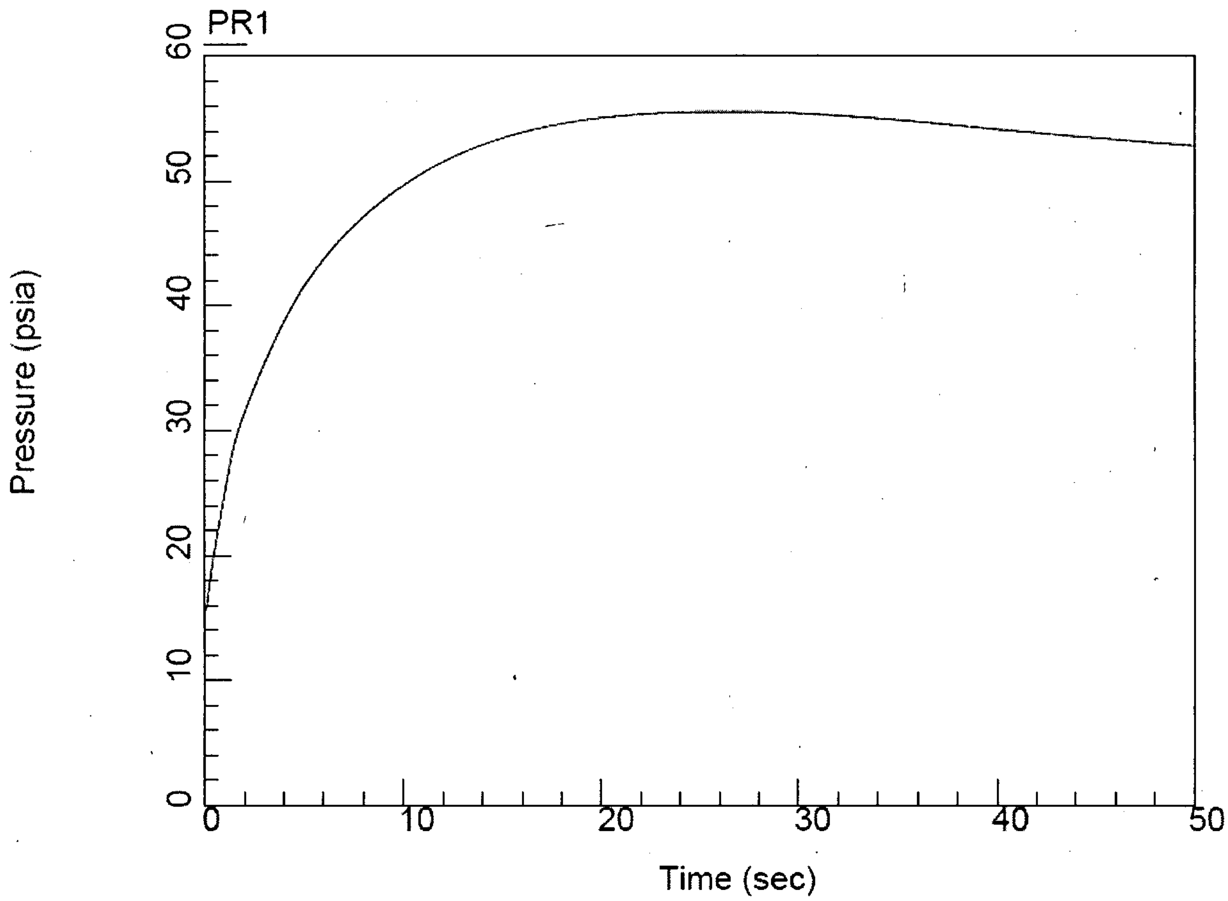


Figure 2 Time-Dependent FWLB Drywell Pressure (Short Term)

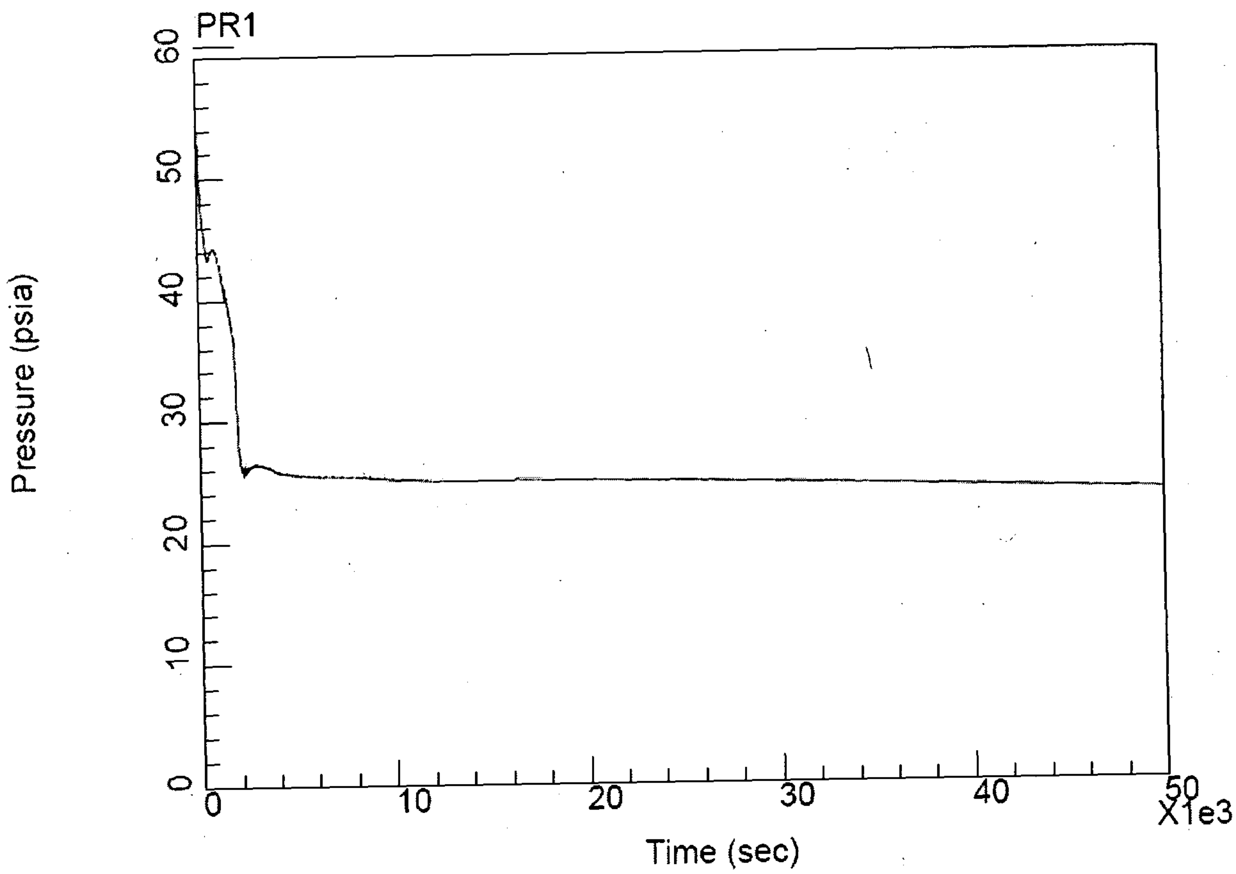


Figure 3 Time-Dependent FWLB Drywell Pressure (Long Term)

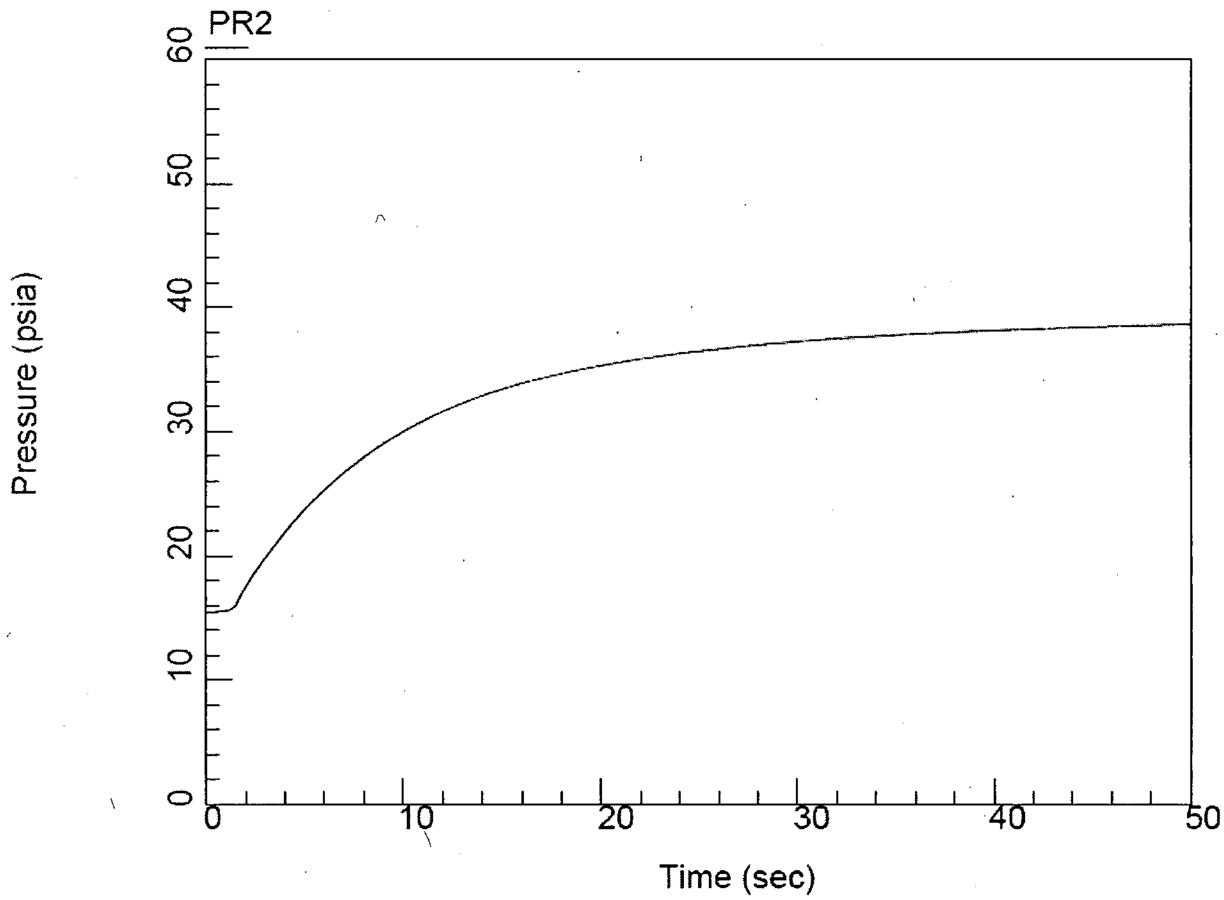


Figure 4 Time Dependent FWLB Wetwell Pressure (Short Term)

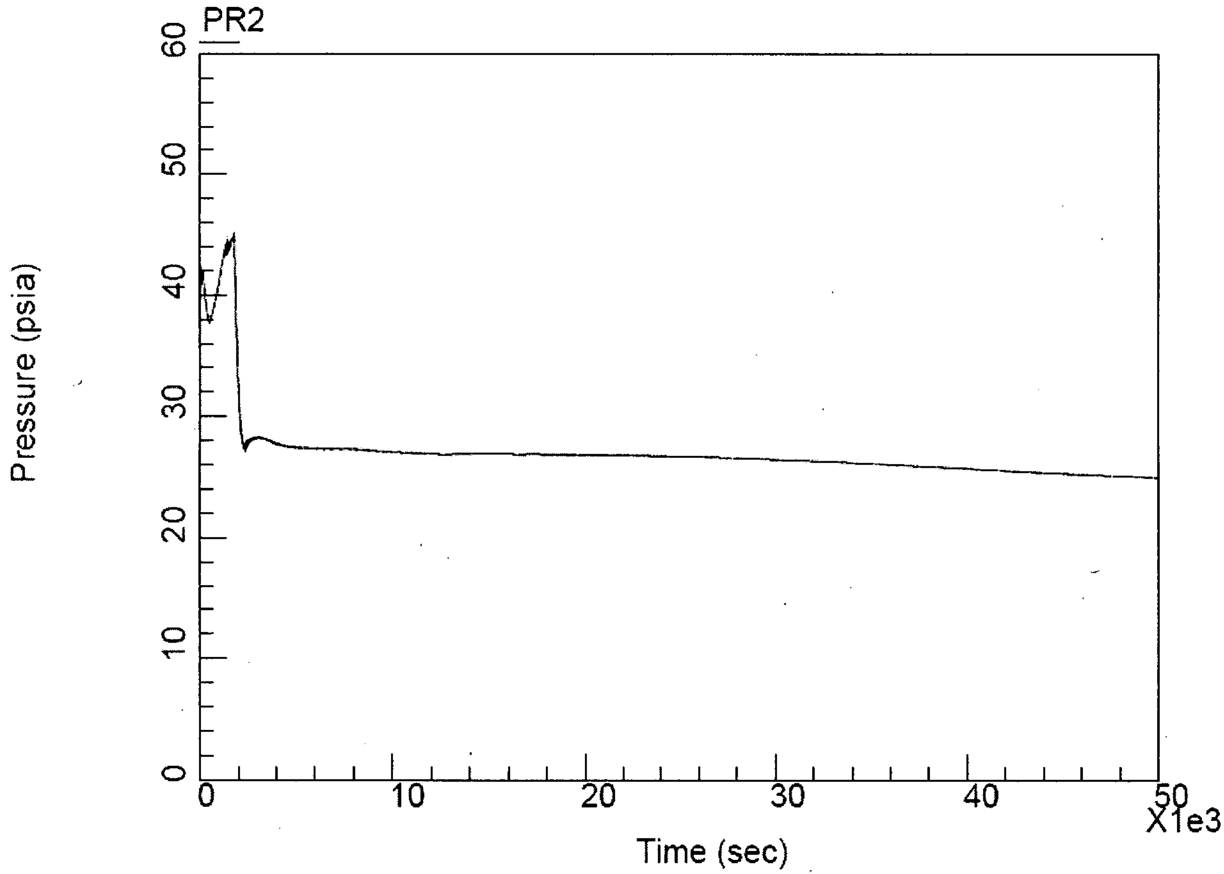


Figure 5 Time Dependent FWLB Wetwell Pressure (Long Term)

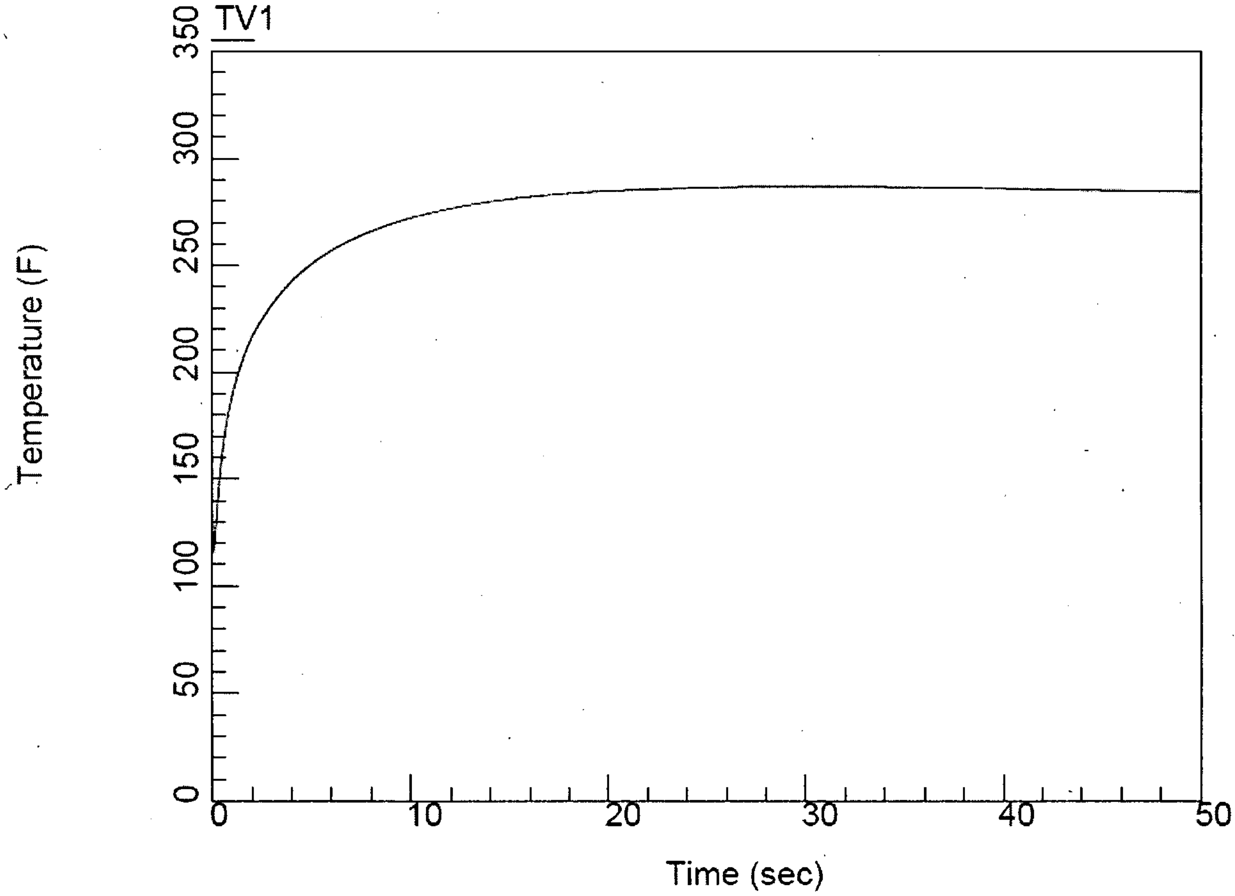


Figure 6 Time-Dependent FWLB Drywell Temperature (Short Term)

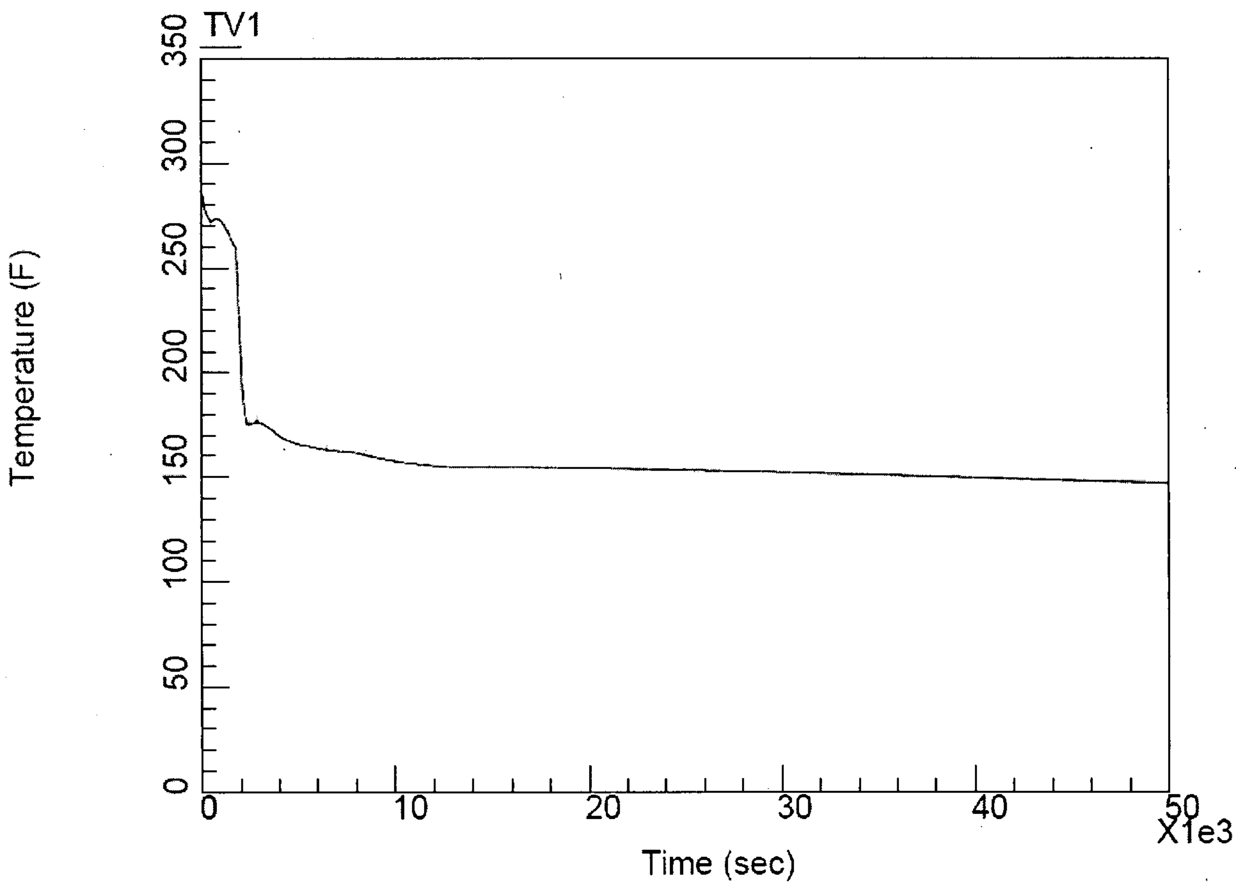


Figure 7 Time-Dependent FWLB Drywell Temperature (Long Term)

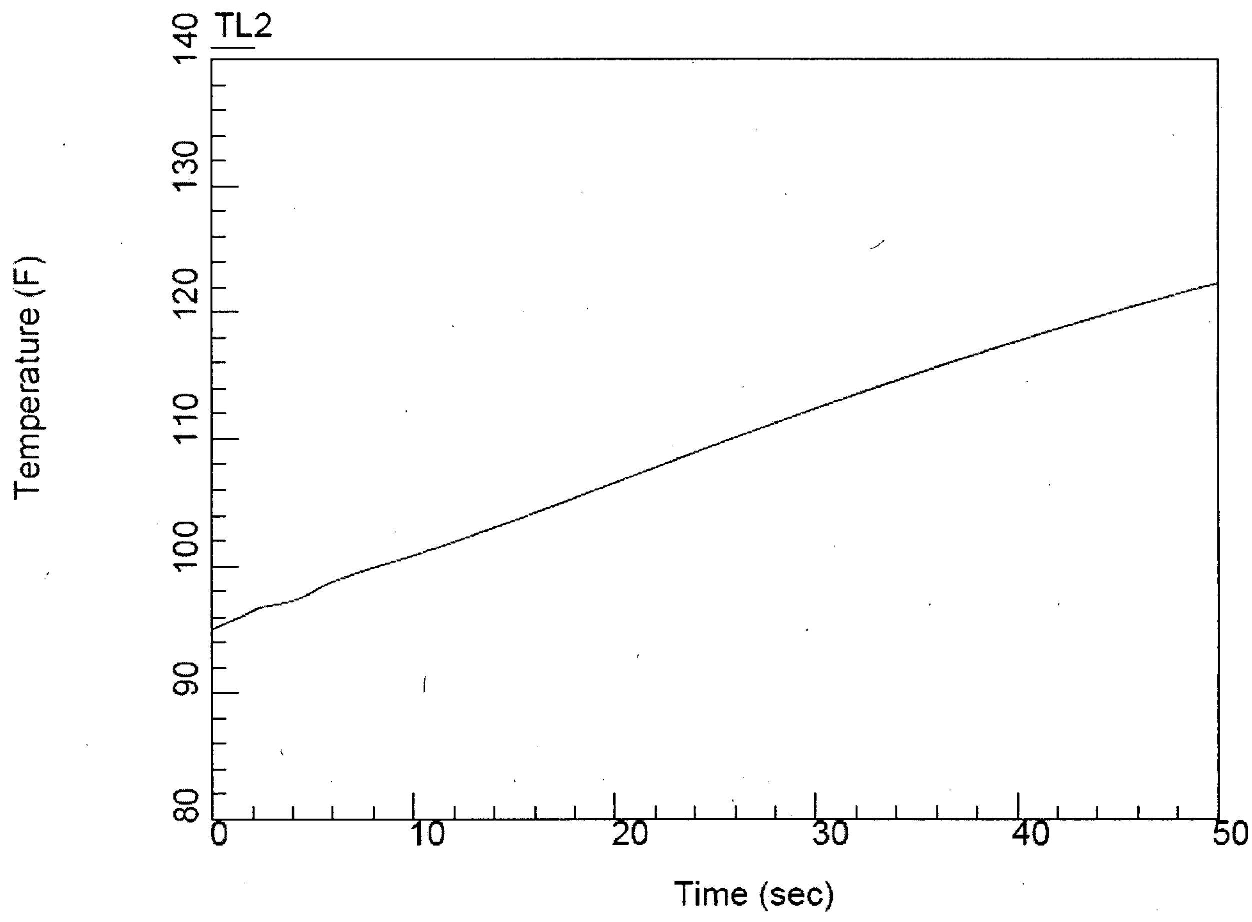


Figure 8 Time-Dependent FWLB Suppression Pool Temperature (Short Term)

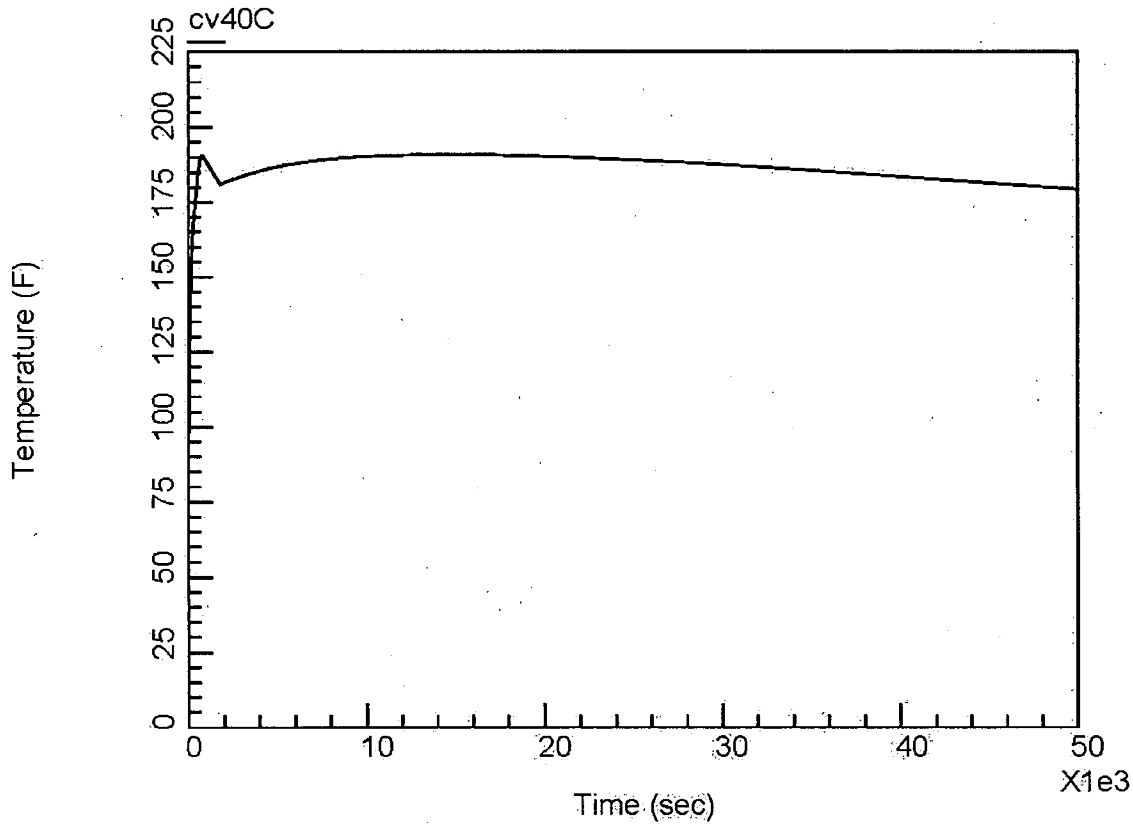


Figure 9 Time-Dependent FWLB Suppression Pool Temperature (Long Term)

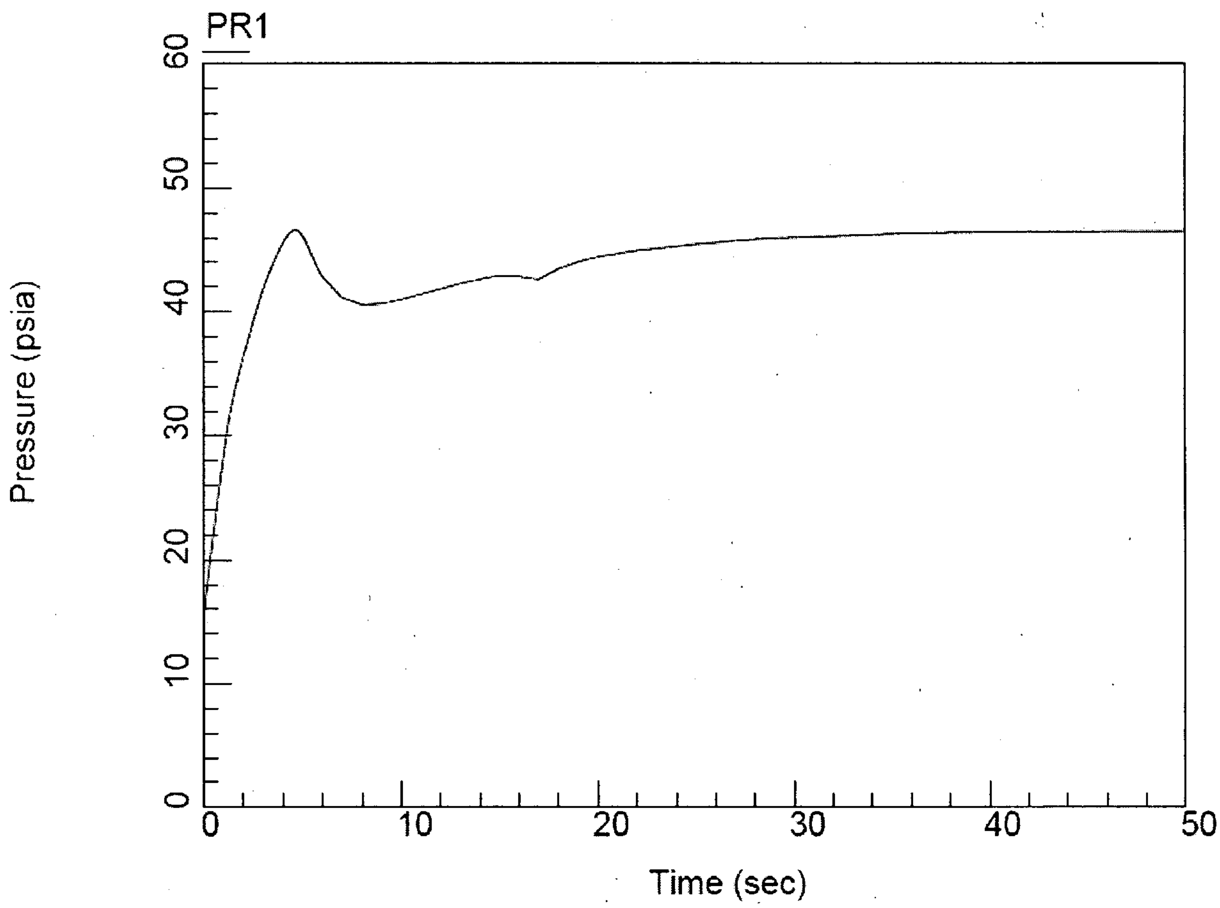


Figure 10 Time–Dependent MSLB Drywell Pressure (Short Term)

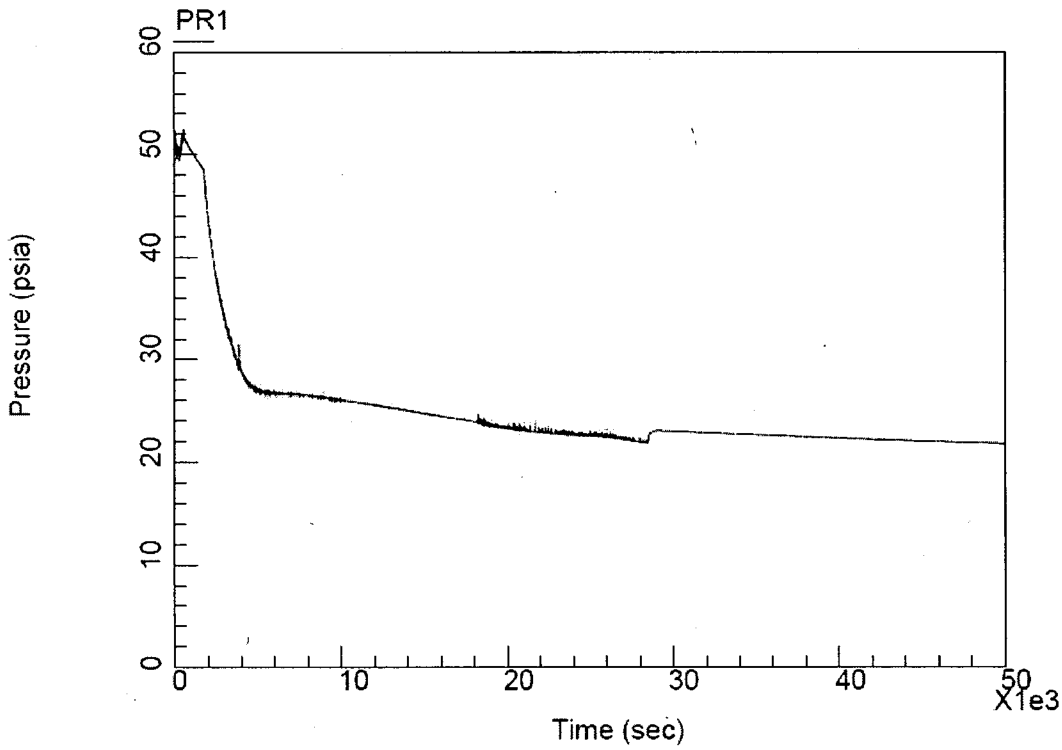


Figure 11 Time–Dependent MSLB Drywell Pressure (Long Term)

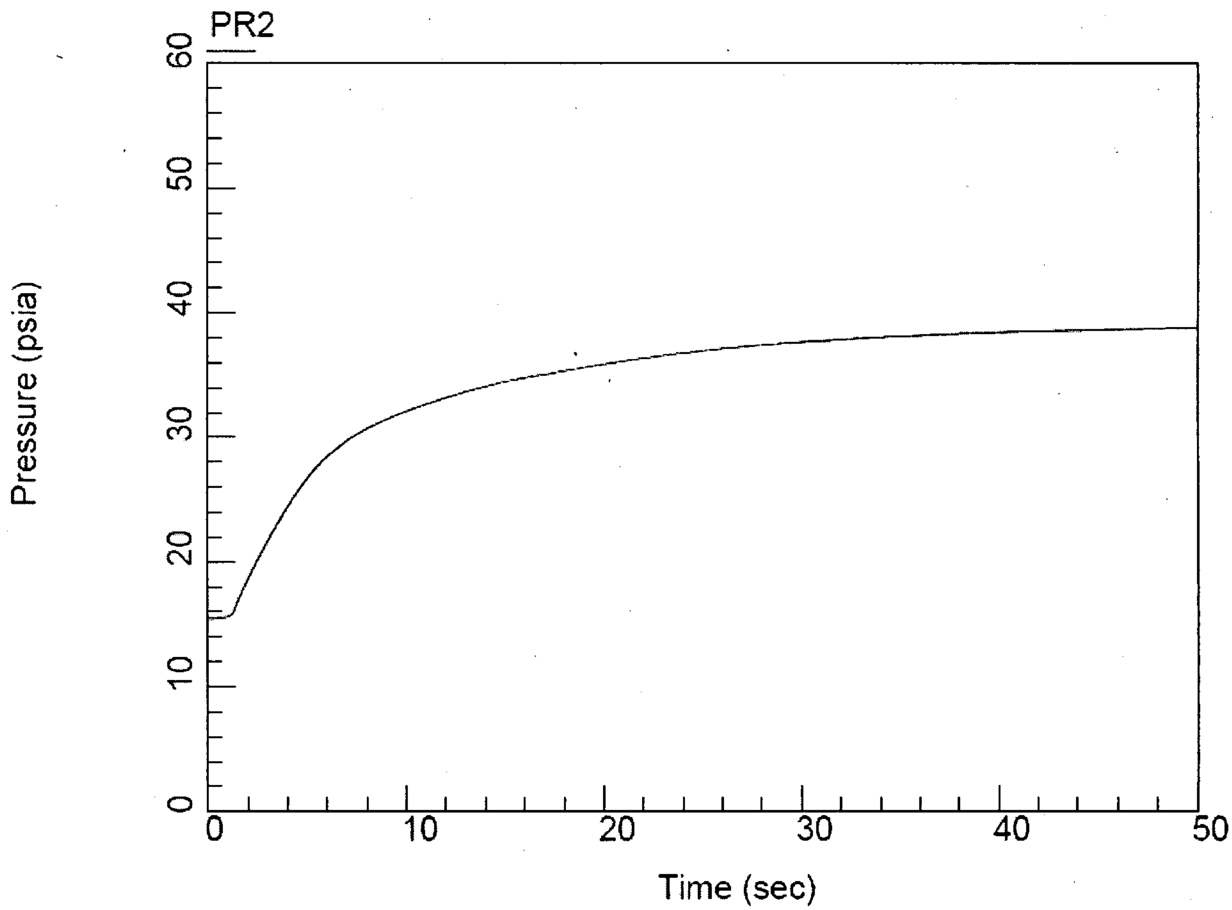


Figure 12 Time-Dependent MSLB Wetwell Pressure (Short Term)

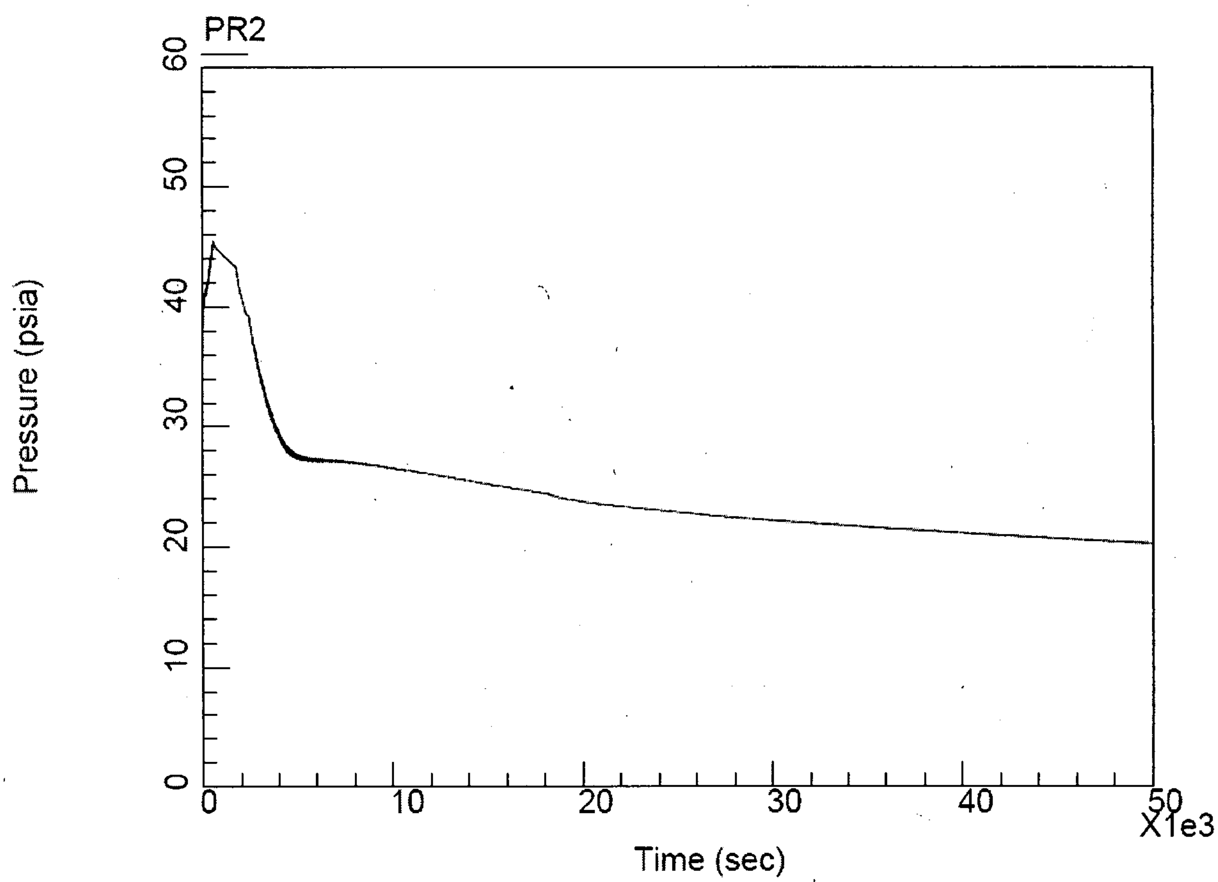


Figure 13 Time-Dependent MSLB Wetwell Pressure (Long Term)

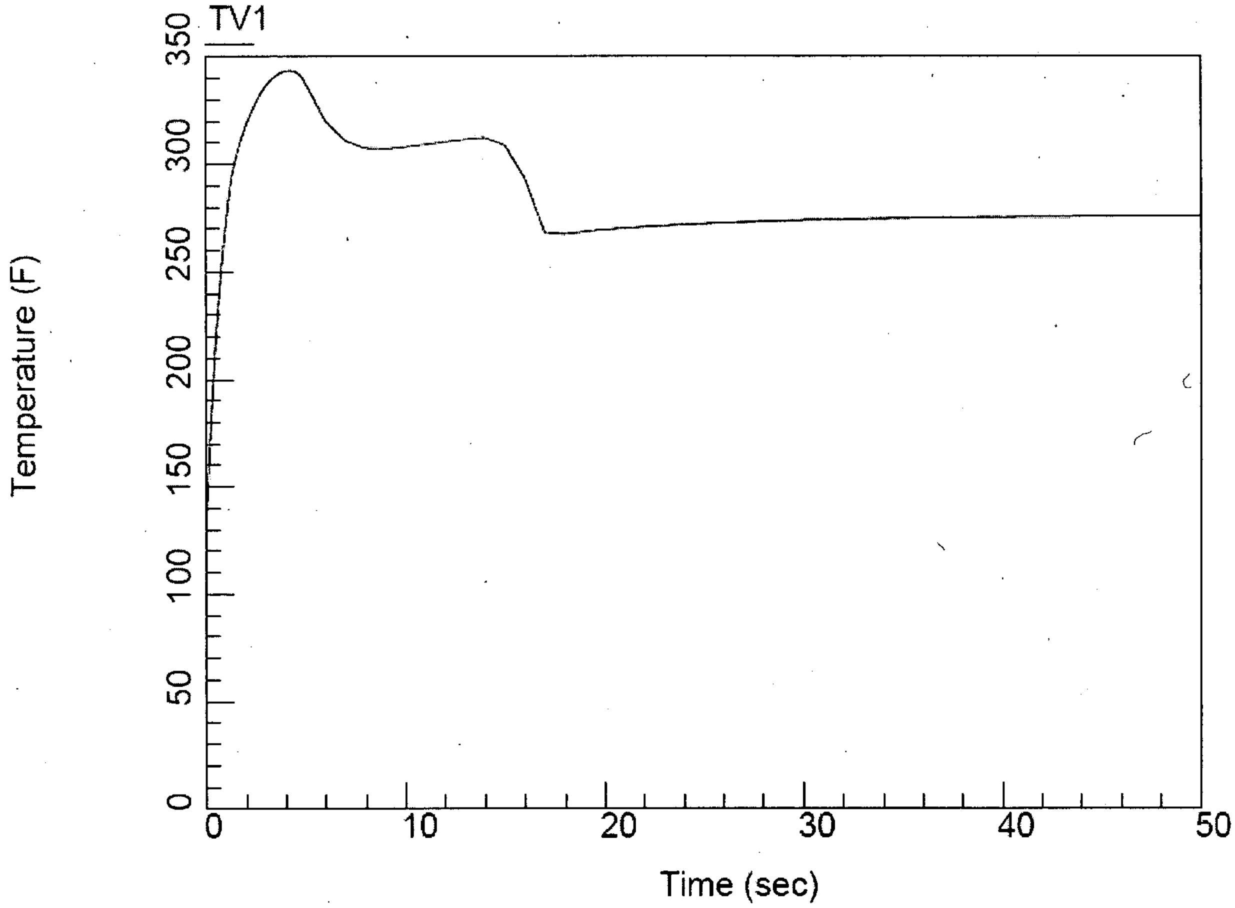


Figure 14 Time-Dependent MSLB Drywell Temperature (Short Term)

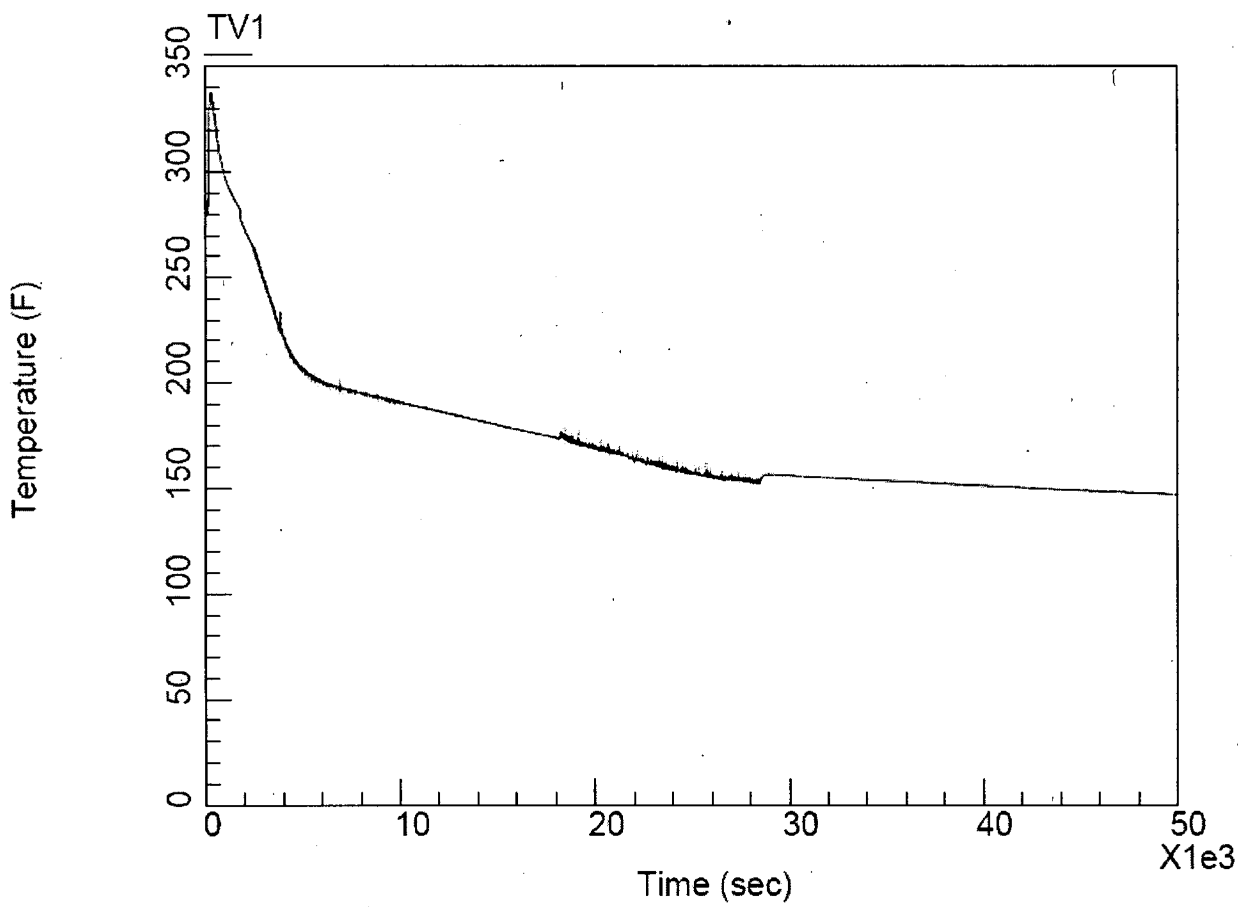


Figure 15 Time-Dependent MSLB Drywell Temperature (Long Term)

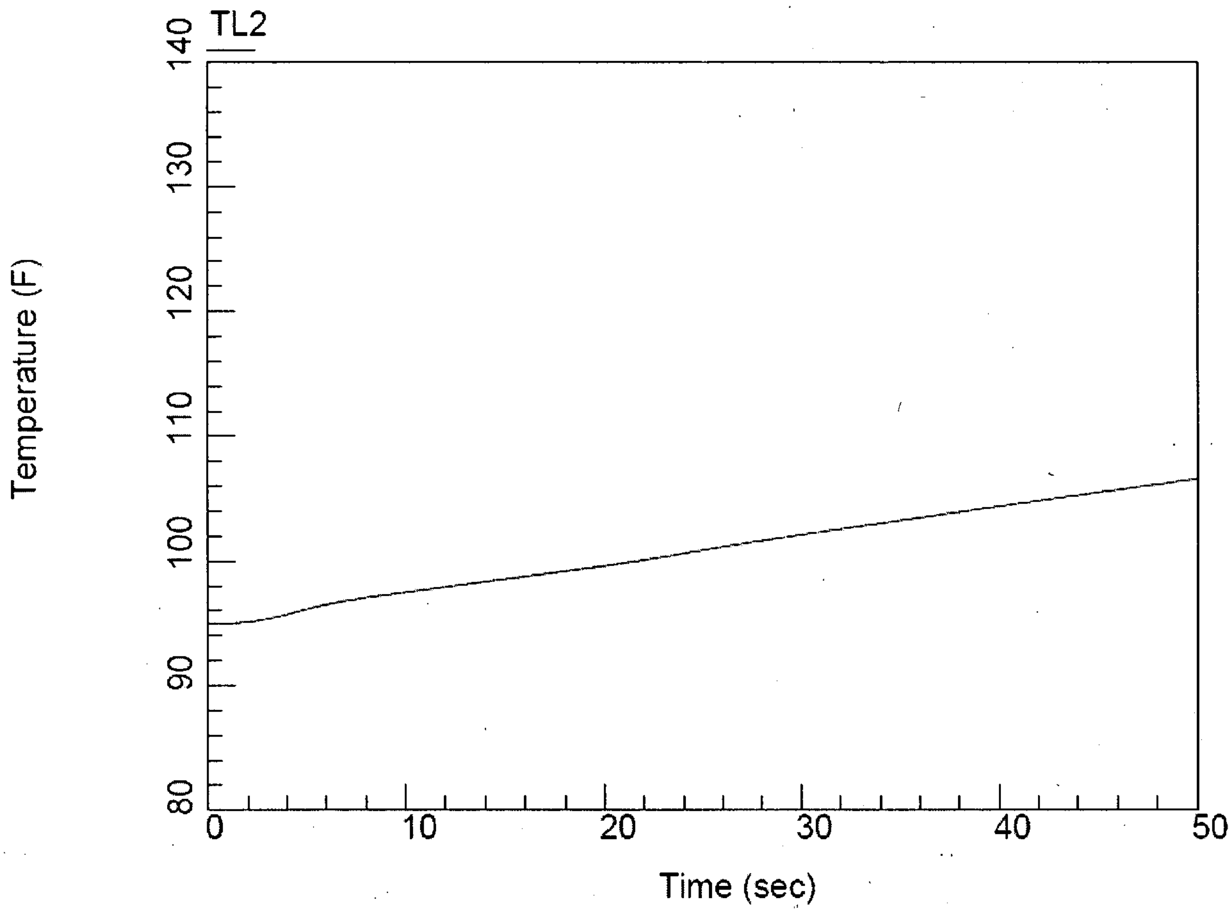


Figure 16 Time-Dependent MSLB Mixed Mean Suppression Pool Temperature (Short Term)

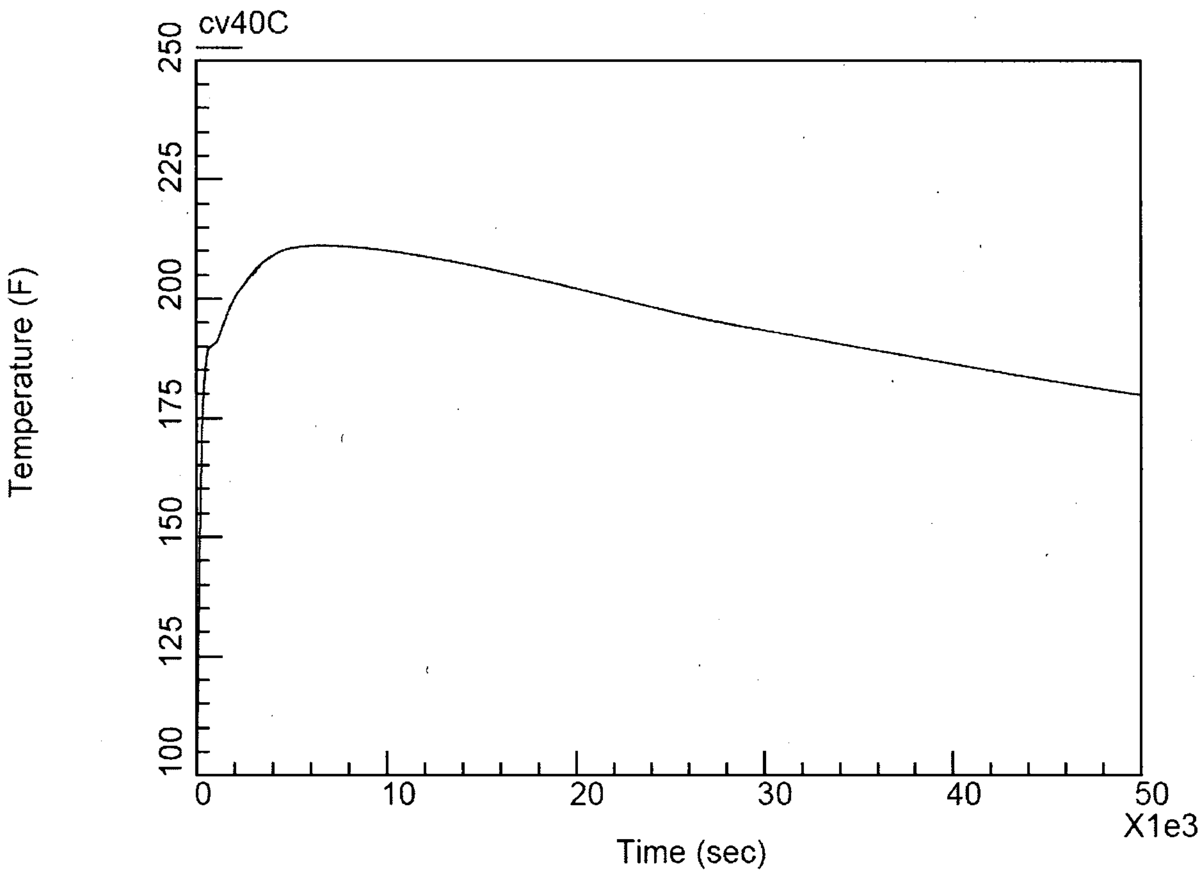


Figure 17 Time-Dependent MSLB Mixed Mean Suppression Pool Temperature (Long Term)