Entergy

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Ken Peters
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2004-0086

October 8, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request NPF-38-249,
Extended Power Uprate
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter dated November 13, 2003, "License Amendment Request NPF-38-249 Extended Power Uprate"
2. Entergy Letter dated May 13, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
3. Entergy Letter dated August 10, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
4. Entergy Letter dated July 14, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License and Technical Specifications to increase the unit's rated thermal power level from 3441 megawatts thermal (MWt) to 3716 MWt.

Entergy, Westinghouse, and members of your staff held a series of calls to discuss civil/mechanical aspects of the Extended Power Uprate (EPU) amendment request previously provided in Reference 1, 2, and 3. As a result of these calls, the responses to five questions were determined to need formal response. Entergy's responses to these questions are contained in Attachment 1.

The no significant hazards consideration included in Reference 4 is not affected by any information contained in this supplemental letter. This submittal includes one new commitment as summarized in Attachment 2.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 8, 2004.

Sincerely,

KJP/DBM/cbh

Attachments:
1. Response to Request for Additional Information
2. List of Regulatory Commitments

cc: Dr. Bruce S. Mallett
U. S. Nuclear Regulatory Commission
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U.S. Nuclear Regulatory Commission
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Baton Rouge, LA 70821-4312

American Nuclear Insurers
Attn: Library
Town Center Suite 300S
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West Hartford, CT 06107-2445

Attachment 1

To

W3F1-2004-0086

Response to Request for Additional Information

Response to Request for Additional Information

Question 1:

In Section 2.2.2.1.1, you indicated that power uprate resulted in minor changes to as-calculated normal operation transients, and a detailed evaluation of the changes in stresses and fatigue levels in the Reactor Coolant System (RCS) due to these transients was performed to demonstrate that the originally specified design transients for Waterford 3 remain applicable under Extended Power Uprate (EPU) conditions. This was accomplished by demonstrating that transients affected by EPU have no significant effect on the stresses or cumulative usage factors (CUFs) of the limiting RCS components. Provide a comparison of transients at the power uprate conditions to the original design-basis transients with respect to the number of occurrences and changes in pressure and temperature in the stress and cumulative fatigue usage calculations.

Response 1:

The originally specified design transients were evaluated for the potential effect EPU would have on them. The same list of events and number of event cycles were maintained for the EPU design transient evaluation. The Reactor Trip, Loss of Load, Loss of Flow and Loss of Secondary Pressure events were rerun using EPU conditions. The plots for the rerun cases are attached. The rerun cases and results of the evaluation were documented and evaluated for any effects on stress and fatigue levels.

Figure B-1 RCS Loop Temperature Deviation for Reactor Trip Transient

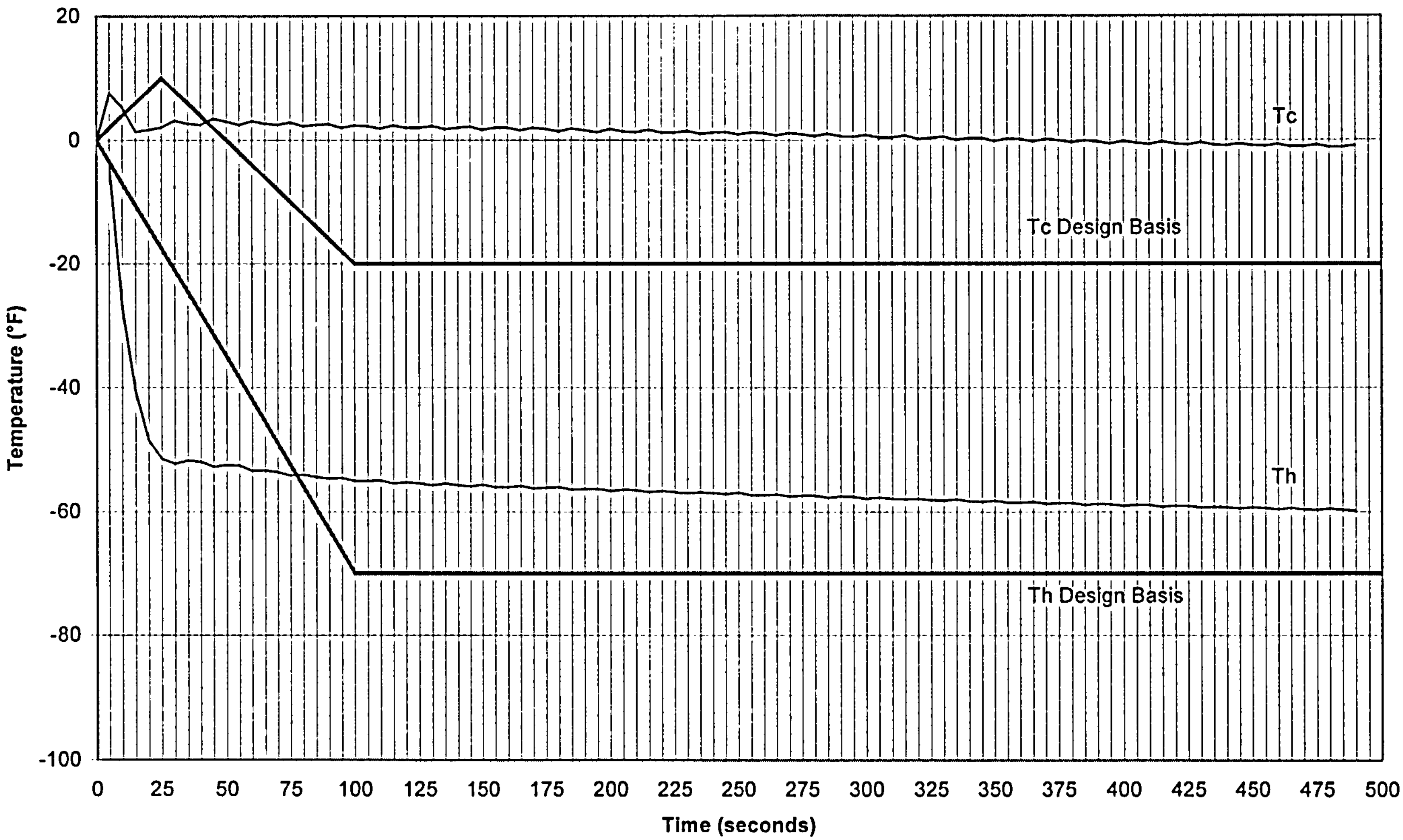


Figure B-2 Pressurizer Pressure Deviation for Reactor Trip Transient

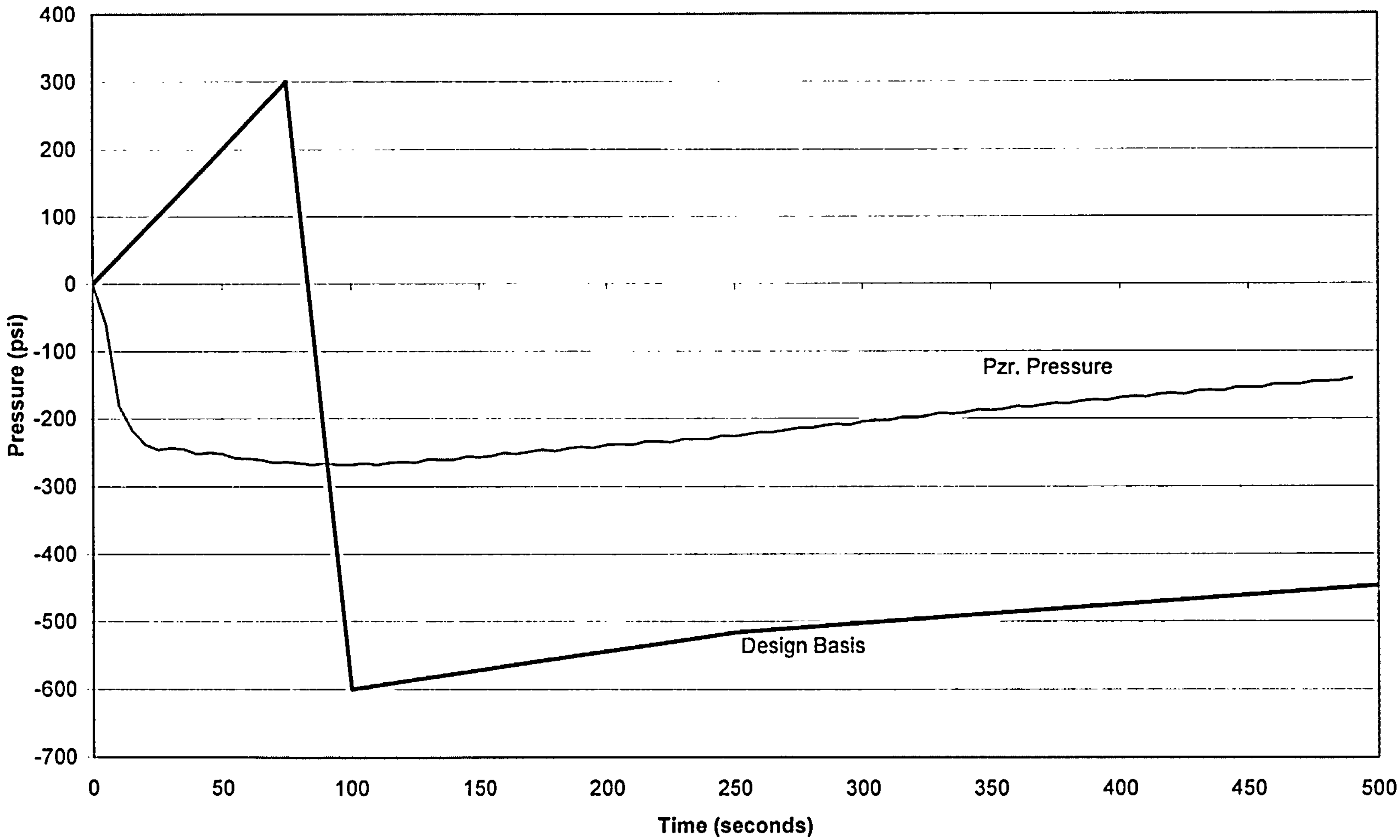


Figure B-3 RCS Loop Temperature Deviation for Loss of Load Transient

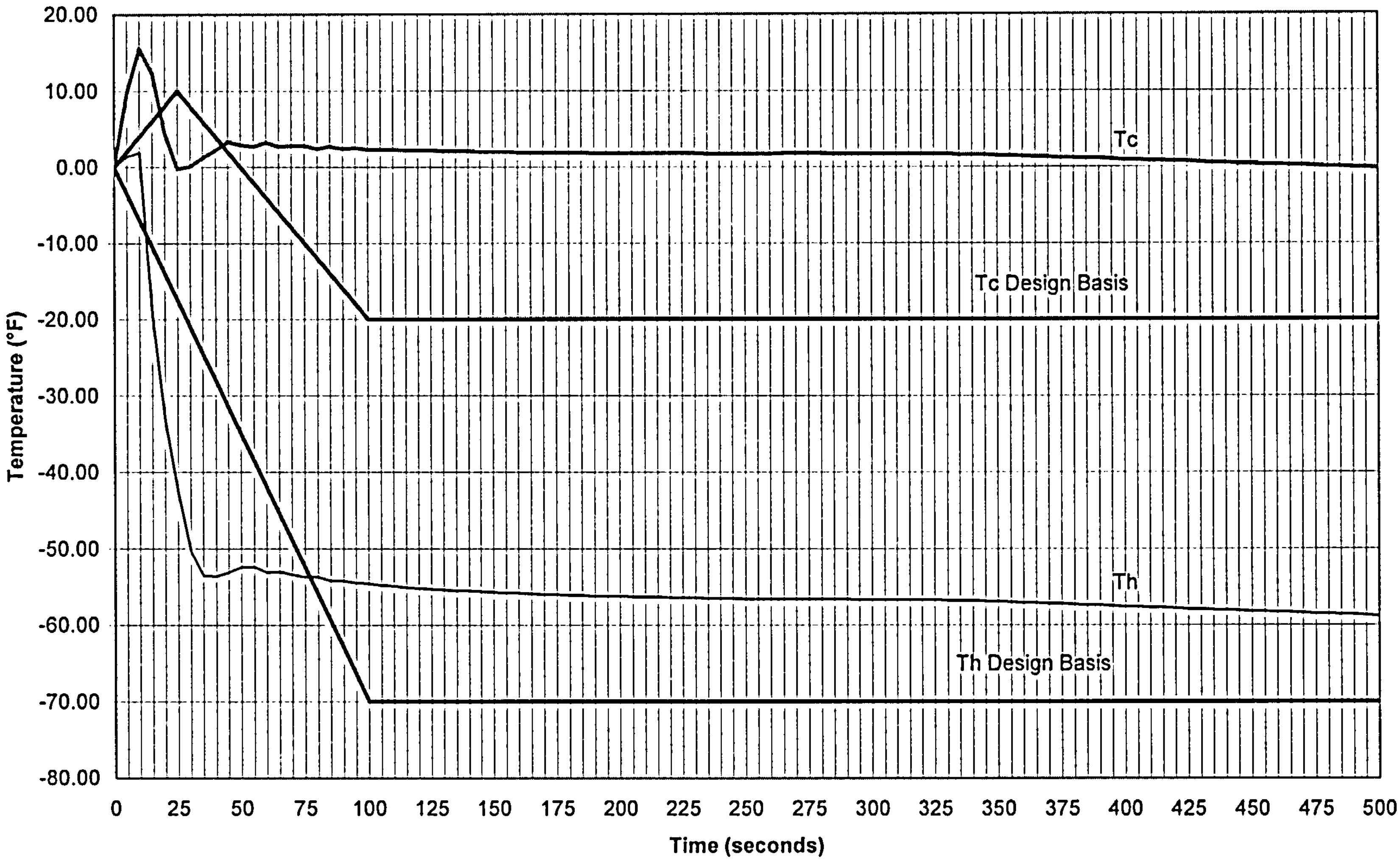


Figure B-4 Pressurizer Pressure Deviation for Loss of Load Transient

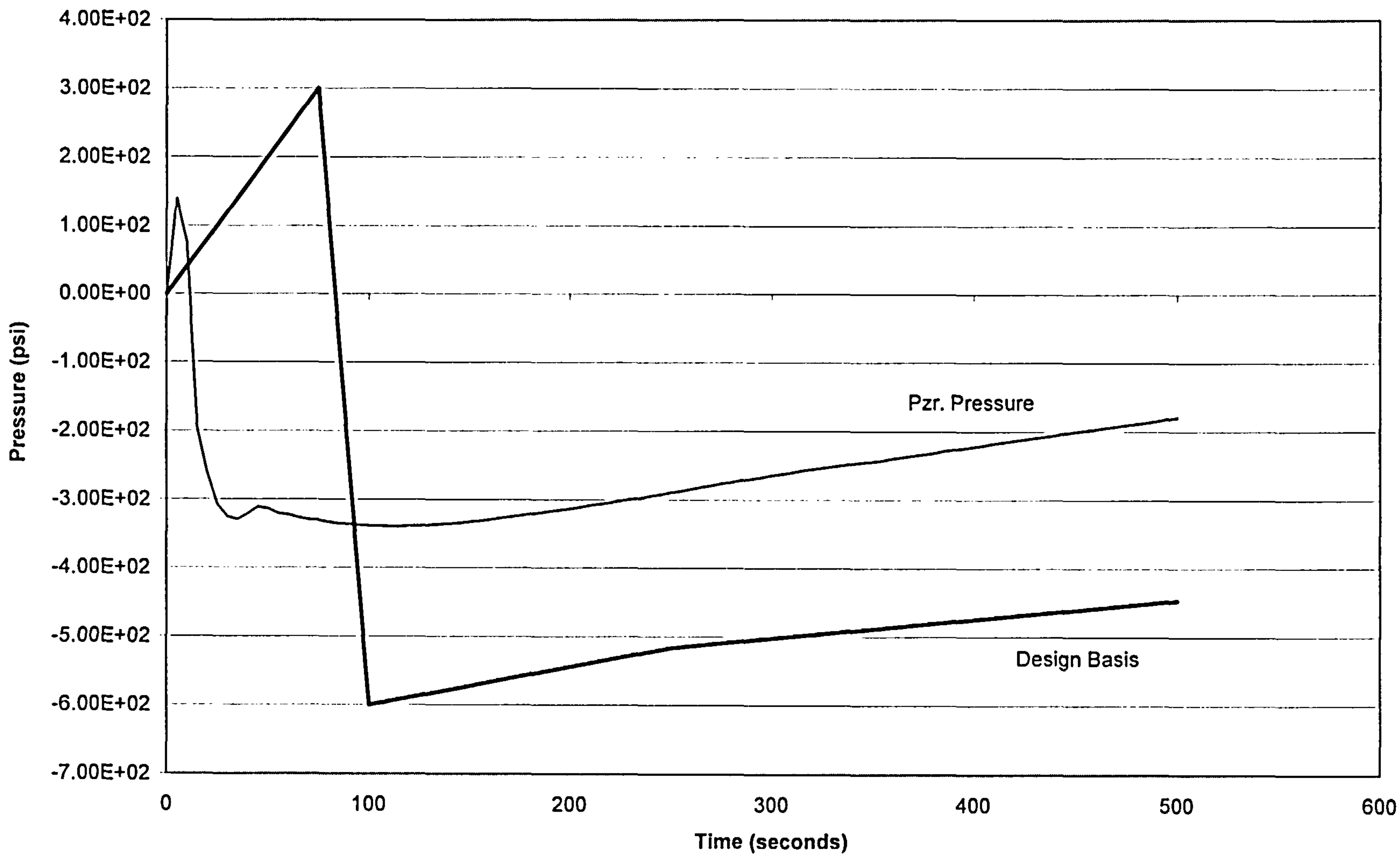


Figure B-5 RCS Loop Temperature Deviation for Loss Of Flow Transient

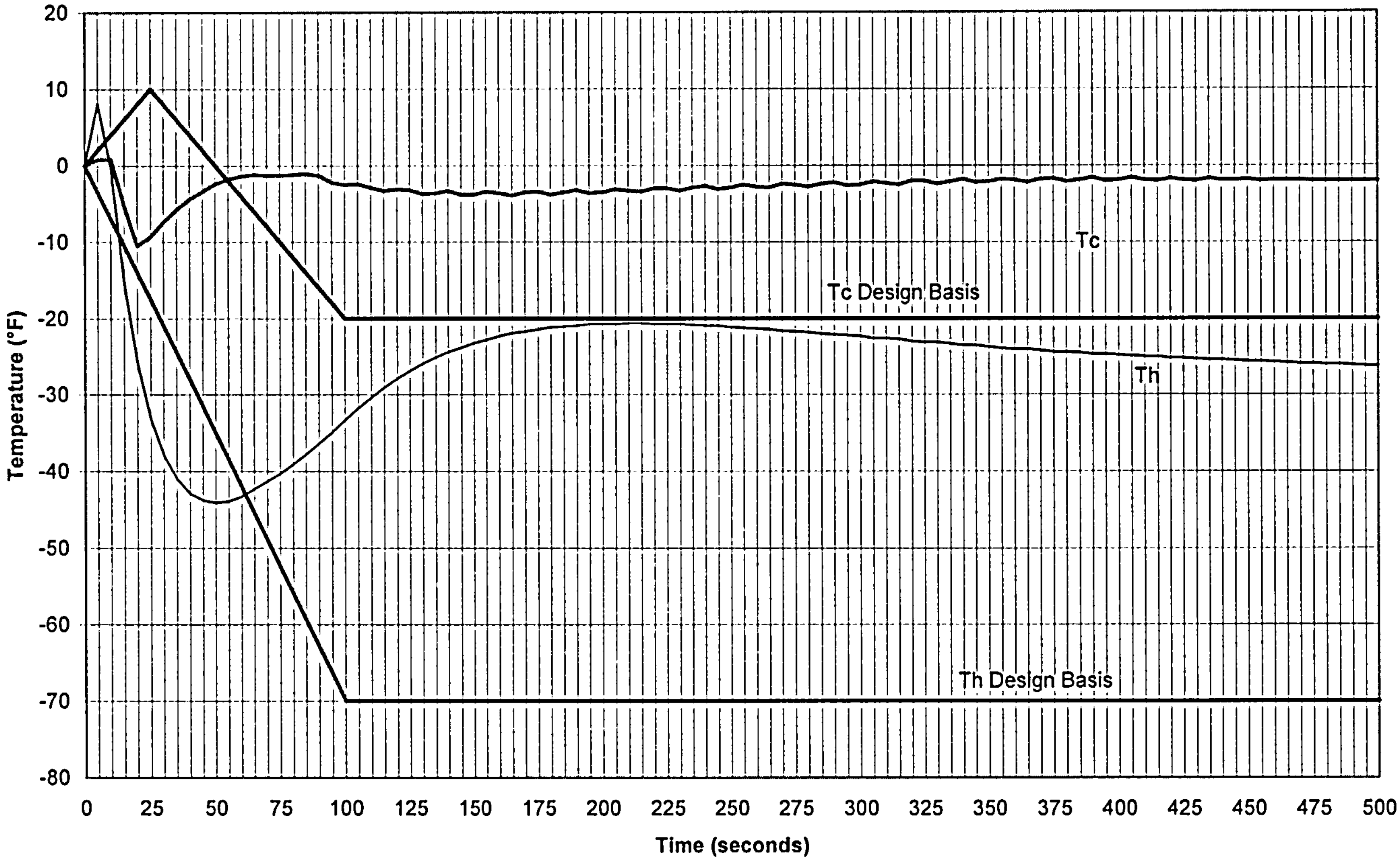


Figure B-6 Pressurizer Pressure Deviation for Loss Of Flow Transient

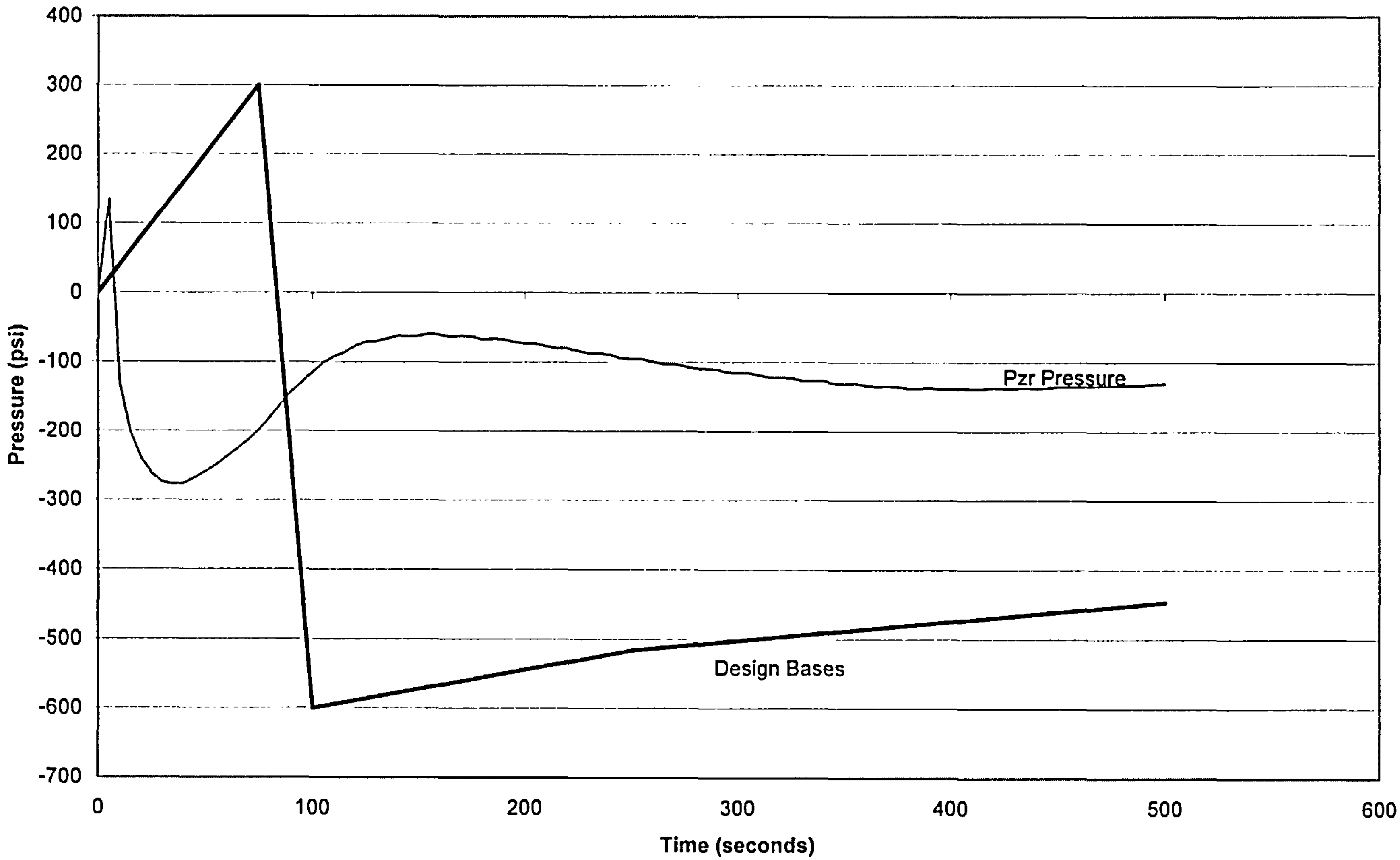


Figure B-7 RCS Loop Temperature Deviation for Full Load MSLB Transient

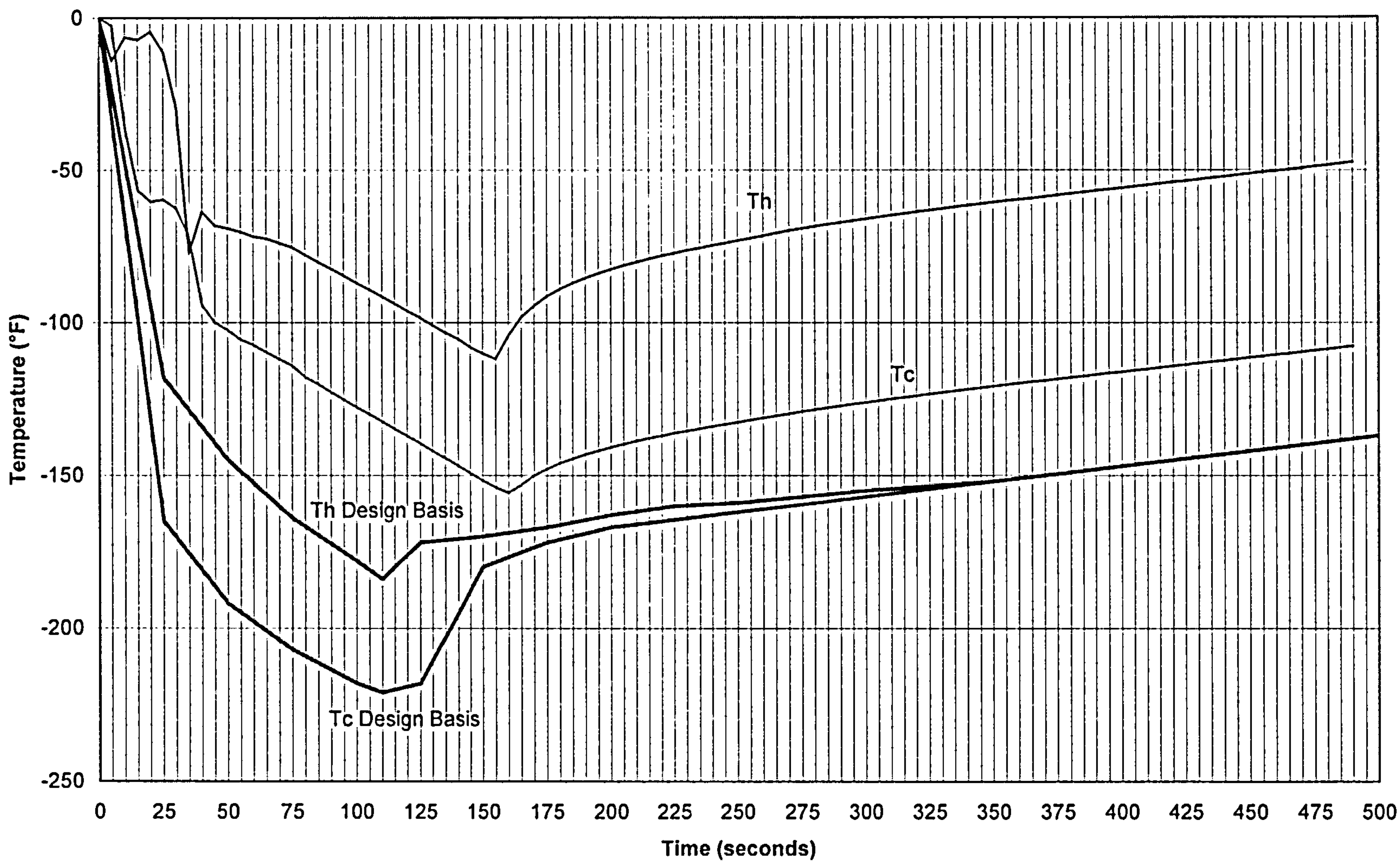


Figure B-8 Pressurizer Pressure Deviation for Full Load MSLB Transient

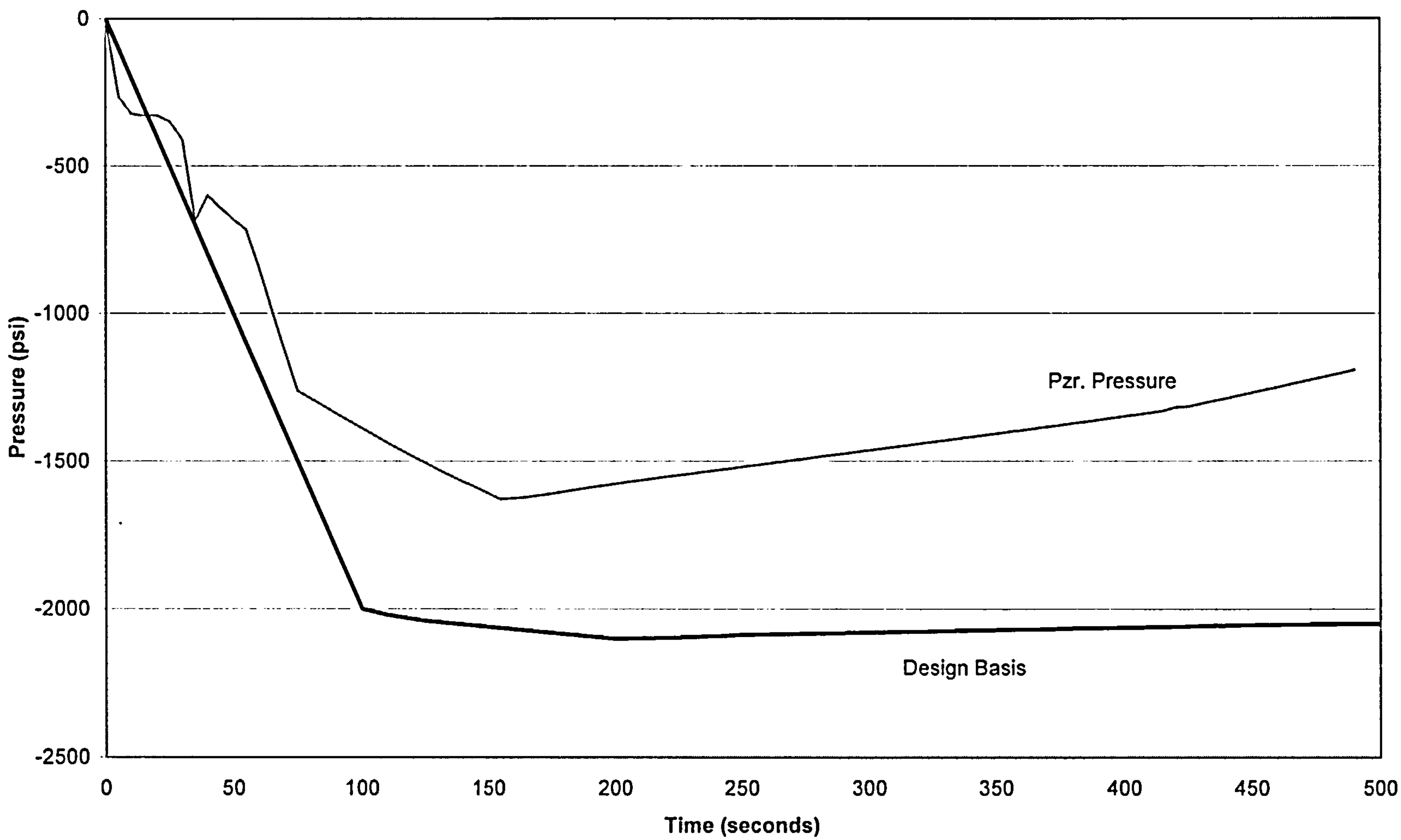


Figure B-9 RCS Loop Temperature Deviation for No-Load MSLB Transient

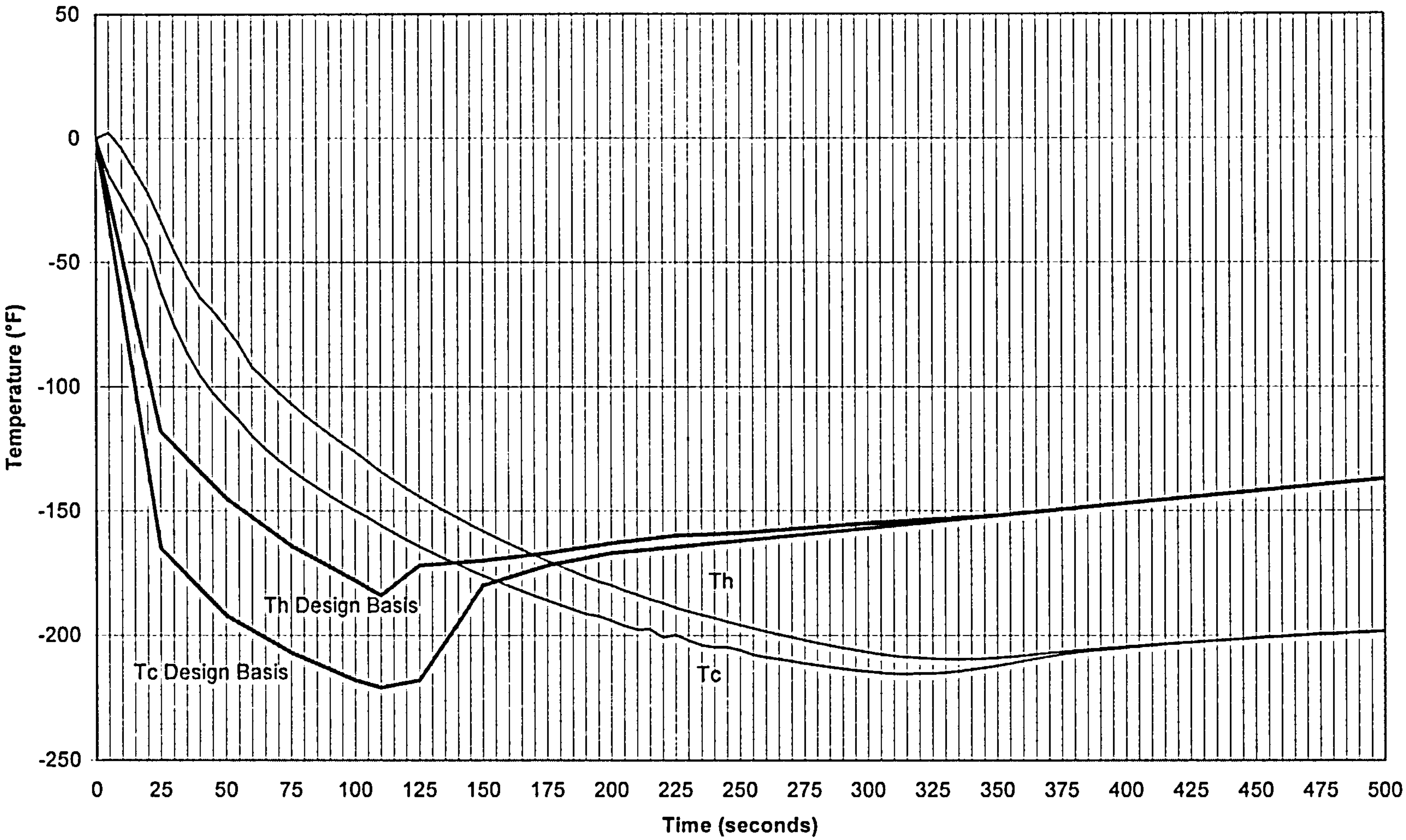


Figure B-10 Pressurizer Pressure Deviation for No-Load MSLB Transient

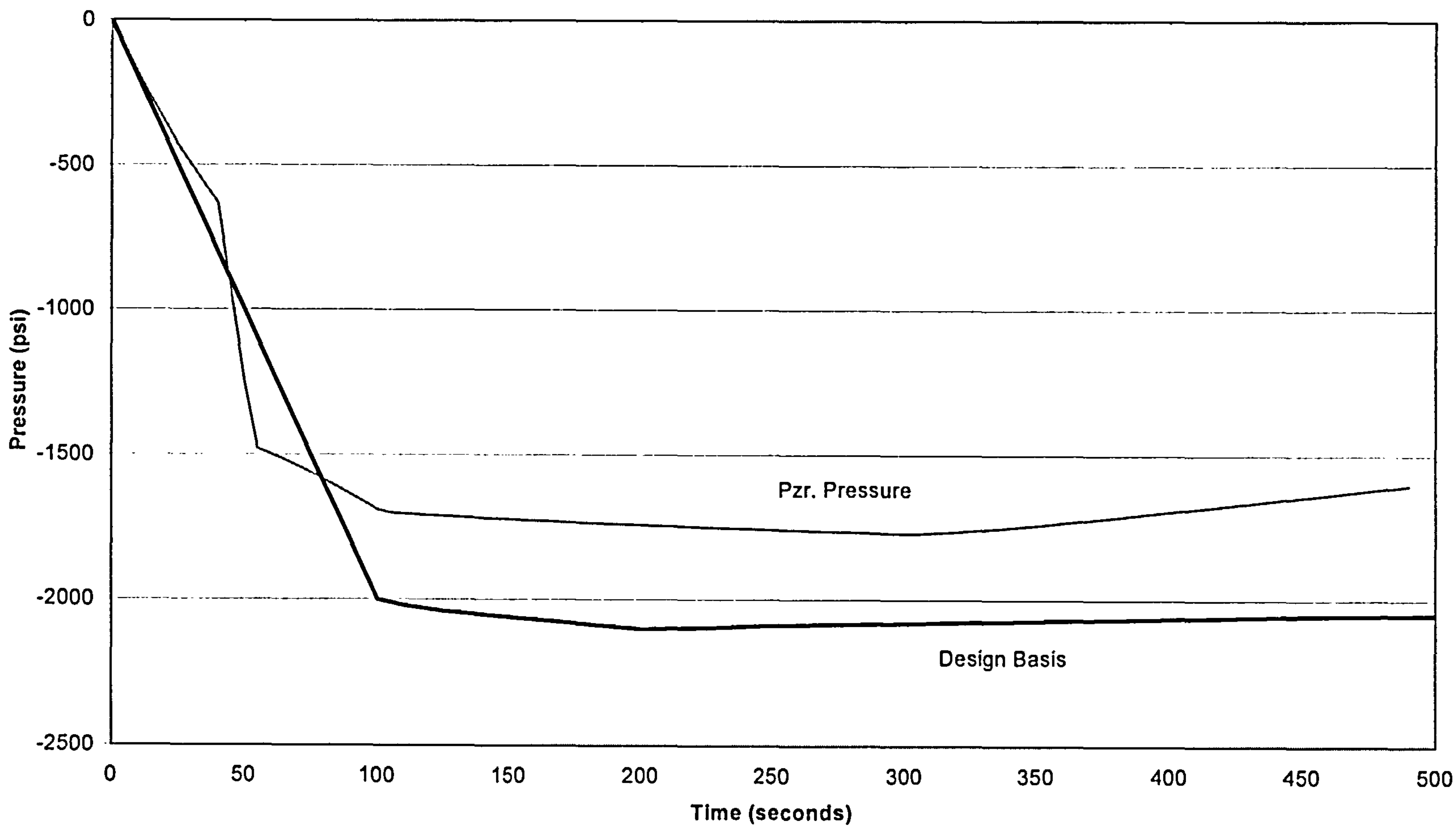


Figure B-11 Pressurizer Temperature Deviation for Loss of Load

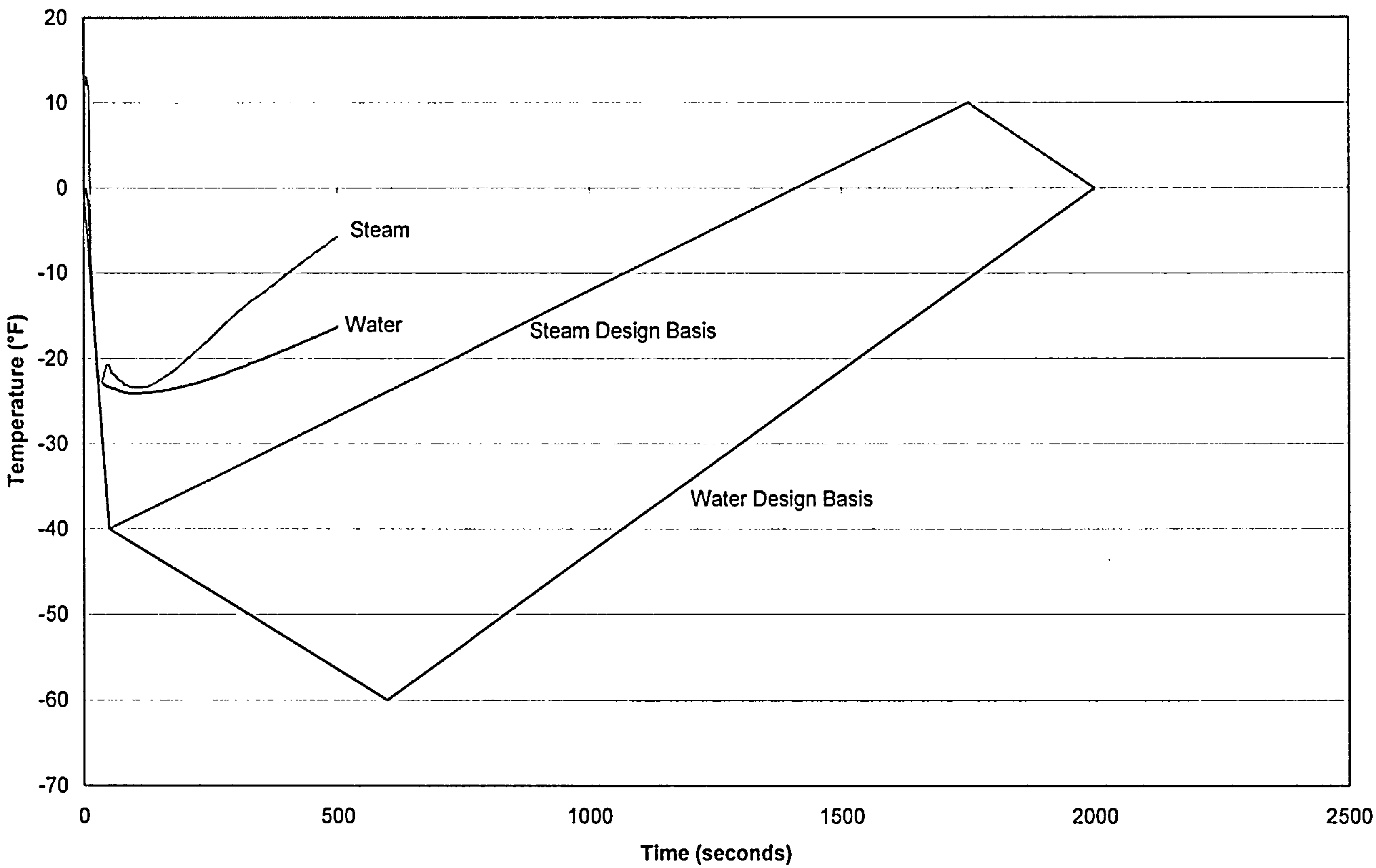
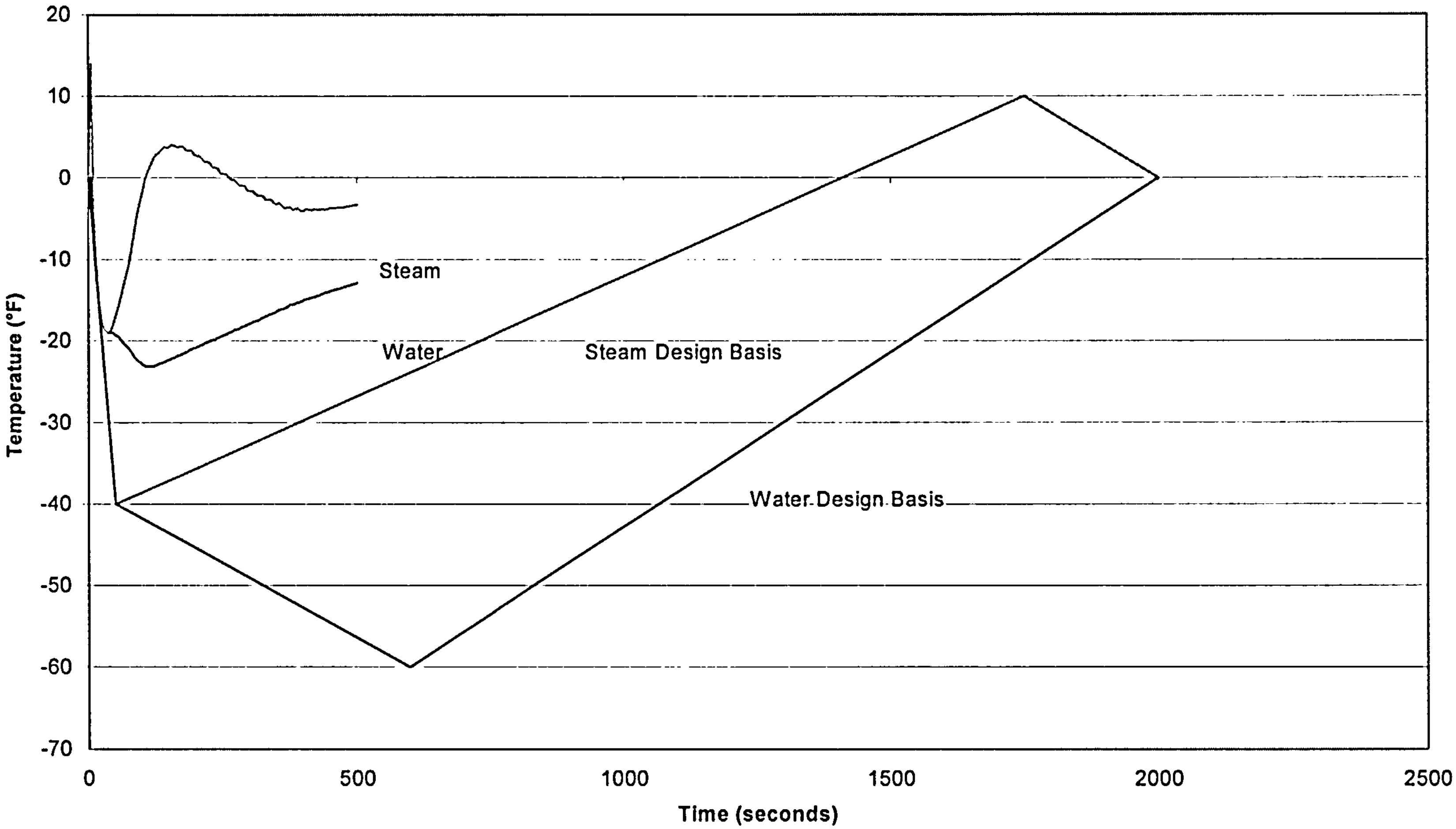
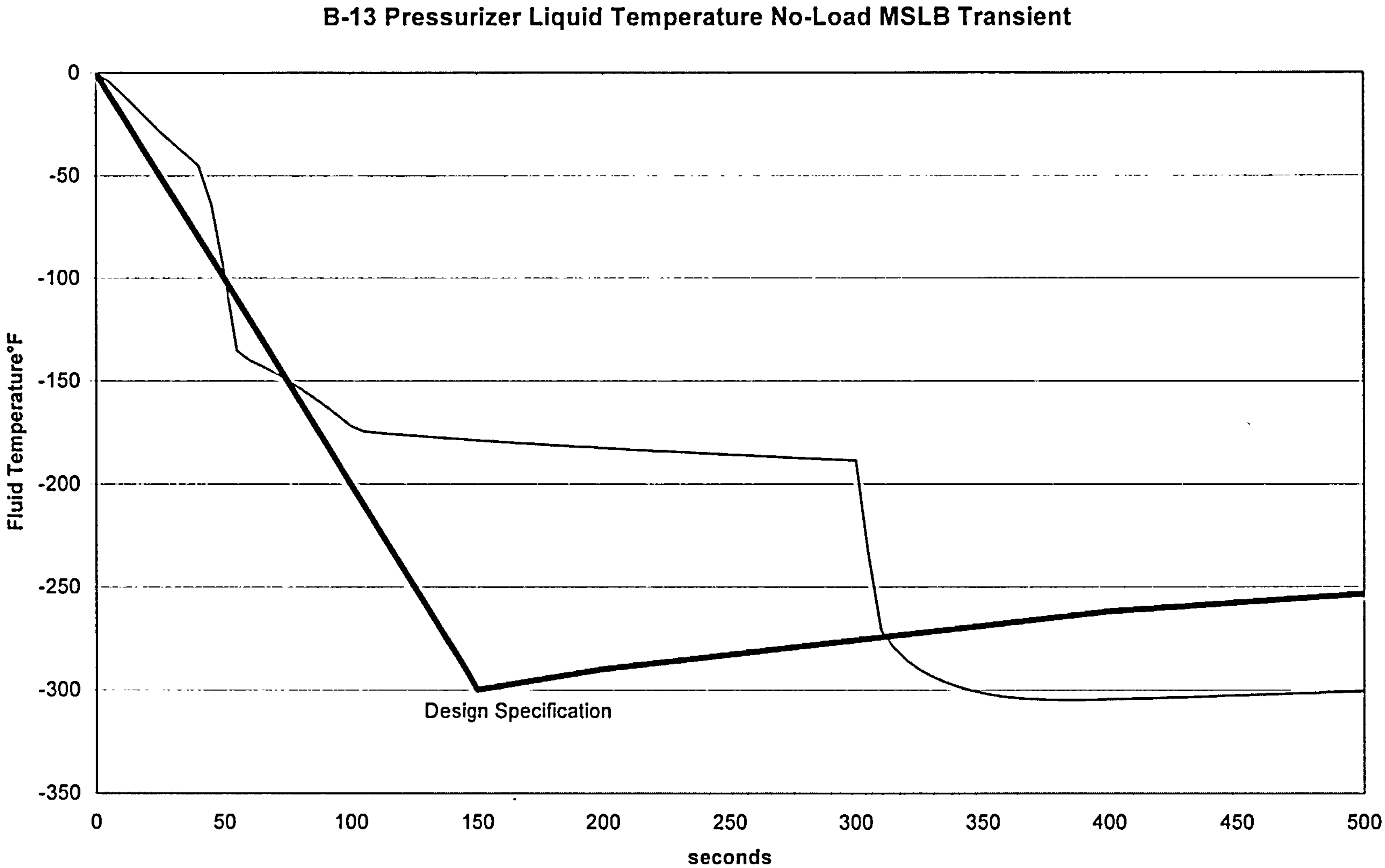
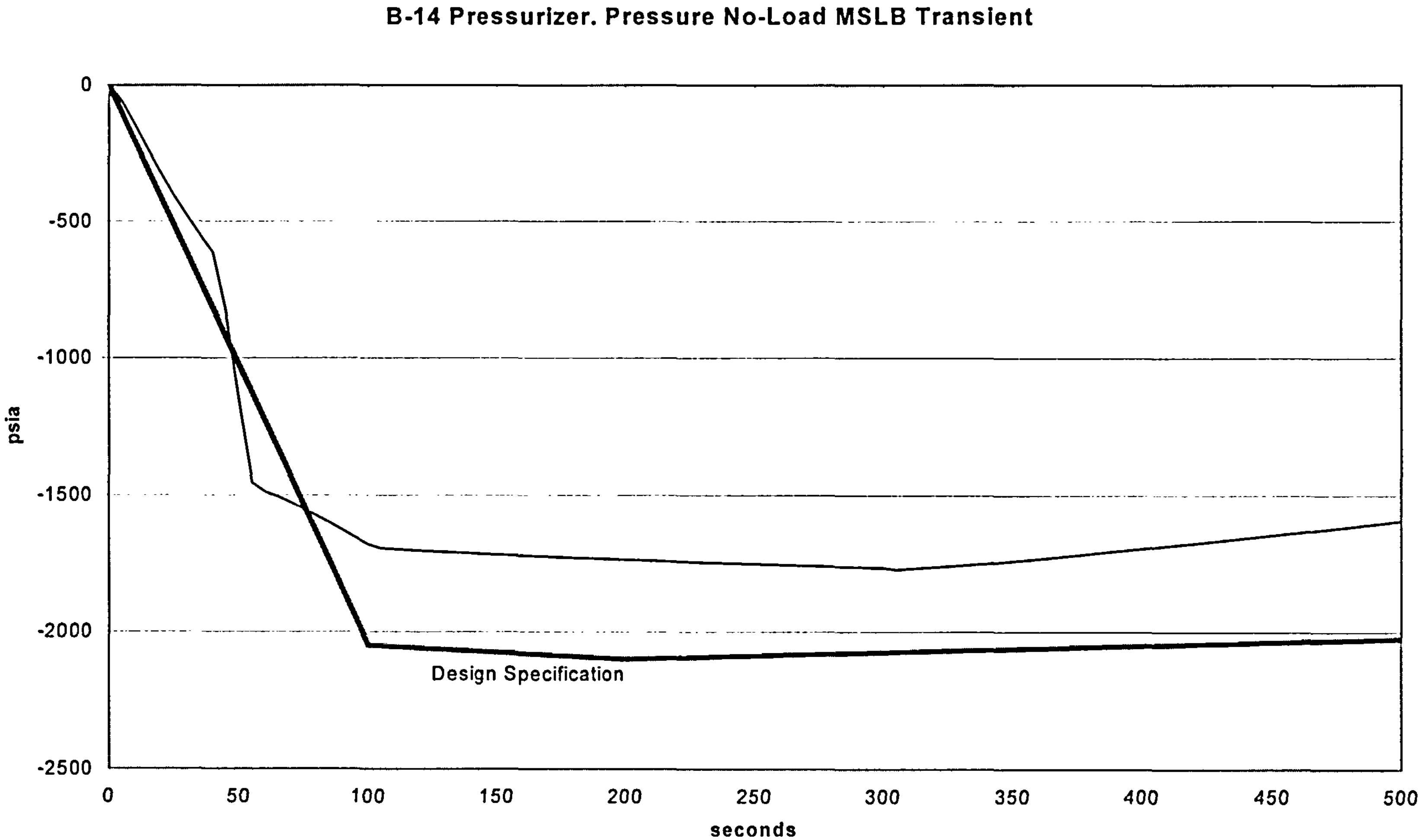


Figure B-12 Pressurizer Temperature Deviation for Loss of Flow Transient




B-13 Pressurizer Liquid Temperature No-Load MSLB Transient
Fluid Temperature°F
0
-50
-100
-150
-200
-250
-300
-350
0
50
100
150
200
250
300
350
400
450
500
seconds
Design Specification


B-14 Pressurizer. Pressure No-Load MSLB Transient
0
-500
-1000
-1500
-2000
-2500
psia
0
50
100
150
200
250
300
350
400
450
500
seconds
Design Specification

Question 2:

In your response 4 to the staff's request for additional information dated May 13, 2004, you indicated that EPU steam flow will not cause unacceptable vibration effects on the steam dryers, dryer supports, or flow deflector plate by comparing the thermal-hydraulic parameters with those at Palo Verde since Palo Verde and Waterford 3 have the same steam dryer design, and Palo Verde steam generators have been operating for nearly 20 years and had no dryer failures. You also indicated that the flow deflector is different from that of Palo Verde since it has two main steam outlet nozzles, the overall stress on this component remains small compared to its allowable value, and the natural frequency of the plate was shown in the analysis of record to be 34.7 Hz in comparison to the frequency likely to cause resonance (approximately 20 Hz). Describe what is the excitation frequency of 20 Hz regarding the nature of flow, acoustic loading, or fluid-elastic instability. Confirm whether the stress analysis has been performed for the dryer at Waterford 3. If so, provide the calculated stresses for the dryer components in comparison to the code allowable limits.

Response 2:

The Waterford 3 steam generator design specification requires that the steam generator be designed such that no damage is done to the component parts at frequency ranges of 19 to 20 Hz and 95 to 100 Hz. The lower frequency range is defined as a mechanical vibration (typically from the pumps and pump supports) and the upper frequency range is defined as a sinusoidal pressure variation of ±6 psi in the primary pipe that contains the reactor coolant pump. Each internal component of the Waterford 3 steam generator is specifically designed to avoid these frequency ranges.

Individual dryers are not susceptible to vibration or stress-related failures because they are bolted to each other on two sides (five one-half inch bolts on each side) and to a dryer support channel on the other two sides (three one-half inch bolts on each side). This arrangement ties the dryers and support structure together such that they act essentially as a single assembly. These 16 bolts ensure the structure of the dryers is very stiff and results in a high natural frequency. In addition to stiffening the structure, the bolts transfer loadings on the dryers to the support structure.

Westinghouse has not calculated the fundamental frequency of an individual dryer or the dryer/dryer support assembly. However, since the assembly is quite stiff and there have been no dryer failures in over 30 years of operating experience, it is assumed the fundamental frequency of this assembly is too high for the dryer to experience a resonant vibration.

Acceptable performance of the dryers at EPU conditions has been demonstrated through testing and from operational experience at other plants. The only significant loads on the dryer supports are from dead weight and seismic conditions. These loads are not affected by EPU conditions. Based on test data, the increase in steam flow and lower operating pressure associated with the EPU will increase the pressure differential across the dryers from approximately 0.15 psi to 0.19 psi. However, this loading condition is acting in the opposite direction from the dead weight load and results in a lower stress on the dryer supports than the zero-power condition. Hence, EPU conditions will not adversely affect the dryers or dryer supports.

Dryers installed in the Waterford steam generators are identical to those installed in all original steam generators designed by Combustion Engineering. The design was essentially the same as that used in the fossil power industry. Testing was performed on these dryers in the 1970s to ensure they would be suitable for the higher flows associated with the Palo Verde design. Flow testing was performed over a range of 30,000 to 60,000 lbs/hr at pressures of 600 to 1200 psia. Although this testing was primarily to determine the capability of the dryer to limit moisture carryover, any structural concerns would have been noted at that time. No structural failures were reported.

A stress analysis has been performed for the dryer supports but not for the dryers themselves. The Main Steam Isolation Valve / Turbine Stop Valve (MSIV/TSV) transient loads were not addressed in the original analysis of the dryers nor were they addressed for the power uprate. Flow loadings on the dryer were not considered to be significant since they were less than, and in the opposite direction from, the dead weight loading.

Westinghouse does not consider flow induced vibration or acoustic loadings in the dryer evaluation. Operating experience at other nuclear plants has shown that this dryer design is not susceptible to fatigue failure. For example, Palo Verde (before their power uprate) had calculated flow loadings per dryer of 59,859 lbs/hr at 989 psia. This flow load results in a steam density of 2.214 lbs/ft^3 and a velocity of 8.6 ft/sec for a dynamic pressure of 0.01776 lbf/in^2. After the EPU at Waterford, the flow loadings are calculated to be 51,232 lbs/hr at 803 psia. This flow load results in a steam density of 1.764 lbs/ft^3 and a velocity of 9.3 ft/sec for a dynamic pressure of 0.01633 lbf/in^2. Since Waterford will be operating with a dynamic pressure on the dryers approximately 8% lower than Palo Verde, there are no concerns with the structural integrity of the Waterford dryers at the extended power uprate conditions.

As noted earlier, the dryers in the Waterford steam generators are identical to those installed in all original steam generators designed by Combustion Engineering since the late 1960s. There has never been a failure in this dryer design in over 30 years of operation.

Question 3:

In your response 8, you mentioned Table 1 in Section 2.2.2.1.4.5.1 of the licensing report. Do you mean Table 2.2-5 rather than Table 1? In your summary table of stresses and CUFs for the main coolant loop piping, you indicated that the CUF of the hot leg pipe was calculated to be 0.382 due to Mechanical Nozzle Seal Assembly (MNSA) holes in the hot leg straight pipe. Discuss whether and how the MNSA installed on the hot leg are considered permanent and are there any other components (i.e., reactor vessel) also installing MNSA?

Response 3:

The reference to Table 1 should be a reference to Table 2.2-5.

MNSAs had been installed on the hot leg in refueling outage 9, but were removed in refueling outage 10. MNSAs are now authorized for installation only on pressurizer instrument nozzles and pressurizer heater sleeves for a maximum of two operating cycles during the current inspection interval for Waterford 3 (References 3-1 and 3-2). MNSAs are presently installed on two pressurizer heater sleeves and, under the restrictions of References 3-1 and 3-2, are not permanent repairs.

References:

3-1 NRC letter, R. Gramm (NRC) to M. Krupa (Entergy), "Waterford Steam Electric Station, Unit 3 – Re: Request for Relief from the Requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Concerning Authorization to Use New Design of Mechanical Nozzle Seal Assembly (MNSA) (TAC No. MB4272)," July 3, 2002

3-2 NRC Letter, N. Kalynanam (NRC) to M. Krupa (Entergy), "Waterford Steam Electric Station, Unit 3 – Correction to Authorization of Relief Request Re: Request for Relief from the Requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Concerning Authorization to Use New Design of Mechanical Nozzle Seal Assembly (MNSA) (TAC No. MB4272)," July 22, 2002

Question 4:

In your response 13, you indicated that the analysis of affected portions of the CCW system return piping downstream of the Shutdown Cooling heat exchangers is currently being evaluated for the expected temperature increase during normal plant shutdown. This is as committed to in the November 2003 licensing submittal in Section 2.5.5.3. You also indicated that all analyses are expected to be complete by August 3, 2004. Confirm whether the analysis has been completed. If so, provide the results of evaluation including stresses, CUFs and whether any modification is required for CCW piping and supports due to the power uprate.

Response 4:

The analysis of higher post-EPU component cooling water temperatures has not been completed. Reference 4-1 provides the status of the analysis, revises the commitment regarding the completion of the analysis, and describes compensatory measures that will remain in place until final resolution is identified and implemented.

As discussed in Reference 4-1, component cooling water (CCW) outlet temperature from the shutdown cooling (SDC) heat exchanger (Hx) has previously exceeded its 175°F design temperature. Engineering evaluations of the CCW SDC Hx outlet piping, pipe supports, and components have been performed to support continued operation and are summarized below. These evaluations determined that the system will remain operable with a CCW outlet temperature up to 225°F. Therefore, the compensatory actions currently in place are for Operations to monitor the CCW outlet temperature from the SDC Hx and maintain this temperature at or below 225°F. These compensatory actions have been incorporated into current operating procedures. A review of the available historical data for the CCW outlet temperature indicate that, prior to instituting procedural controls to monitor and control temperature, both the "A" and "B" trains had exceeded the 175°F design temperature on several occasions. The "B" train is known to have exceeded 225° F at least once previously and that was during a special steam generator cleaning evolution (i.e., a non-routine evolution).

<u>Effect on SDCHX Piping and Supports</u>

The current design temperature of the CCW Shutdown Cooling Heat Exchanger return piping is 175° F. This piping was structurally analyzed at 162° F up to the tie-in from the containment fan coolers (CFC's). The increase in temperature causes an increase in thermal stresses as well as support loads. The change from 162° F to 225° F increases thermal stresses and loads by about (225 - 70) / (162 - 70) = 1.68 or about 70%.

Pipe Stresses:
The change in thermal stress is acceptable. The maximum thermal stress in the piping between the SDC Hx and the tie-in with the CFC return lines is 10,407 psi. Increasing this by 70% results in a stress of 17,692 psi which is still less than the allowable of 22,500 psi.

Equipment Loads:
The most critical nozzle load on the SDC Hx's is the shear force of the "A" heat exchanger. Increasing the previously calculated shear load of 3,028 lbs by 70% results in a load of 5,148 lbs which is still less than the allowable of 5,523 lbs.

Support Loads:
The supports will also experience increases in thermal loads. Since the system was designed for 162° F, the higher temperature could adversely affect the support integrity. For the 10" return line, the maximum thermal load on any support is about 2,000 lbs. An increase of .70 x 2,000 lbs = 1,400 lbs is not considered significant for a seismically designed support for a 10" line.

Additionally, most of the supports are struts. There is built in slack and gaps in the supports along with flexibilities both at the supports and at the nozzles which were not accounted for in the original analysis. Thus, the piping will move prior to contacting a support which reduces the thermal loads. Furthermore, the higher thermal loads are the result of constrained thermal expansion between two restraints. A small amount of localized yielding of the support components will relieve the high support loads.

Effect on Components:
The Flow Diagram was used to generate a list of components downstream of the Shutdown Cooling Heat Exchangers that could potentially be subjected to an elevated temperature of 225° F and these components were reviewed. The limiting components are the ANSI B16.5, 150# Carbon steel flanged valves. Per the ANSI standard, these valves have a maximum operating pressure of 210 psig at 300° F. The CCW Design Pressure down stream of the SDC Hx is 125 or 150 psig.

Effect on Remaining CCW Return Piping

The evaluation shows that once the SDC Hx return flow is mixed with the CFC and safeguards pump return flows, the maximum temperature for the combined flow will be 203°F. This piping was analyzed between 167° F and 171° F. This is a maximum increase of 36° F. Entergy procedures for evaluating piping as-built tolerances state that increases of up to 40° F (on previously analyzed piping systems) requires no further evaluation. Additionally, Entergy procedures for evaluation of piping systems at low temperatures state that for justification of continued operation or interim operability evaluations, a cutoff temperature of 200° F may be used. The referenced Entergy standards/procedures are based on qualitative assessments of thermal analysis techniques. They cite, among other issues, the following factors which lead to overly conservative results for analyses at lower temperatures.

- Effects of gaps at supports
- Support stiffness effects on thermal expansion
- Equipment flexibility effects
- Column stability

The remaining components downstream of the SDC Hx and up to the dry cooling towers may experience temperatures up to 203° F. The Design Temperature of these components is 175°F. Per ANSI B16.5, the minimum pressure rating is 150#. As noted above, carbon steel valves have a maximum operating pressure of 210 psig at 300° F. The CCW Design Pressure down stream of the SDC Hx is 125 or 150 psig. Therefore, a temperature of 203° F is acceptable for these components.

The CCW pumps have a nameplate design temperature of 162° F. However, 200° F was used in the Design Report indicating the pump is qualified for operation at 200° F. The additional 3° is insignificant.

Effect on Room Heat Loads

Additional Engineering evaluations were performed for the higher room heat loads, resulting from the higher CCW outlet piping temperature. It was determined that the Shutdown Cooling Heat Exchanger Rooms and the Safeguard Pump Rooms remained within the capacity of the room coolers. Therefore, the design basis room temperature would not be exceeded when operating with a CCW outlet temperature of 225°F. However, the higher CCW temperature causes the heat load in the CCW Pump Rooms to exceed the capacity of the room coolers and in turn the CCW pump rooms "A" and "B" may potentially exceed 104°F slightly for a short duration (12 to 24 hours based on RCS cool-down rates). An evaluation performed in 1998 evaluated an elevated ambient temperature of 116°F in the CCW pump rooms and confirmed that the equipment would remain functional. Based on this prior evaluation it would be acceptable to slightly exceed 104°F for a short duration.

EPU Impacts

EPU will increase the decay heat load. The associated operating procedure maintains the CCW outlet temperature below 225°F when CCW outlet temperature from the SDC Hx can be monitored from the control room or from local/remote indication. (Note that Operations has control of both the primary system flow through the SDC Hx and CCW flow through the SDC Hx and thus the capability to control CCW outlet temperature to below 225°F.) In the event that CCW outlet temperature from the SDC Hx cannot be monitored, analysis has determined a maximum RCS cooldown rate versus time after shutdown curve that ensures that 225°F is not exceeded at pre-EPU conditions. As committed in the August 10, 2004 supplement, the compensatory action will be evaluated and updated, as necessary, to support post-EPU conditions.

References:

4-1 Entergy letter W3F1-2004-0068, K. Peters to USNRC Document Control Desk, "Supplement to Amendment Request NPF-38-240, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket Number 50-382, License No. NPF-38," August 10, 2004

Question 5:

In your response 12 to the staff's request for additional information dated May 13, 2004, you indicated that vibration monitoring will be performed on the main steam lines and main feedwater lines both before EPU and after EPU. Does this vibration monitoring include the branch lines on these systems?

Response 5:

Yes: Waterford 3 will monitor accessible branch connections off of the main steam and main feedwater lines, outside containment, for vibration. Accessible branch connections are those that do not require that scaffolding be built or insulation removed.

With respect to the main steam and main feedwater lines inside containment; there is one vent and two drains lines on each main feedwater line while each main steam line has branch lines associated with flow instrumentation and a vent line. In addition main steam line B has an additional drain line. These branch lines inside containment will not be monitored directly. Instead, these branch lines will be evaluated based on the vibration data obtained from the sensors installed on the main lines. If the vibration levels of the main lines do not increase significantly, it may be concluded that the same applies to the branch lines. Should vibrations in the main lines increase, analyses of the branch lines can be performed to deal with such increases.

As specified in the May 13, 2004 submittal, vibration monitoring and evaluation of measured data will be in accordance with ASME OM, Part 3, Operations and Maintenance of Nuclear Power Plants.

Attachment 2

To

W3F1-2004-0086

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Waterford 3 will monitor accessible branch connections off of the main steam and main feedwater lines, outside containment, for vibration. Accessible branch connections are those that do not require that scaffolding be built or insulation removed. With respect to the main steam and main feedwater lines inside containment; there is one vent and two drains lines on each main feedwater line while each main steam line has branch lines associated with flow instrumentation and a vent line. In addition main steam line B has an additional drain line. These branch lines inside containment will not be monitored directly. Instead, these branch lines will be evaluated based on the vibration data obtained from the sensors installed on the main lines. If the vibration levels of the main lines do not increase significantly, it may be concluded that the same applies to the branch lines. Should vibrations in the main lines increase, analyses of the branch lines can be performed to deal with such increases. As specified in the May 13, 2004 submittal, vibration monitoring and evaluation of measured data will be in accordance with ASME OM, Part 3, Operations and Maintenance of Nuclear Power Plants.	X		End of Cycle 13