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July 18, 2000

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

Subject: River Bend Station
Docket No. 50-458
License No. NPF-47
Additional Information Related to License Amendment
License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station

File No.: G9.5, G9.4.2

Reference:
1) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45077, dated July 30, 1999
2) U.S. Nuclear Regulatory Commission letter to EOI dated February 3, 2000 (TAC NO. MA6185)
3) U. S. Nuclear Regulatory Commission letter to EOI dated February 25, 2000 (Meeting Minutes of February 10, 2000 Meeting)
4) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45293, dated April 4, 2000
5) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45337, dated May 9, 2000

RBF1-00-0143
RBG-45428

Ladies and Gentlemen:

In the reference (1) letter, EOI requested a license amendment to NPF-47 and Appendix A – Technical Specifications, of the River Bend Station (RBS). This request is to extend operation of RBS from its current licensed power level of 2894 megawatts thermal (MWt) by five percent to an uprated power level of 3039 MWt.

The proposed changes were developed using generic guidelines for boiling water reactors (BWR) power uprates described in General Electric (GE) reports. In the reference (2) letter, the NRC requested additional information in 17 areas concerning the Mechanical & Civil Engineering Branch and the Electrical & Instrumentation Branch. The responses to the questions are included in Reference 4.

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EOI has received two additional questions that were sent as part of Reference (3). In addition to the formal questions, the NRC staff has requested information on a number of other issues concerning Power Uprate at RBS. The response to these questions are provided in Enclosure 2.

Commitments made in this letter are included as Enclosure 1.

EOI provided a supplement to the initial submittal (Reference 5). This supplement described a phased implementation of power uprate. It included proposed temporary conditions to allow on-line implementation of the flow only phase of the uprate. To ensure all necessary plant procedures, training and necessary modifications and configurations are complete prior to initiating the uprate implementation, EOI requests the start date of the requested 30 day temporary conditions be identified by EOI. EOI will continue to update the NRC regarding the expected start date and provide a minimum of 5 working days notice to the NRC.

If you have further questions, contact Mr. Barry M. Burmeister of my staff at 225-381-4148.

Sincerely,

Rick J. King

RJK/bmb
Enclosures

cc:

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Enclosure 1

Commitment Identification Form

Response to NRC Questions Related to River Bend Station Power Uprate Amendment
July 18, 2000
RBF1-00-0143
RBG-45428

COMMITMENT	ONE-TIME ACTION*	CONTINUING COMPLIANCE*
EOI will maintain the pool design limits (i.e., maximum temperature and corresponding heat removal capacity) by controlling the rate of the discharge to the spent fuel pool. This proposal is in lieu of the Standard Review Plan (SRP) criteria of Section 9.1.3		*X*
All identified changes to control room alarms, controls, and displays will be implemented prior to operation at uprated conditions.	*X*	
Power Uprate Testing will be performed as described on pages 38 & 39 of Enclosure 2 to this letter.	*X*	

*Check one only

Enclosure 2
Response to NRC Questions Related to River Bend Station Power Uprate Amendment
(TAC NO.MA6185)

Questions 1-3 are from the EOI NRC meeting February 10, 2000

Background for Questions 1-3

EOI's response to the requests concerning spent fuel pool decay heat loads involves committing to control the rate of offload of fuel in the unlikely event of a full core offload. Specifically, if a full core offload would be required in the future, EOI will maintain the pool design limits (i.e., maximum temperature and corresponding heat removal capacity) by controlling the rate of the discharge to the spent fuel pool. This proposal is in lieu of the Standard Review Plan (SRP) criteria of Section 9.1.3.

Both the SRP and the EOI proposal involve two decay heat evaluations. The first is for a normal outage with an offload of approximately 1/3 of the total 624 fuel bundles being transferred to the spent fuel pool. The second(abnormal) is the emergency full core offload evaluation. For the full core offload evaluation, RBS assumes 200 fuel bundles are placed in the upper fuel pool storage and the remaining 424 are transferred to the spent fuel pool. The change in this proposed approach will be the time to complete an emergency full core offload after startup from a refueling outage. The following table summarizes the pre and post uprate evaluations of the spent fuel pool decay heat removal.

Normal refueling offload	*Pre-uprate*	*Post-uprate*
Number of bundles offloaded	*248*	*248*
Heat load	*16.62E6 Btu/hr*	*16.04E6 Btu/hr*
Peak pool temperature	*139.8 F*	*< 139.8 F*
Total bundles contributing to heat load	*2680*	*744*

Note: The reload batch size of 248 bundles is conservative compared to the expected RBS reload size after power uprate.

Emergency Full Core Offload	*Pre-uprate*	*Post-uprate*
Number of bundles offloaded	*424*	*424*
Heat load	*24.68E6 Btu/hr*	*24.68E6 Btu/hr*
Peak pool temperature	*155.6 F*	*155.6 F*
Total number of bundles in pool after full core offload	*3104*	*3104*
Total number of days from initial shutdown	*45 (Per USAR)*	*60 (approximately)*

This approach to evaluating the full core offload case is proposed for the following reasons:

- *The guidance in the SRP to offload after 150 hours of operation (6 days) is not the result of a design basis accident (DBA) or event currently in the RBS design. As discussed in the response to question 1 the SRP specifies acceptable assumptions for the decay heat load for this evaluation for the normal maximum heat load. Note: RBS current design and licensing basis includes a total of 15 days to complete the core offload. This includes the time necessary to physically disassemble the vessel and to transfer the irradiated fuel to the spent fuel pool as discussed in the USAR Section 9.1.3.*

- *The guidance in the SRP to use 30 days as a standard refueling outage duration is no longer an outage restriction and therefore, not a limiting condition. EOI is currently expecting future RBS outages to be less than 30 days. A shorter outage results in a slightly higher decay heat load from the spent fuel. With the pool design temperature and heat removal being fixed, these limits will restrict the rate fuel can be placed into the spent fuel pool in the unlikely event of an early cycle full core offload.*

Question 1: As a result of plant operations at the proposed uprated power level, the decay heat load for any specific fuel discharge scenario will increase. The information is necessary to allow the staff to determine whether the analyses are consistent with the guidance described in Standard Review Plan, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." Please provide the following information:

Note: Actual spent fuel pool capacity is under review and may be increased in the future. The following responses apply to conservatisms in the analysis regarding current spent fuel pool cooling and cleanup system.

a. Provide the heat loads and corresponding peak calculated spent fuel pool (SFP) temperatures during planned refueling outages and unplanned full core off-load.

Response 1a: *The calculated heat load for a planned refuel outage is 16.62E6 Btu/hr. This results in a pool temperature of 139.8°F. This is the heat load and pool temperature that was calculated for pre-power uprate conditions. The pre-power uprate calculation conservatively assumed 2680 spent fuel bundles are in the spent fuel pool. This is 10 offloads of 248 and one of 200 spent fuel bundles offloaded in the refuel outages.*

Normal Refueling Evaluation: *If a reload consistent with Section 9.1.3, parts III.1.h.ii and iv, of the SRP is considered, decay heat loads from the latest 3 offloads are considered. For this calculation the refuel load was assumed to be 248 bundles (1/3 core, 208 bundles, plus an additional 40 bundles). The total number of spent fuel bundles in the pool using the SRP guidance becomes 744. This results in a heat load of 16.04E6 Btu/hr. This is less than the results currently in the USAR of 16.62E6 Btu/hr. Therefore, the power uprate evaluation will support a planned reload at uprated conditions while maintaining the pool temperature within the current 139.8°F.*

Full Core Offload Evaluation: *In the unlikely event an unplanned full core offload in the early portion of a fuel cycle is necessary there are no changes in assumptions from the pre-power uprate assumptions except that a higher heat load results due to power uprate. In this scenario, the reactor is assumed to require a full-core offload shortly into a fuel Cycle after completion of a normal refueling outage, such that the decay heat contribution from the fuel offloaded during that normal refueling outage remains a contributor to pool heat loads. The resulting full core offload heat load is 26.44E6 Btu/hr at 45 days after the start of the initial refueling outage. This is 7% higher than that determined for pre-power uprate. Assuming no changes in the number of bundles and limiting the*

spent fuel pool temperature to the pre-uprate value of 155.6 °F a full core (emergency) offload to the spent fuel pool can be completed within approximately 60 days of initial shutdown. By controlling the rate of full core offload EOI will ensure the existing spent fuel pool design value for temperature is maintained with the uprated core. This results in a small increase in the number of days after a normal refueling that a fuel core offload to the spent fuel pool can be completed from the current 45 days to approximately 60 days from the start of normal refueling shutdown. As noted above this limit will also maintain current temperature limits in the spent fuel pool. These conditions will support RBS's plans for reduced outage duration while not requiring extensive reanalysis of the Spent Fuel Pool and Cooling Systems to the higher design temperature conditions.

b. If the residual heat removal (RHR) system serves as a back-up system to the SFP cooling system, prior to a planned or unplanned full core offload event, how many trains of SFP cooling system and RHR system are required to be operable and available for SFP cooling?

Response 1b: *The RHR system does not serve as a back-up to the SFP cooling system. During a full core offload 200 bundles are assumed placed in the upper pool. In that situation RHR may be used to cool those bundles in the containment pool. However, RHR is not a back-up to the SFP cooling system. The calculations that determine the SFP temperature during planned offloads and unplanned offloads assume that 1 train of the Spent Fuel Pool Cooling System is in operation.*

c. Discuss the provisions that have been established in the plant operating procedures to ensure that the RHR system will be aligned for SFP cooling.

Response 1c: *The RHR system can not be aligned to cool the SFP. However, the RHR system may be aligned to cool the containment pool during an unplanned offload when 200 spent fuel bundles are placed in the containment pool. In Abnormal Operating Procedure (AOP) 0051, Loss of Decay Heat Removal, there are instructions to ensure adequate decay heat removal capability exists for fuel in the reactor vessel and for irradiated fuel in the containment or spent fuel pool. If containment pool cooling is lost, then train B of the Spent Fuel Cooling (SFC) System can be placed in service on the containment pool or the RHR system can be aligned in the Fuel Pool Cooling Assist Mode. If SFP cooling is lost then the other train of SFC can be aligned to cool the SFP. If both trains of SFC are lost then feed and bleed of the SFP can be done using the fuel pool purification pumps. The SFP is fed from the condensate storage tank and the pool is bled to the condensate storage tank, condenser hotwell, or radwaste.*

Question 2: As stated in the Updated Final Safety Analysis Report (UFSAR), the SFP cooling system is designed to maintain the SFP at or below 139.8 F with a decay heat load of 16.62 x 10^6 Btu/hr from all the previously discharged Spent Fuel Assemblies (SFAs) and a freshly discharged partial (approximately 1/2) core. Also, as stated in the UFSAR, in an event of an unplanned (emergency) full core offload, the SFP temperature will be maintained below 155.6 F with a decay heat load of 24.68 x 10 6 Btu/hr from all the previously discharged SFAs and a freshly discharged full core. As a result of plant operations at the proposed uprated power level, the decay heat load and its corresponding peak calculated SFP temperature for any specific fuel discharge scenario will increase slightly. Discuss the effects of the elevated pool temperatures during planned refueling outages and unplanned full core off-load events on SFP (i.e., structures, SFP linings, etc.) and the SFP cooling and cleaning systems.

The preceding information is necessary to allow the staff to determine whether the design of the SFP (i.e., structures, SFP linings, etc.) and the SFP cooling and cleaning systems is consistent with the guidance described in Standard Review Plan, Section 9.1.3.

If an entire core off-loaded to the SFP is the normal practice during planned refueling outages at River Bend Station, a single failure of the SFP cooling system should be assumed in the SFP thermal analysis for the planned refueling outages. A single failure of the SFP cooling system need not be assumed for the unplanned full core off-load events.

Response 2: *Power uprate will not impact previous analysis results for determining the spent fuel pool temperature during a planned offload. This is described in EOI Response 1a. Note: A full core offload is not the normal refueling practice at RBS.*

For the unplanned (emergency) full core offload there is no change in assumptions from the pre-power uprate assumptions except that a higher heat load results due to power uprate. The resulting full core offload heat load is 26.44E6 Btu/hr at the current 45 days from initial shutdown, 7% higher than current (pre-power uprate) calculated heat load. By maintaining the current pre-power uprate temperature limit of 155.6°F in the spent fuel pool an increase to approximately 60 days in the time to complete the full core offload will maintain the heat load within the previously calculated 24.68E6 Btu/hr. As stated in Response 1a, in the unlikely event a full core offload early in a fuel cycle is necessary RBS will control the time of a full core offload to ensure the original design conditions are maintained.

Question 3: In the unlikely event that there is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling to boiling) and the boil-off rate (based on the highest heat load from the planned or unplanned full core off-load). Also, discuss sources and capacity of make-up water and the methods/systems (indicating system seismic design Category) used to provide the make-up water.

The above information is necessary to allow the staff to determine whether the analyses are consistent with the guidance described in Standard Review Plan, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

Response 3: *There are two alarms associated with spent fuel pool temperature. The first alarm occurs when pool temperature rises to 135°F and the second alarm occurs when pool temperature rises to 151°F. At 135°F operations personnel take action to ensure cooling of the spent fuel pool is maintained. From 135°F it takes 10.4 hours to reach boiling. This is based on a heat load of 16.62×10^6 Btu/hr for a planned offload. This also assumes that gates are open in the spent fuel pool, which provide maximum water inventory if spent fuel pool cooling is lost; this is the normal configuration of the spent fuel pool. This time to boil is conservative in that only the water inventory above the spent fuel is included, not the water that surrounds the fuel. For an unplanned full core offload the heat load is 26.44×10^6 Btu/hr. With this heat load, from 135°F it takes 6.6 hours to reach boiling if the gates are open at a boil-off rate of approximately 3300 gals/hr. The boil off rate is determined using the unplanned full core offload heat load, the higher heat load between the planned and unplanned full core offload.*

Other sources of makeup to the spent fuel pool include the condensate storage tank, which has a maximum of 495,000 gallons available for discharge to the spent fuel pool, and the Makeup Water System, which can supply approximately 700,000 gallons from the demineralizer water storage tanks. These sources are both classified as non-seismic. In the event of a station blackout, the fire protection system, which has a capacity of approximately 600,000 gallons, may be used to fill the spent fuel pool. Each of these sources are addressed in current procedures.

Questions 4-6 are from fax received 4/24/2000

Question 4: In GE Report NEDC-32778, Section 4.1.1.1 (b) for Local Pool Temperature with SRV Discharge, it is indicated that NEDO-30832 provides justification for elimination of local pool temperature limit for SRV discharge with quenchers. Please indicate how this limit is met by River Bend such as location of ECCS suction strainers with respect to quenchers' elevation.

Response 4: *Although NEDO-30832-A was referenced as providing the basis for elimination of the local suppression pool temperature limit, an evaluation for local suppression pool temperature limits was also performed per NUREG-0783. The NUREG-0783 value of 20 °F of local suppression pool subcooling was demonstrated. Further, it was determined that the River Bend SRV X-quencher elevations were at an elevation above the ECCS pump suction elevations. As a result, local pool temperature limits for SRV discharge with quenchers are not required.*

Local suppression pool temperature limits were imposed in NUREG-0783 for plants with quencher devices to ensure stable steam condensation without significant containment loads. After NUREG-0783, SRV X-quencher test data was compiled to confirm that stable condensation was ensured even with local pool temperatures approaching saturation conditions. The X-quencher test data is compiled in NEDO-30832-A (Reference 1). NRC reviewed and approved NEDO-30832-A in Reference 2 and provided justification for elimination of the local pool temperature limit for plants with X-quenchers. The NRC raised an additional concern with regard to the transfer of non-condensed SRV steam to the ECCS suction strainer if the ECCS suction strainer is at a higher elevation than the SRV quencher. The NRC stated in Reference 2 that local pool temperature limits can be eliminated if the plant has emergency pump inlets located below the elevation of the quencher elevation. From a comparison of River Bend USAR Figures 6A.16-2 and A.6A.4-1, it was determined that the quencher centerline is at an elevation above the ECCS pump suction elevations. Therefore, the results of NEDO-30832-A were applicable to River Bend. Nonetheless, River Bend performed analysis during power uprate to demonstrate continued compliance with the NUREG-0783 local suppression pool temperature limits (to demonstrate 20 °F of local suppression pool subcooling).

***Response 4 References**:*

1. *NEDO-30832-A, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," May 1995. (Includes copy of NRC Safety Evaluation Report).*

2. *USNRC, "Safety Evaluation Report by the Office of Nuclear Reactor Regulation on the Review of Two GE Topical Reports for the Elimination of Local Temperature Limits and Raising Pool Temperature Technical Specification Limits," Transmitted with Letter from G. Halahan (NRC) to R. Pinelli (BWROG), August 29, 1994*

Question 5: Please provide the containment analyses maximum pressure and temperature curves at uprated power after a LOCA. Discuss the limiting initial conditions assumed for the DBA LOCA and for the sensitivity case for the MSLB for peak drywell pressure as indicated in Section 4 1.1.3. Also provide the containment drywell pressure limit in psig.

Response 5: *Drywell & Containment Pressure Temperature Plots are attached at the end of this response. Please note that these figures are considered GE proprietary information. The original USAR analysis was based upon a nominal set of initial conditions. These results were supplemented by a sensitivity analysis to establish the range of acceptable conditions for the containment (Technical Specifications 3.6.1.4, 3.6.1.5, 3.6.2.1, 3.6.5.4, and 3.6.5.5). The following table summarizes the initial conditions, and key results for the nominal case, and significant sensitivities.*

Original USAR Analysis				
Parameter	*Nominal*	*Peak $P_{DW-Cont.}$ Sensitivity*	*Peak T_{DW} Sensitivity*	*Peak $P_{Cont.}$ Sensitivity*
Drywell Pressure (psig)	*0.0*	*0.0*	*0.0*	*1.5*
Drywell Temperature (°F)	*135.0*	*100.0*	*145.0*	*100.0*
Drywell Dew Point Temperature (°F)	*109.8*	*60*	*60*	*60*
Drywell Relative Humidity (%)[3]	*50.0*	*27.0*	*7.8*	*27.0*
Containment Pressure (psig)	*0.0*	*0.3*	*0.3*	*0.3*
Containment Temperature (°F)	*90.0*	*100.0*	*100.0*	*70*
Containment Dew Point Temperature (°F)	*68.9*	*60*	*60*	*60*
Containment Relative Humidity (%)[3]	*50.0*	*27.0*	*27.0*	*70.5*
Suppression Pool Temperature (°F)	*100.0*	*100.0*	*100.0*	*100.0*
Peak Drywell Internal Pressure (psid)[1]	*18.63*	*19.16*	*19.10*	*N/A*
Peak Drywell Temperature (°F)[1]	*316.0*	*296.2*	*328.31*	*N/A*
Peak Containment Pressure (psig)[2]	*6.31*	*N/A*	*N/A*	*7.6*

1. *Main Steam Line Break limiting event*
2. *Recirculation Line Break limiting event*
3. *Relative humidity values inferred from the dew point temperatures*

The DBA-LOCA initial conditions for the short-term containment analyses were developed to be consistent with the USAR with the exception that the initial drywell temperature was set at the current maximum operating Technical Specification value of 145°F. Additionally, the initial conditions in the containment were set up such that thermal equilibrium existed between the suppression pool and containment airspace and saturation conditions existed in the containment airspace. The latter conditions are required by the GE M3CPT computer model which imposes these conditions at all times to maximize the containment pressure.

The initial conditions for the MSLB sensitivity case are consistent with those identified in USAR Section 6.2.1.1.3.1.2. These conditions result in a higher non-condensable gas mass in the drywell and an initial drywell-to-containment pressure difference, both of which produce a higher peak drywell pressure.

The following table summarizes the uprate short term "nominal" case and the "sensitivity" case both of which are run to determine the drywell response. Also included are the conditions for and the results of the uprate long-term "nominal" case for the response of the containment. The long term evaluation was performed with the GE SHEX computer program.

Power Uprate Analysis			
Parameter	*Nominal (Short Term)*	*Sensitivity (Short Term)*	*Nominal (Long Term)*
Drywell Pressure (psig)	*0.0*	*0.0*	*0.0*
Drywell Temperature (°F)	*145.0*	*100.0*	*145.0*
Drywell Relative Humidity (%)	*50.0*	*27.0*	*50.0*
Containment Pressure (psig)	*0.0*	*0.3*	*0.0*
Containment Temperature (°F)	*100.0*	*100.0*	*90.0*
Containment Relative Humidity (%)	*100.0*	*100.0*	*50.0*
Suppression Pool Temperature (°F)	*100.0*	*100.0*	*100.0*
Peak Drywell Internal Pressure (psid)[1]	*20.5*	*20.7*	*N/A*
Peak Drywell Temperature (°F)[1]	*332.8*	*307.2*	*N/A*
Peak Wet Well Pressure (psig)[1]	*7.8*	*9.3*	*N/A*[3]
Peak Containment Pressure (psig)[2]	*N/A*[3]	*N/A*[3]	*3.6*
1. Main Steam Line Break limiting event			
2. Recirculation Line Break limiting event			
3. This parameter is not calculated.			

The following table summarizes the peak values for the various containment parameters of interest, and the analysis which generated the peak values.

Parameter	*Design Limit*	*Uprate*	*Uprate Case*
Peak Drywell Temperature (°F)	*330*	*332.8*	*Short Term - Main Steam Line Break – Nominal Initial Conditions*
Peak Containment Temperature (°F)	*185*	*123.8*	*Long Term – Main Steam Line Break & Recirculation Line Break*
Peak Drywell Differential Pressure (psid)	*25*	*20.7*	*Short Term - Main Steam Line Break – Technical Specification Initial Conditions*
Peak Wetwell Pressure (psig)	*15*	*9.3*	*Short Term - Main Steam Line Break – Technical Specification Initial Conditions*
Peak Containment Pressure (psig)	*15*	*3.6*	*Long Term – Main Steam Line Break & Recirculation Line Break*
Peak Suppression Pool Temperature (°F)	*185*	*170.7*	*Long Term – Main Steam Line Break*

In the table above, the peak drywell temperature is shown to exceed the design limit by approximately 3 °F. This is discussed in Section 4.1.1.3 of NEDC-32778P. The computer program used to calculate the short term response does not include the impact of heat sinks on the temperature rise. If heat sinks are considered, the peak drywell temperature will not exceed 330 °F.

The current value of P_a used for containment testing is 7.6 psig, and is based upon the containment response assuming Technical Specification initial conditions. This value of Pa bounds the peak containment pressure calculated for uprate. Therefore, the value of P_a for containment testing will remain 7.6 psig.

Summary Of Plots

B-3 *DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)*

B-4 *DBA-LOCA Short-Term Drywell and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)*

B-11 *DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)*

B-12 *DBA-LOCA Short-Term Drywell and Containment Airspace Temperature. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)*

C-9 *DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V)*

C-10 *DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V)*

C-11 *DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, t >1800 sec, (SHEX-04V)*

C-12 *DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, t > 1800 sec, (SHEX-04V)*

D-5 *DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)*

D-6 *DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)*

H-3 *SBA - 0.01 ft^2 Steam Break, Long-Term Drywell and Containment Airspace Pressure. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)*

H-4 *SBA - 0.01 ft^2 Steam Break, Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)*

H-7 *IBA - 0.1 ft^2 Steam Break, Long-Term Drywell and Containment Airspace Pressure. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)*

H-8 *IBA - 0.1 ft^2 Steam Break, Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)*

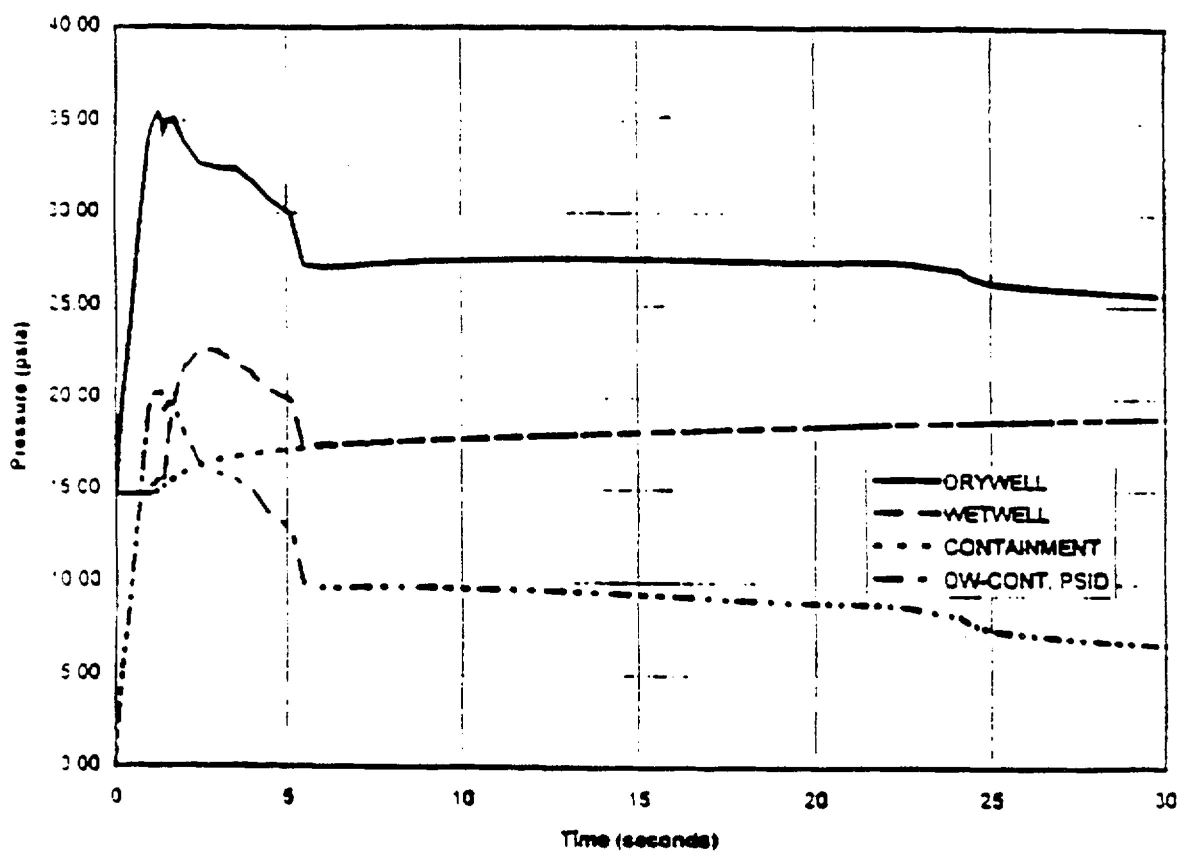


Figure B-3 DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)

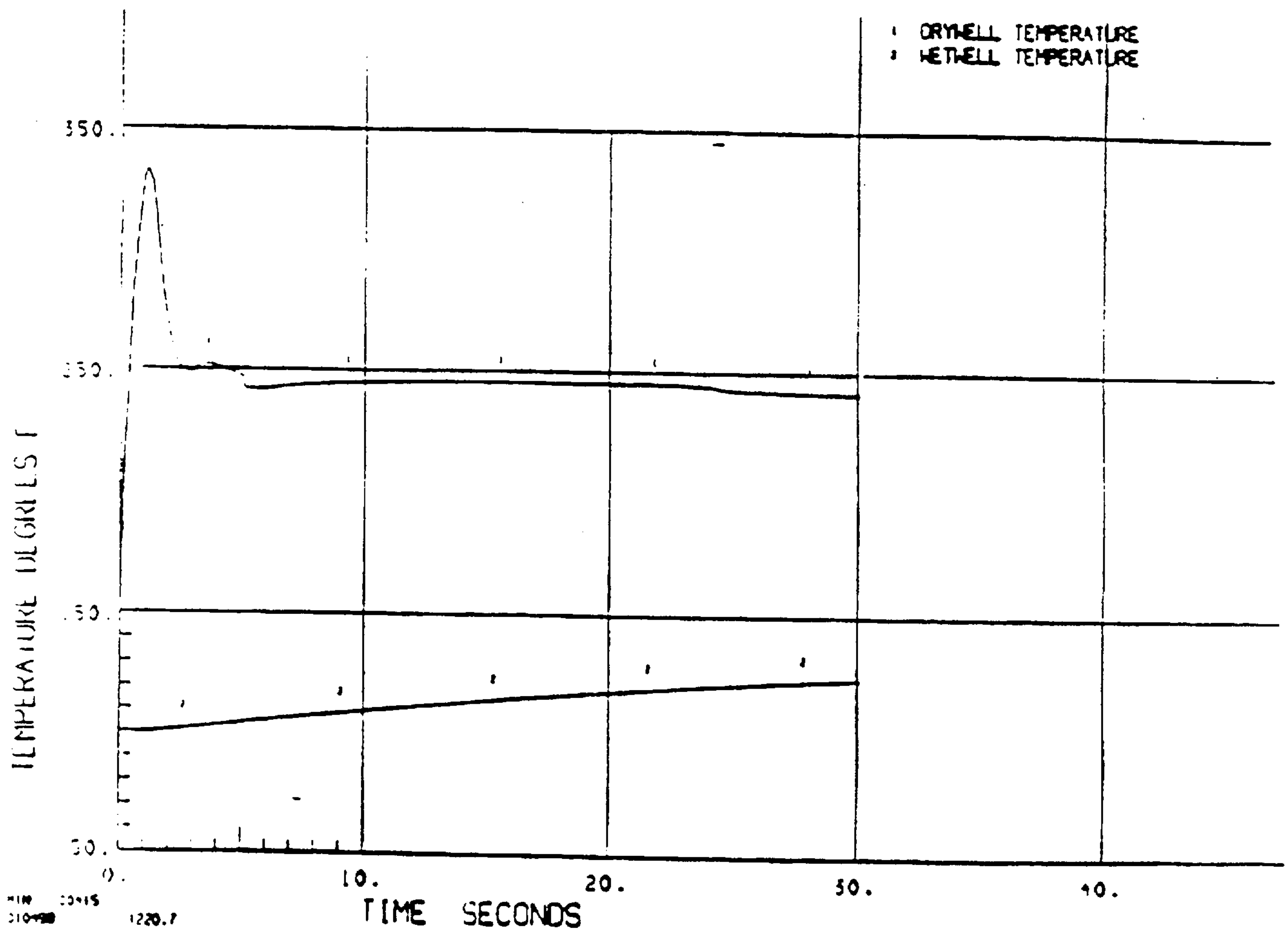


Figure B-4 **DBA-LOCA Short-Term Drywell and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)**

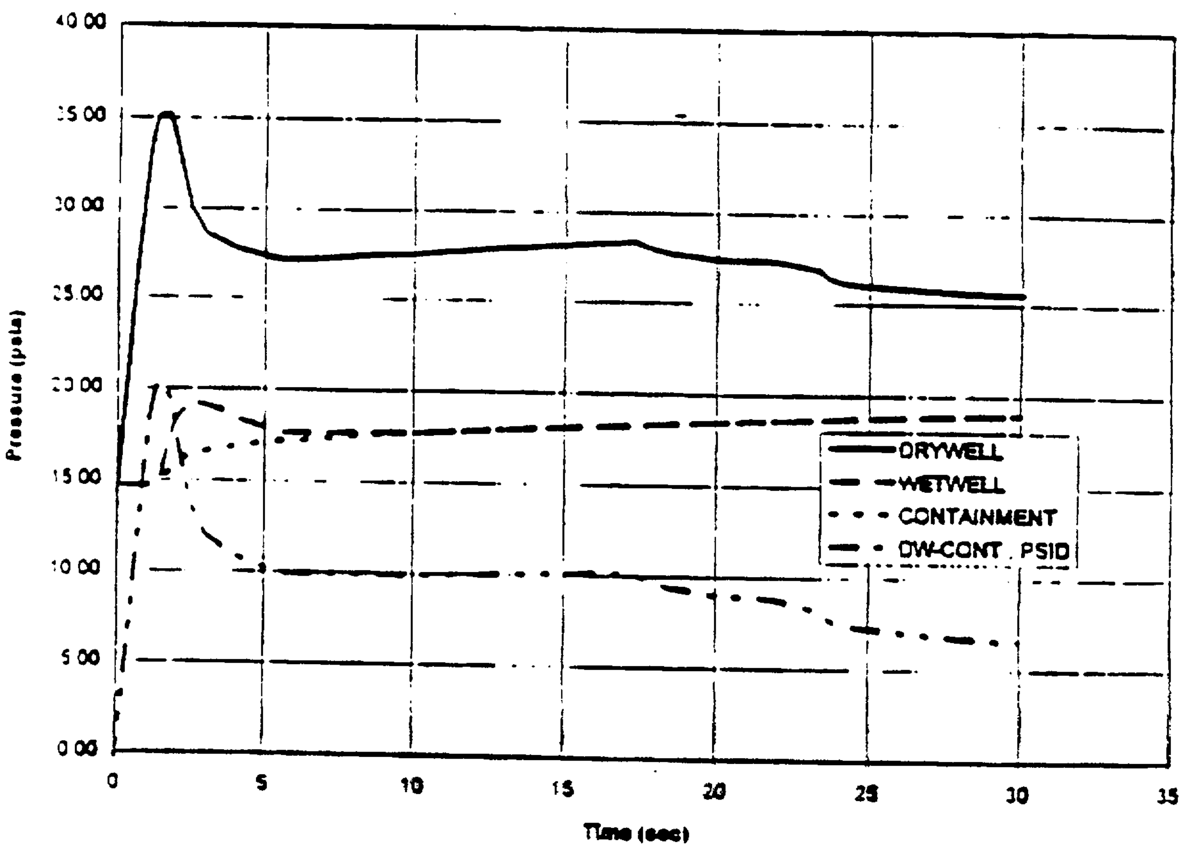


Figure B-11 DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)

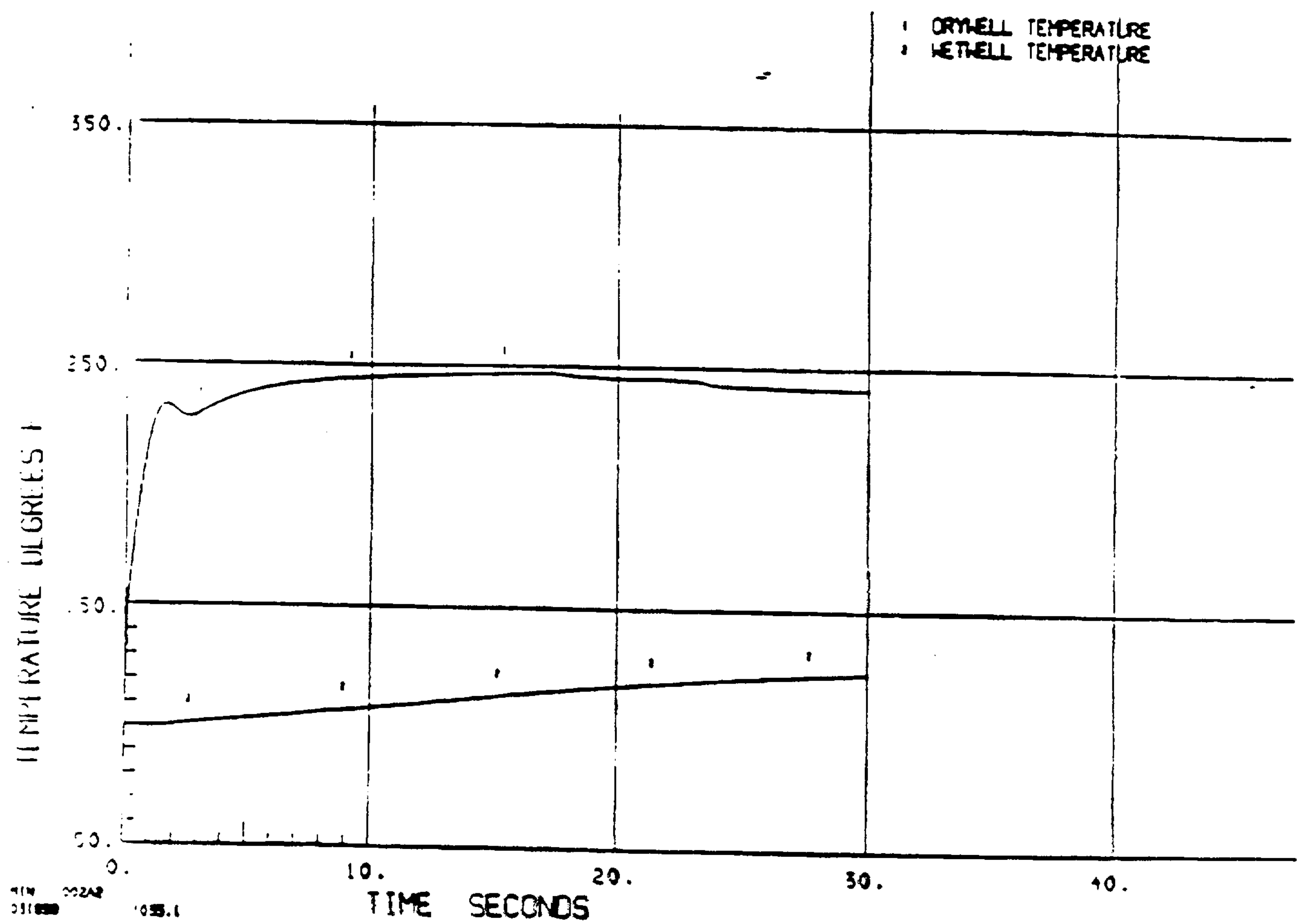


Figure B-12 DBA-LOCA Short-Term Drywell and Containment Airspace Temperature. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)

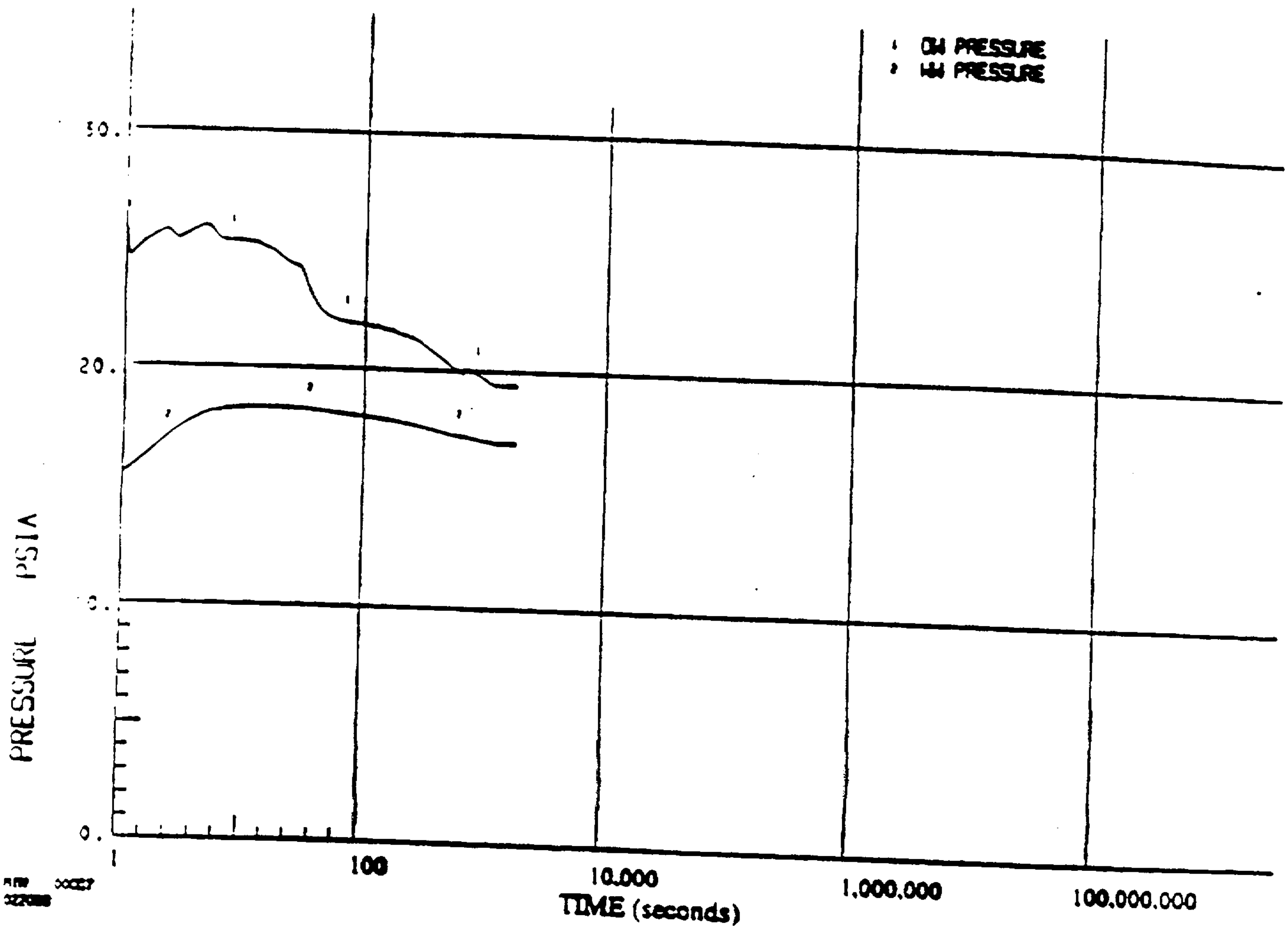


Figure C-9 DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V)

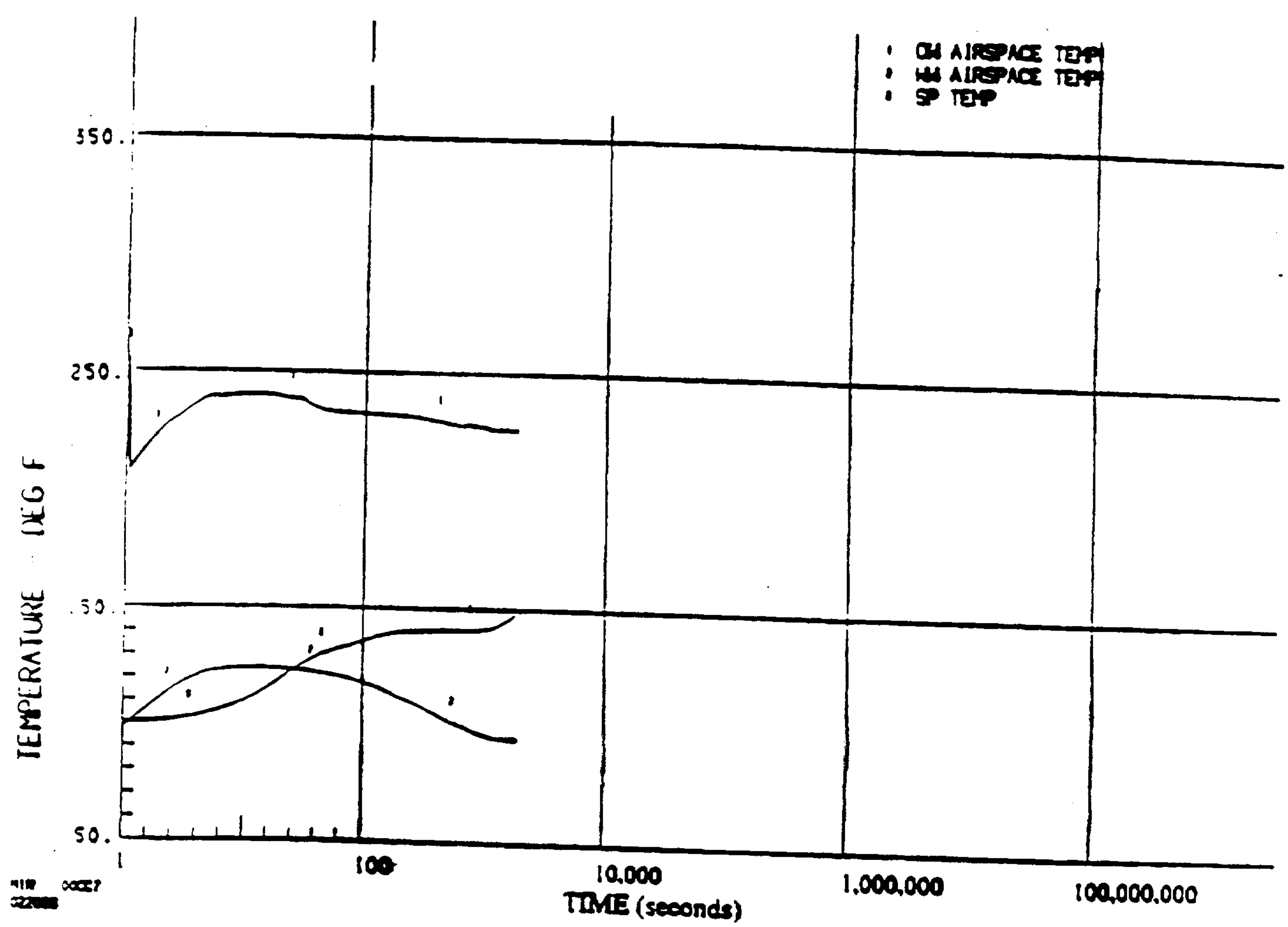


Figure C-10 DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V)

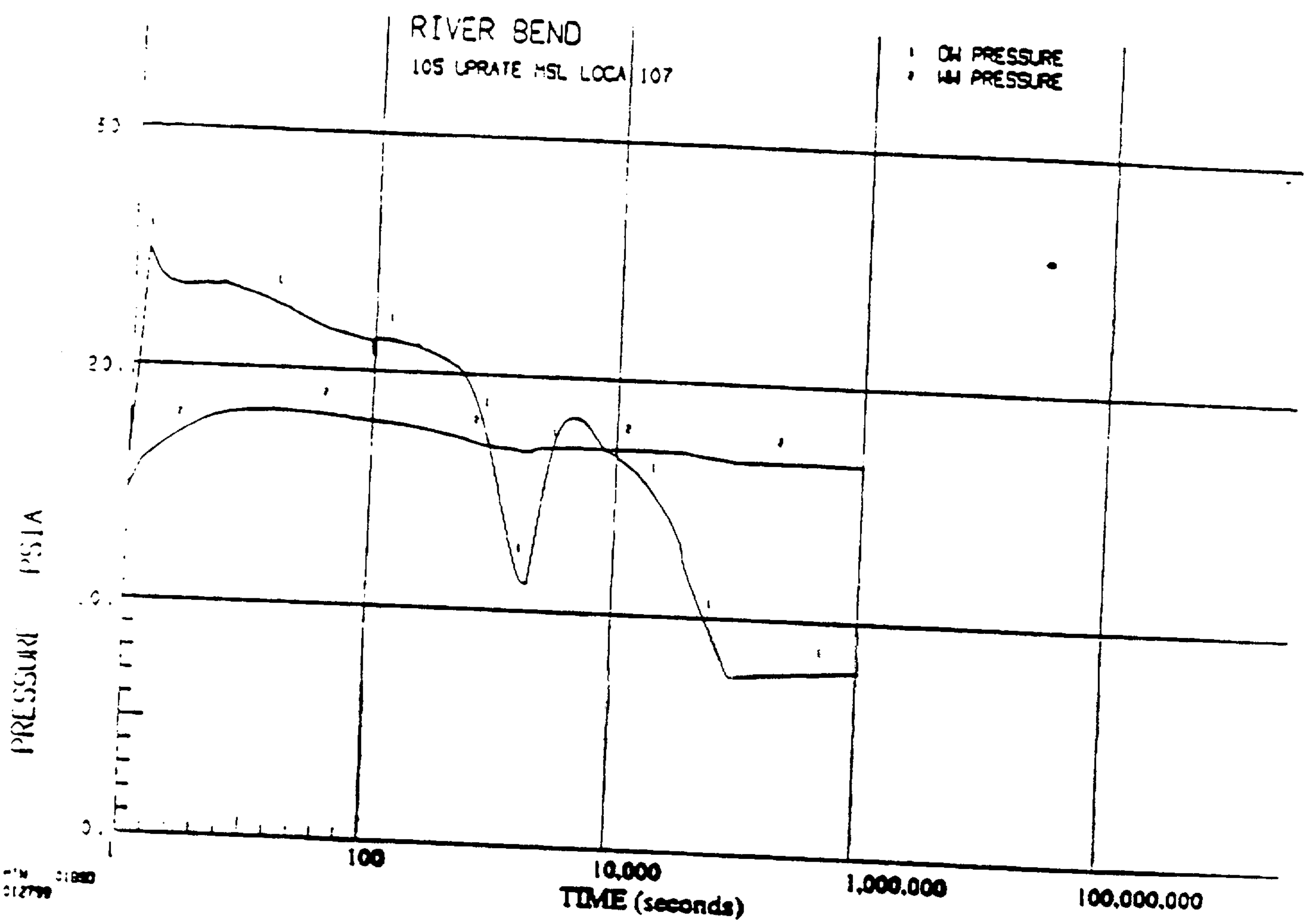


Figure C-11 DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, t >1800 sec, (SHEX-04V)

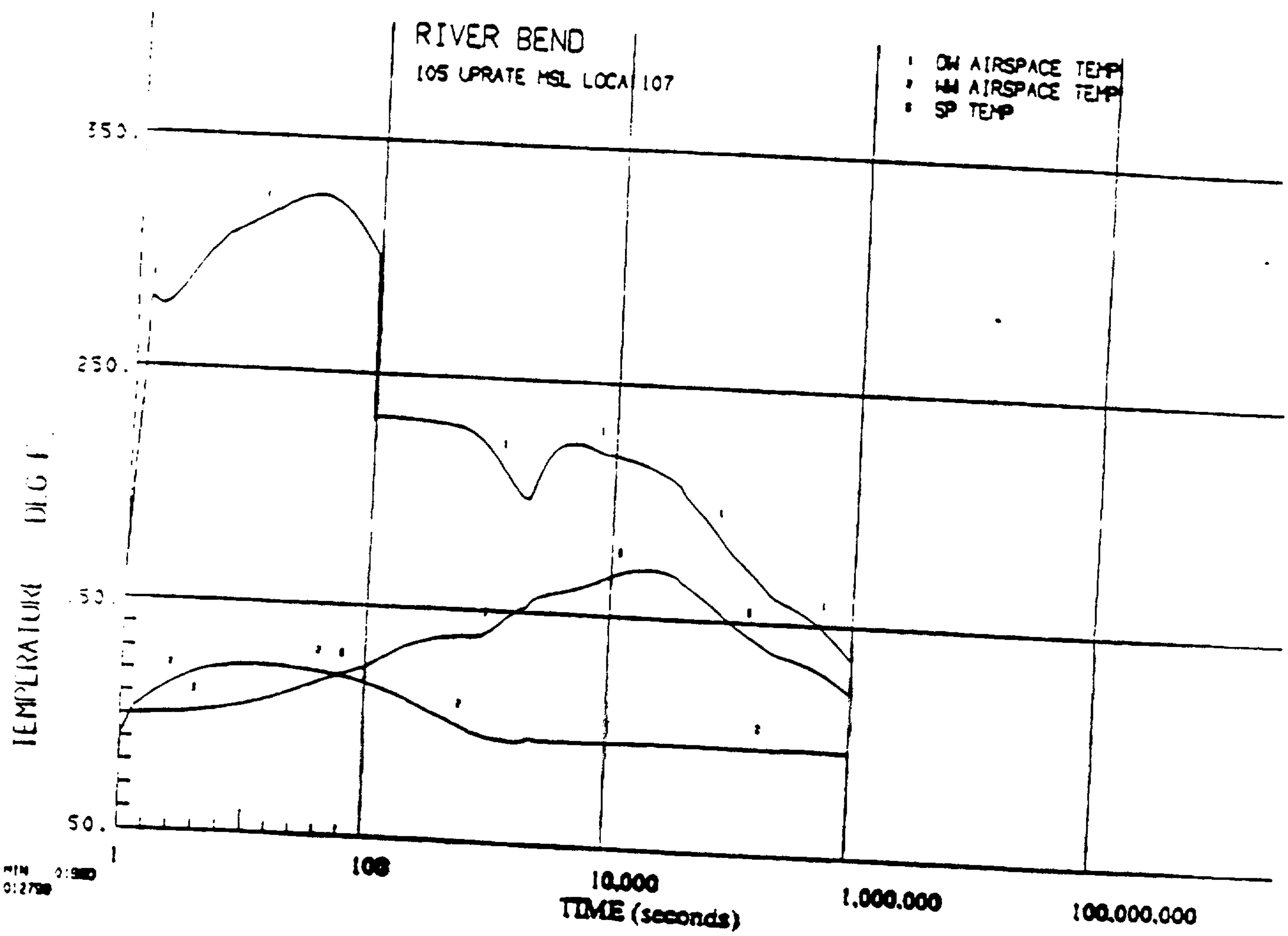


Figure C-12 DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, t > 1800 sec, (SHEX-04V)

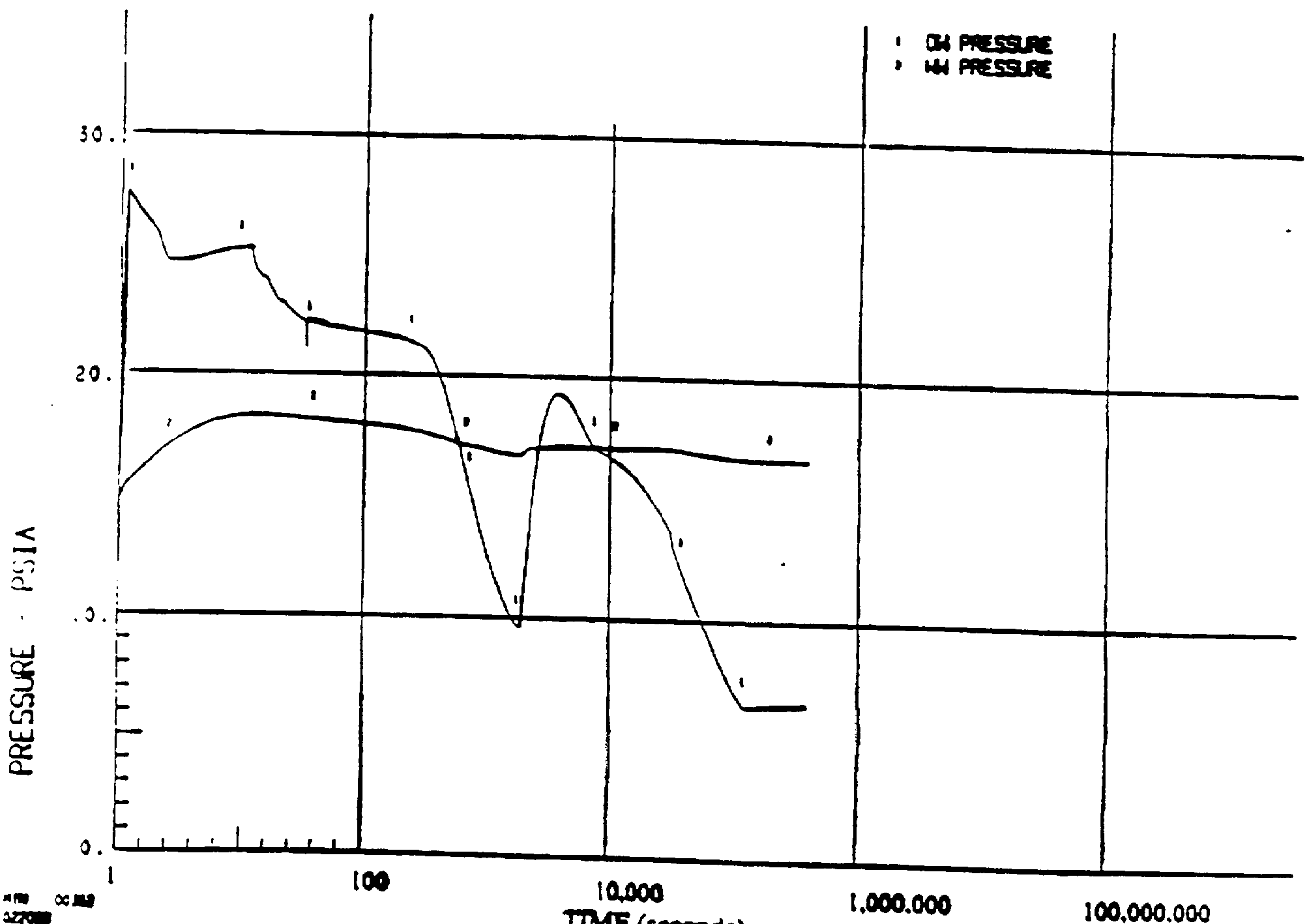


Figure D-5 DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)

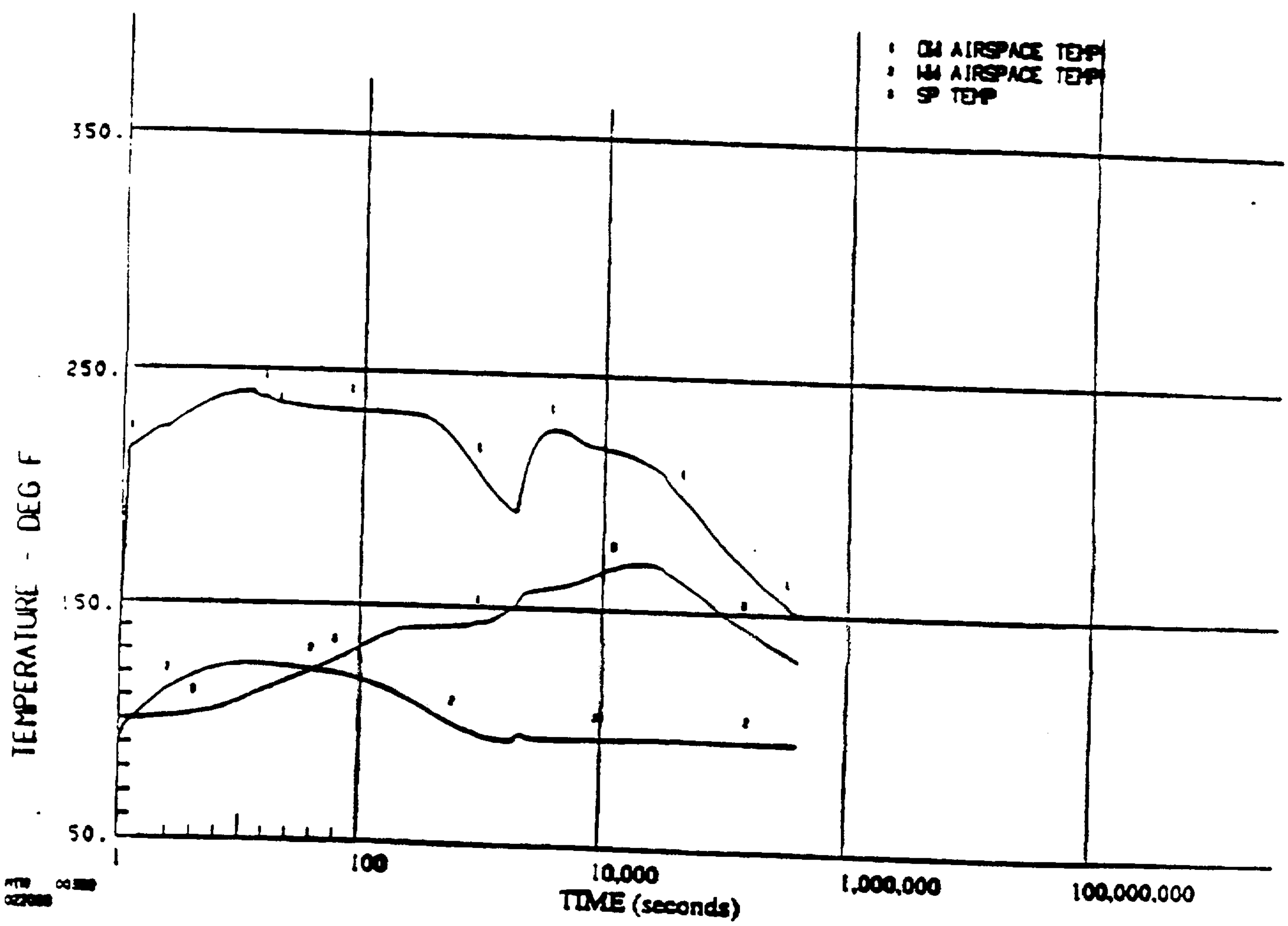


Figure D-6 DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)

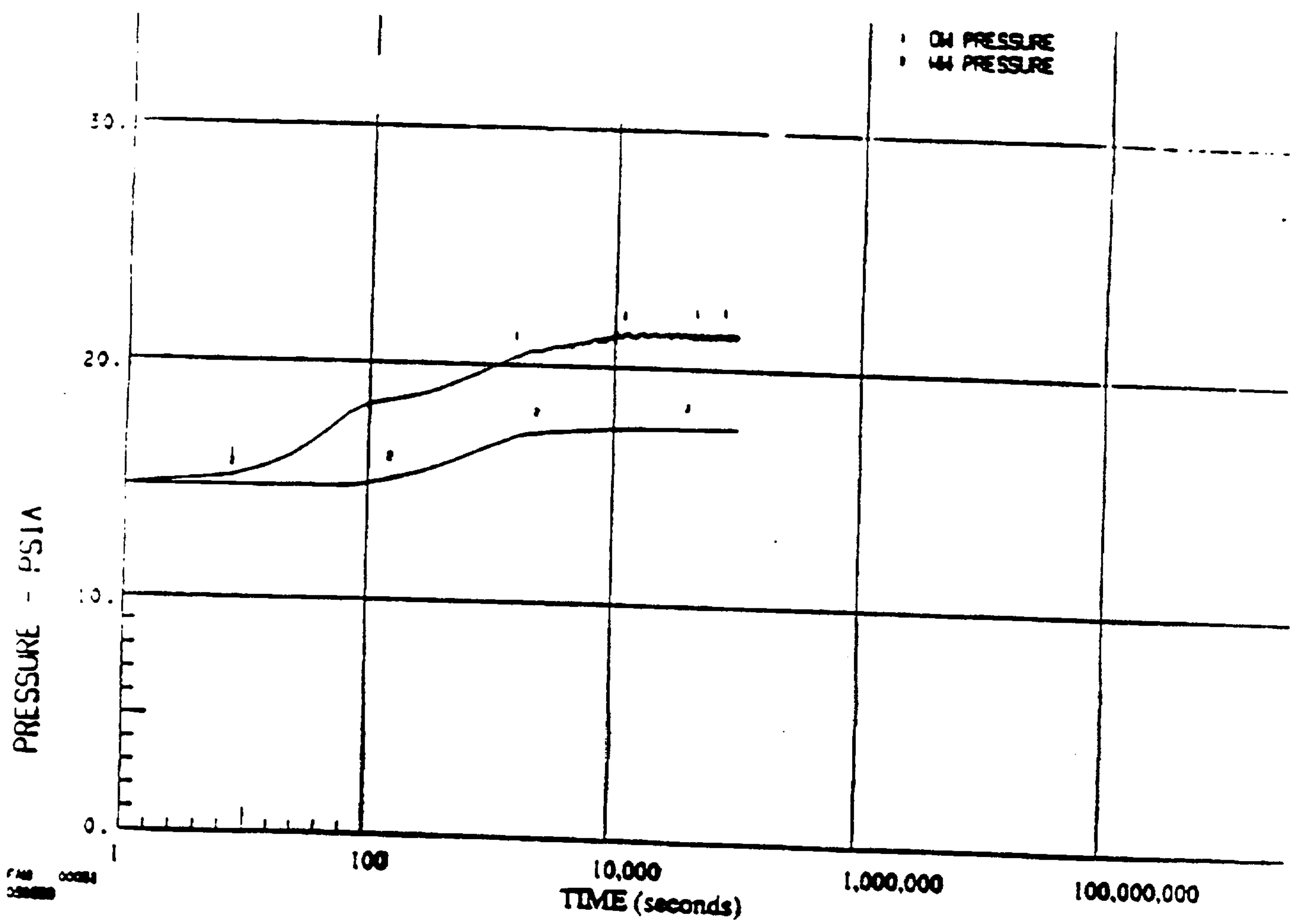


Figure H-3 SBA - 0.01 ft² Steam Break, Long-Term Drywell and Containment Airspace Pressure. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)

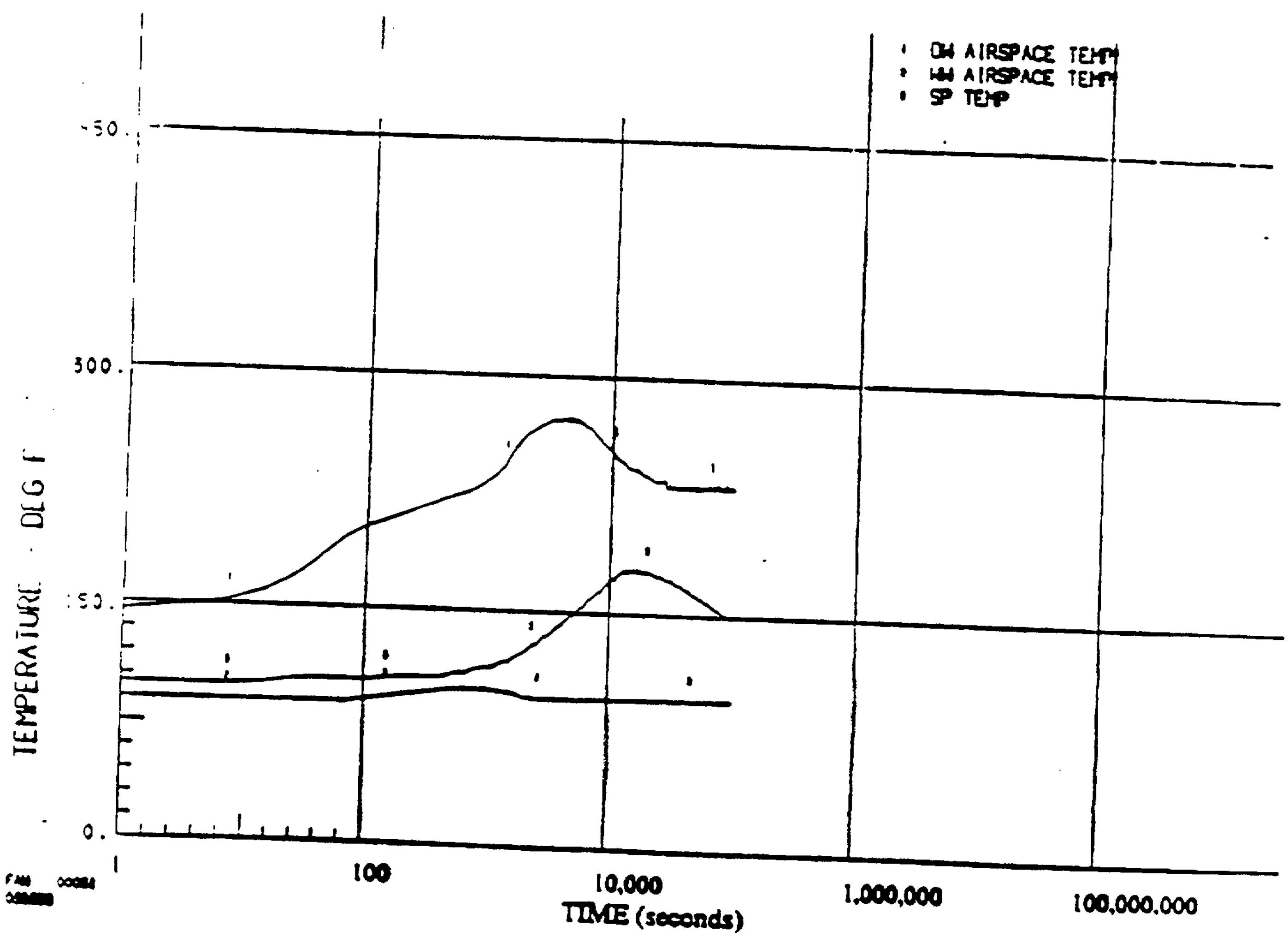


Figure H-4 **SBA - 0.01 ft² Steam Break, Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)**

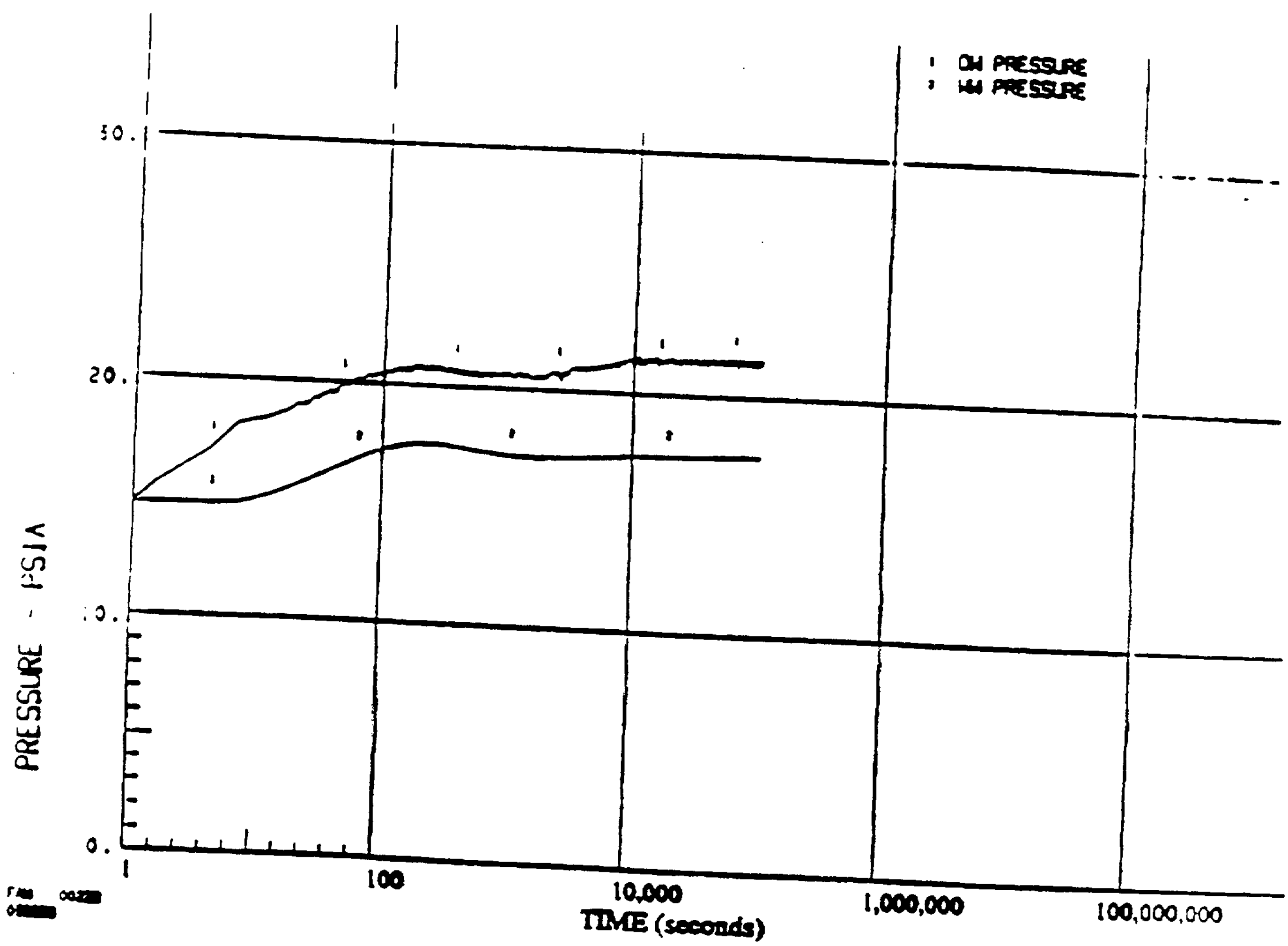


Figure H-7 IBA - 0.1 ft² Steam Break, Long-Term Drywell and Containment Airspace Pressure. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)

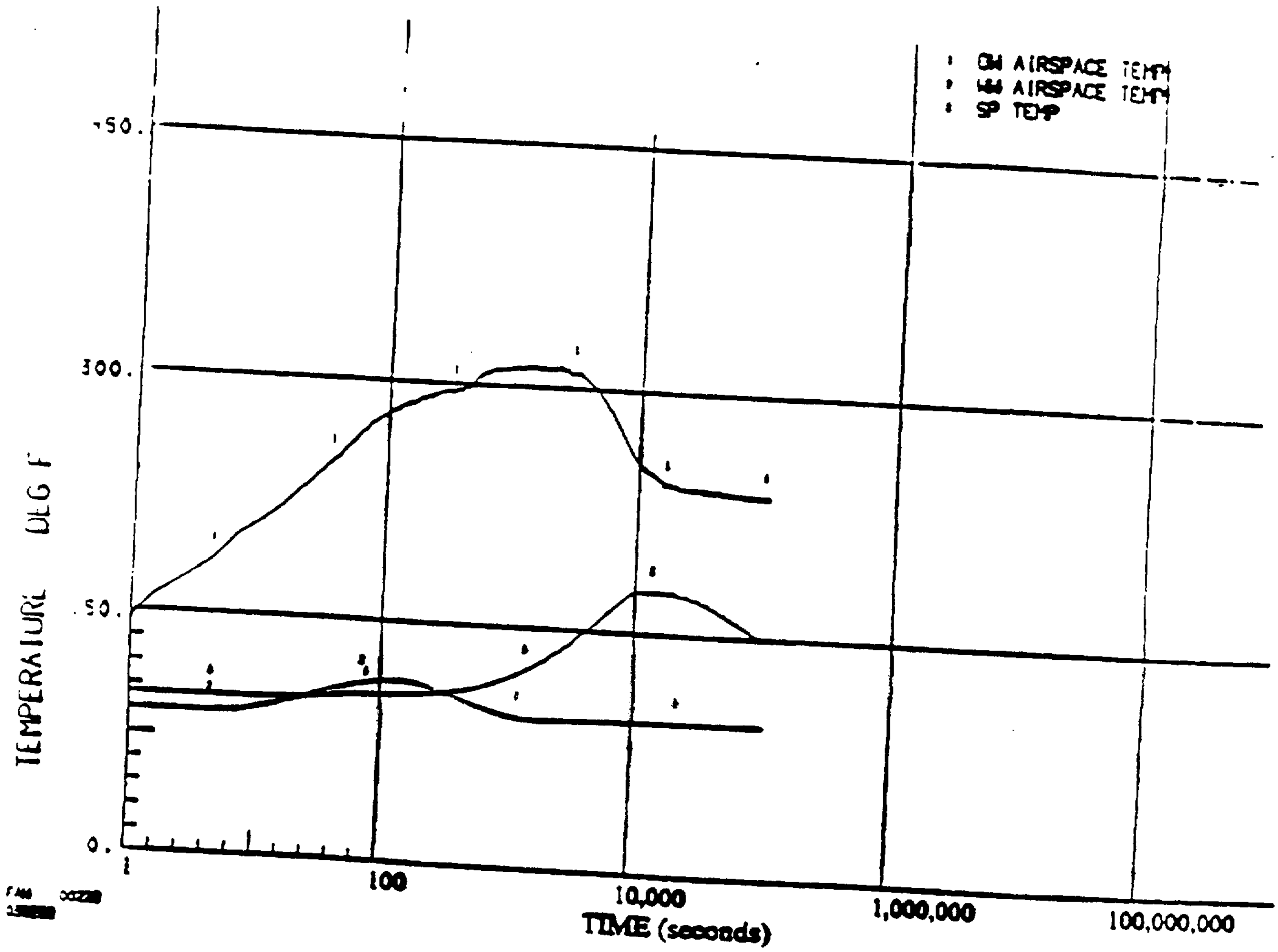
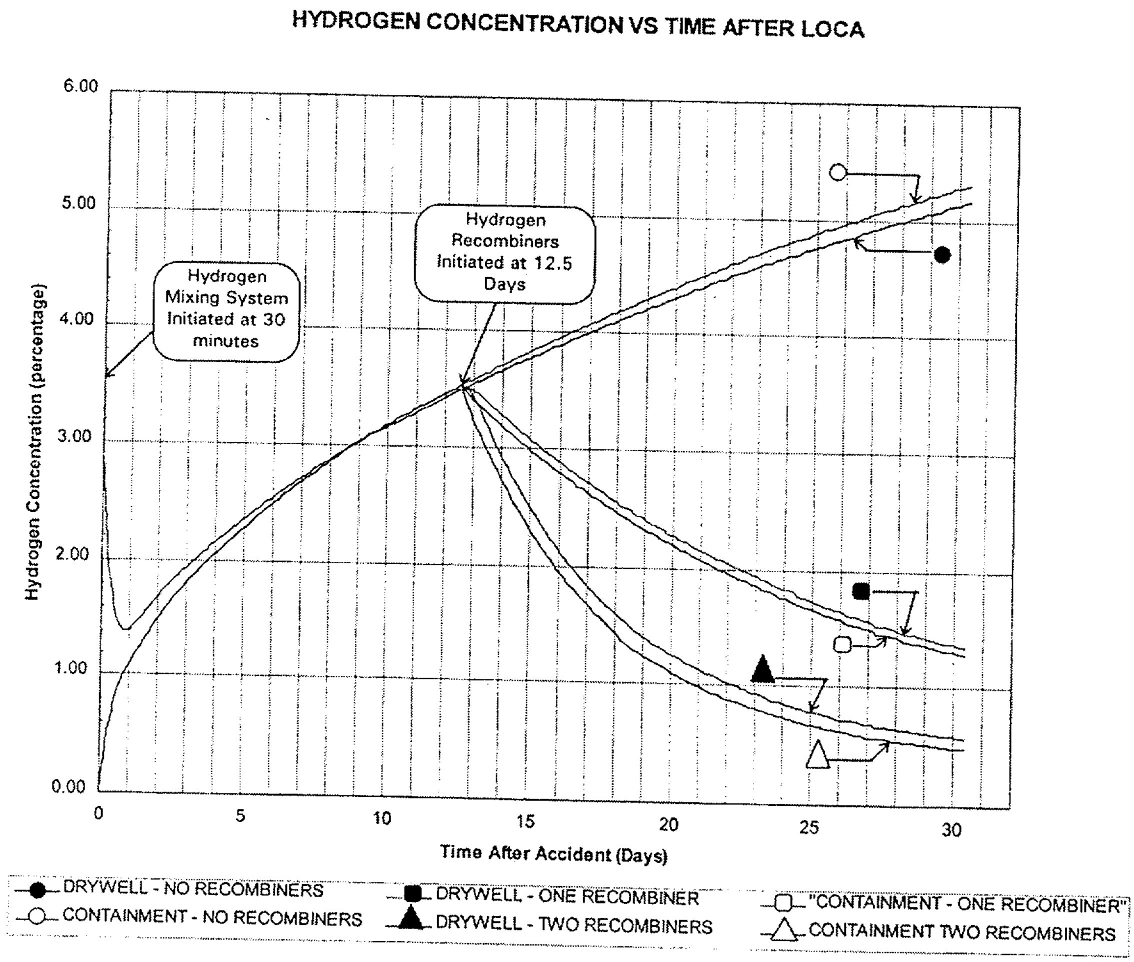


Figure H-8 **IBA - 0.1 ft² Steam Break, Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)**

Question 6: In Section 4.7 for Post-LOCA Combustible Gas Control, it is indicated that post-LOCA production of hydrogen and oxygen from radiolysis increases proportionally with the power level. Please provide the hydrogen and oxygen generation rate Volume % vs. time curves after a LOCA and indicate the effect on the start-up time of the recombiner to keep the hydrogen level below the flammability limit at uprate power compared to current power level.

Response 6: *The radiolytic source term change, which is directly proportional to thermal power, was accounted for in the calculation. The hydrogen generation rate is dominated by Metal Water Reaction at the beginning of the accident (i.e. short term hydrogen control). The contribution from radiolysis becomes significant a few days after LOCA, when recombiners are needed to mitigate the H2 concentration in the containment. The predicted start time of the recombiners after uprate is on the order of 12 1/2 days, a shorter time comparing to the pre-uprate recombiners start time of 14 days. The following is the hydrogen vol. vs. time curve:*



Questions 7-9 are from fax received 4/26/00

Question 7: Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to compete the required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

Response 7:

Operations Simulator Observations

Initial simulator observations comparing simulator response at current 100% power to the simulator response at 105% (Uprated Power) indicate no appreciable time frame or parameter differences. The observations were based on simulator response with no operator actions. Confirmation of this information is expected by August 11, 2000.

Operator Reliability Discussion

The following is a discussion on the impact of power uprate on operator reliability. It is from a probabilistic risk assessment (PRA) perspective in which the operator actions are examined with a Human Reliability Analysis (HRA). From the HRA a human error probability (HEP) was determined. The HRA process was done using the methodology provided in NUREG/CR–1278, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, and NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis Procedure.

An HRA was performed for at-power and shutdown conditions. For at-power conditions, three HRA categories were performed. These were pre-accident, post accident, and ATWS. For shutdown an HRA was performed for two categories, pre-accident and post accident. Power uprate has potential impact on post accident and ATWS HRA.

***At-power, post accident** - When the HRA was performed, the methodology in NUREG-1278 and 4772 was used. The methodology in these NUREGs employ a screening process and a nominal process. The screening process is more conservative in the estimate of human error probability (HEP). In other words, if the screening process is used then the results will give a higher HEP than if the nominal method is used. The screening process was used for nearly all the at-power post accident HRAs. Therefore, a conservatism is built into the HEPs which off sets the potential impact of power uprate. Also, for the HRAs that were done using the nominal process, the inputs to the HRA due to power uprate are small and should not impact the final HEP.*

***At-power, ATWS** – The HRA assumes that operators recognize an ATWS. They are trained extensively on ATWS conditions in the simulator. Therefore, in HEP determination it is assumed that there are no errors in the operators to diagnose plant conditions. Unlike post accident HRA for at-power and shutdown, the ATWS HRA assumes that diagnosis is properly made and that only the HEP for operator action is determined. Nearly all of the HEPs considered are errors of omission. It is assumed that once the ATWS has been recognized, the level of stress is "moderately high", representing a "heavy task load." However, if the transient continues beyond the attempted injection of boron because SLC does not work, certain tasks are*

assessed as being performed under "extremely high stress" which represents "threat stress." With power uprate conditions these stress levels are not expected to change since operators will be trained to diagnose and take action the same as they were under non-uprate conditions.

To illustrate, how little the impact that power uprate is expected to have on CDF, the River Bend PRA model assumes that the HEP for failure to inject SLC is 1.0E-3 which produces a CDF of 1.066E-11. If the failure to inject SLC is increased 10 fold to 1.0E-2 to represent power uprate the CDF increases to 1.066E-10. In either case, cutsets containing the term for failure to inject SLC are truncated during quantification. A truncation limit of 1E-9 is used. It should be noted that this example is for a single cutset and does not represent all the cutsets for an ATWS. The initiator for this example is loss of power conversion, with HPCS failure, and reactor protection system failure as well as failure to inject with SLC.

***Shutdown. post accident** – Power uprate does not impact the HEPs for shutdown. Due to how mission times are determined based on plant states (decay heat and water level), the HRA for pre-power uprate contains ample conservatism in its assumptions that a 5% increase in power will not change the HEPs for power uprate.*

Question 8: Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g. normal range, marginal range, and out-or-tolerance range)? If changes will occur, discuss how they will be addressed.

***Response 8:** Changes to control room displays are identified and implemented during the modification process. There are several parameters that are expected to change during power uprate that will have an impact on control room indicator colorbanding. These parameters include reactor pressure, main steam flow, feedwater flow, RCIC turbine speed, standby liquid control system storage tank level, moisture separator-reheater pressure and main steam pressure. The zone markings for the indicators affected by these changes will be adjusted to accommodate the uprate conditions.*

Power uprate is not expected to affect control room panel layouts or annunciator window legends. Alarm setpoints will be adjusted to accommodate uprate conditions. Setpoints that will increase as a result of power uprate include the reactor vessel high pressure SCRAM setpoint, the high pressure ATWS RPT setpoint and the main steam high flow isolation setpoint. The feedwater pump suction header low pressure setpoint is expected to decrease as a result of power uprate.

Eight instrument loops have been identified as requiring range changes as a result of power uprate. These include four main steam flow, one turbine load set, one turbine load and two main steam line pressure loops.

The changes in instrumentation in the main control room will be prepared in accordance with the plant modification process, which incorporates detailed review of the proposed control room design change package. Operations Training and Procedures Groups will be among those who will review the modification packages. This will ensure timely update of procedures and training programs to include uprate related changes in scheduled operator training before implementation. All identified changes to control room alarms, controls, and displays will be implemented prior to operation at uprated conditions.

Question 9: We understand that the power uprate will be implemented during the current fuel cycle and also the plant is now operating with fuel leaks. The fuel leaks may be controlled in the future through flux suppression by insertion of control rods. Confirm that the licensee has analyzed the effects of control rods on fuel performance at uprated power conditions. Describe in detail plant operation at uprated power conditions with fuel leaks. Describe any effect on existing fuel leaks or any discernable impact on overall fuel integrity.

Response 9: *River Bend is anticipating implementing Power Uprate while on line during its current fuel cycle number 10. River Bend currently has no identified fuel leaks. One leaking fuel bundle was removed from the core during the recently completed Refueling Outage Number 9. The fuel leaks identified during Cycle 8 were removed from the core during Refueling Outage Number 8. A higher than normal iron and copper corrosion product deposition on the fuels rods in combination with an early in cycle plant chemistry transient was reasoned to be a contributor to the identified failures during cycle 8. Strict Water Chemistry controls were implemented prior to startup from Refuel Outage Number 8 to reduce the iron and copper corrosion products in the feedwater and condensate systems and any subsequent deposition on the fuel rods. The single fuel failure removed from the core during Refueling Outage Number 9 was not attributed to the corrosion product deposition phenomenon identified during Refueling Outage Number 8. Fuel inspections were conducted during RF09 to confirm that the condition that caused the cycle 8 fuel failures no longer exist.*

Licensed margins for fuel performance will be maintained during uprated power operations. Operation at uprated power will have no adverse impacts on overall fuel integrity. The cycle 10 core was designed to include the planned mid-cycle uprate while maintaining thermal limit requirements. Future cycles will maintain thermal requirements for uprated operation through the normal design process including increases in the reload batch size as appropriate. Management of fuel performance will continue to be governed by the COLR prepared for uprated power and any fuel degradation identified in the future will continued to be managed by the station's existing Fuel Integrity Monitoring Program and Failed Fuel Action Programs. Any fuel leaks identified in the future may be suppressed by the insertion of control rods while still maintaining licensed fuel limits.

The SAR for the 105% uprate at River Bend Station submitted with LAR 99-15 concluded the RWCU system will continue to reduce the concentration of radioactive and corrosive species in the reactor coolant. The change in iron input to the reactor is expected to increase very slightly as a result of the increased feedwater flow. However, the change is considered insignificant and will not impact RWCU performance and overall reactor water chemistry. Improved Feedwater and Condensate monitoring and performance along with the conclusions regarding RWCU performance result in no discernable impact on overall fuel integrity during operation at uprated power.

Questions 10-12 are from a conference call with 3 questions on piping

Response 10: *The table below was revised to* ***include the correct*** *uprate factor for piping not located in break exclusion area for Main Steam Piping*

MAXIMUM STRESS TABLE FOR THE MAIN STEAM (MSS) PIPING SYSTEM Piping Material = SA106 Gr B Carbon Steel							
Attribute	Node No.	Maximum Levels				Max. Uprate Ration = Calc/Allow	Acceptability
		Existing Value	Uprate Factor	Calculated Uprate Value	Allowable Value		
Sm (psi)					17828		
Eqn. 9 (psi)	142	10678	1.1500	12280	26742	0.459	Acceptable
Eqn. 9E (psi)	142	10713	1.1500	12320	40113	0.307	Acceptable
Eqn. 9F (psi)	142	11277	1.1500	12969	53484	0.242	Acceptable
Functional Capability (psi)	142	11277	1.1500	12969	26742	0.485	Acceptable
PIPING LOCATED IN BREAK EXCLUSION AREA							
Eqn. 10 (psi)	125	69751	1.0495	73204	42787	1.711	See Eqns. 12 & 13 *
Eqn. 12 (psi)	125	4506	1.0021	4515	42787	0.106	Acceptable
Eqn. 13 (psi)	125	33099	1.0247	33917	42787	0.793	Acceptable
CUF	125	0.0940	**	0.0831	0.1000	0.831	Acceptable
PIPING NOT LOCATED IN BREAK EXCLUSION AREA							
Eqn. 10 (psi)	142	43607	**1.0423**	45452	42787	1.062	See Eqns. 12 & 13 *
Eqn. 12 (psi)	142	3291	**1.0021**	3298	42787	0.077	Acceptable
Eqn. 13 (psi)	142	21453	**1.1500**	24671	42787	0.577	Acceptable
CUF	142	0.0119	**1.0348**	0.0123	0.1000	0.123	Acceptable

LEGEND:
NA – Not Applicable
NC – No Change due to power uprate
* Per NB-3653.6, if Equation 10 cannot be satisfied, the Eqns. 12 & 13 shall be met.
** A detailed evaluation was performed to qualify the equation.

Question 11: Describe the methodology used to qualify Equation 13 for node point 45 for the High Pressure Core Spray system.

Response 11: *In general, the majority of the piping was evaluated by determining an uprate factor and applying it to the existing stresses. The uprate factor was based on the largest percentage change in pressure, temperature or flow (fluid transient loading). The uprate factor increase would then be conservatively applied to the existing stresses.*

For Equation 13 stresses that exceeded the allowable based on the above methodology, a detailed hand calculation was applied. The first step would be to apply the exact uprate factor to each component of the stress instead of applying the largest factor to the entire stress range. In addition, in some cases, the equation 13 stress would be recalculated based on seismic moments using N411 damping when the seismic moments were available.

In the specific case of equation 13 stress at node point 45 for high pressure core spray, the stresses were recalculated based reduction of the C3' indice. The node point is a tapered transition weld between pipe and a valve end. The original calculation used a severe thickness change to determine the C3' indice. The revised calculation for uprate used a precise thickness, which reduced the indice. The stresses were recalculated based on the new indice. Thus, the stresses shown for uprate are actually lower than the original stresses.

Question 12: Describe the methodology used to increase the usage factors for the Reactor Core Isolation Cooling piping (ICS).

Response 12: *The Cumulative Usage Factor (CUF) is dependent on Sn (Equation 10 results) in two respects. The CUF is based on Sp (peak stress, Equation 11) and Ke (for elastic-plastic behavior). Sp is essentially a linear function of Sn by virtue of the K1, K2, & K3 stress indices. Since the relative effect of* $\alpha\Delta T2$ *peak local thermal stress term is generally quite small, the uprate factor for Sp is essentially the same as that for Sn. This is slightly conservative, since the unchanging* $\alpha\Delta T2$ *term will tend to damp the effect of operating pressure and temperature increases.*

Ke remains constant at 1.0 for values of Sn up to 3Sm, then varies with increasing Sn up to 3mSm, after which it remains constant at the value (1/n).

The initial evaluation performed for each ASME Class 1 region determines the uprate factor for CUF using the 5^{th} *power of the Sn factor similar to the methodology of NC/ND-3611.2. For cases where the results to Equation 10 (Sn) stress are less than 3Sm, Ke remains constant and the CUFs uprated by using the Section NC/ND-3611.2 methodology remain conservative. For cases where increases in Sn result in Equation 10 slightly exceeding 3Sm, any slight increase to the Ke values are bounded by conservatism that exist in the analysis and the actual exponent of the fatigue curve such that no further adjustments are necessary to the CUF evaluation.*

For all locations where majority of the CUF is determined from pairs with Sn greater than 3mSm, the Ke parameter is fixed at its upper bound value, and therefore, the CUFs uprated by using the Section NC/ND-3611.2 methodology remain conservative. This method can also be applied to pairs slightly under 3mSm in a manner analogous to pairs with Sn a little under 3Sm in the above discussion.

The cases where the effects of Ke could be significant occur when Sn for most pairs fall between 3Sm and 3mSm. In this region the Ke parameter increases faster than Sp. However, the number of cycles from actual fatigue lookup curves decay at a slower rate than the 5th power methodology estimates. In addition to using the 5th power approach, a second adjustment is made to account for the increasing Ke.

In summary, the CUF for the Reactor Core Isolation Cooling piping (ICS) were determined by applying an uprate factor to the existing CUF. The uprate factor was applied to the 5th power, which is similar to the methodology of NC/ND-3611.2. In addition a second factor is applied to account for the effect of variation in the Ke parameter for Sn between 3Sm and 3mSm.

Question 13 involves the reactor vessel surveillance capsule removal deferral request issues

Background for Question 13

Entergy Operations, Inc. (EOI), submitted a request (Reference 1) to defer withdrawal of the first River Bend Station (RBS) reactor vessel surveillance capsule from 10.4 effective full power years (EFPY) to 13.4 EFPY (approximately three cycles). The staff reviewed the licensee's request, and had several concerns. The issues raised by the staff were documented in the summary of the NRC/EOI public meeting held on February 10, 2000.

EOI subsequently requested that the proposed license amendment to revise the reactor vessel surveillance capsule removal schedule be withdrawn (Reference 3). In addition, this letter mentioned that EOI was considering the possibility of requesting an extension to the test results submittal time frame as allowed by 10 CFR Part 50, Appendix H. The NRC response to this letter (Reference 4) raised a concern that this extension may impact the ongoing review of the power uprate application.

Question 13: The staff noted that resolution of issues regarding the fluence determination will be necessary, as it affects the establishment of reactor pressure vessel pressure-temperature limits.

Response 13: *After additional discussion, RBS committed (per Reference 5) to use the power-uprate 32 EFPY PT curves until we have submitted test results for the first RBS surveillance capsule specimen and received NRC approval of revised PT limit curves. We are revising our submittal so that the proposed Technical Specifications contain only the 32 EFPY PT curves. It is our understanding that the Staff's power-uprate fluence concerns were resolved by this RBS commitment.*

Question 13 References:

1. *RBS letter RBG-45151, dated 10/25/1999, License amendment request (LAR) for revision to the reactor vessel material surveillance program capsule withdrawal schedule*
2. *RBS letter RBG-45225, dated 1/12/2000, Additional information related to LAR for revision to the reactor vessel material surveillance program capsule withdrawal schedule*
3. *RBS letter RBG-45277, dated 3/2/2000, Withdrawal of LAR for revision to the reactor vessel material surveillance program capsule withdrawal schedule*
4. *NRC letter RBC-49186, dated 3/23/2000, NRC response to withdrawal of application for license amendment to revise capsule withdrawal schedule, plus staff recommendation for consideration of impact on power uprate application and encouragement of further discussion on subject*
5. *RBS letter RBG-45343, dated 5/8/2000, Additional information related to power uprate application fluence determination and commitment to use 32 effective full power years (EFPY) pressure-temperature (PT) curves contained in the power uprate submittal until the NRC has approved revised pressure-temperature curves.*

Questions 14-17 are from a conference call on April 17, 2000

Question 14: Clarification of conclusion statement is requested for the Generator output steady-state electrical analysis on page 19 of the RAI.

Response 14: *The steady-state electrical analysis to power uprate conditions was performed at 1,130 MW. The steady-state analysis reveals that the up-grade to 1,130 MW has little impact on Fancy Point 230 kV bus voltage. The 1,130 MW is a bounding condition for the analysis as it exceeds the actual uprated main generator electrical output of 1,043.1 MW at a power factor of 0.91.*

Question 15: Will the modifications discussed in the response to question 7 of the RAI response, dated April 3, 2000, be completed before the power uprate?

Response 15: *No MOV modifications were necessary for the flow only portion of power uprate. The modifications to MOV's necessary to support the pressure increase portion of power uprate will be completed prior to implementation of this phase. The modifications to the program MOV's will include both uprate conditions and LTU 98-01 recommendations in accordance with EOI programs.*

Question 16: Do the revised MOV calculations include consideration of Limitorque Technical Update 98-01?

Response 16: *MOV calculations will be revised to include the LTU 98-01 recommendations. Operability of the affected valves has been verified in accordance with plant procedures and the recommendations of LTU 98-01 in accordance with EOI programs.*

Question 17: Why are no modifications planned for B21-MOVF085?

Response 17: *MSL drain valve, B21-MOVF085 will have a small margin after power uprate implementation however, a modification is* <u>*not*</u> *required because of the following:*

1. *F085 is identical to F019 and 16 (i.e., they have the same actuator/motor and valve limits). The limiting thrust limit is the valve's weak link of 14,500.*

2. *Per RF9 test report, the running load measured was 477 lbs. The design basis calculation assumes 1,100 lbs. The running load assumed for F019 and F016 is approximately 750 lbs. The design basis calculation could be revised to decrease the running load for F085 to 750 lbs. which would provide a design margin of 11% without any field adjustments (torque switch change). Any margin greater than 10% is consider a medium margin (as oppose to low or high margin)*

3. *The current torque switch setting for F085 is 2.0. This MOV has the capability of increasing the thrust output to > 12,000 lbs. without any modification. The C14 thrust output for F019 is 12,261 lbs. (which provides a margin of ~ 20%)*

The following tables have the C14 values <u>***revised***</u> *to the post RF-8 and 9 values:*

Valve Number	B21-MOVF067A		B21-MOVF067B		B21-MOVF067C		B21-MOVF067D	
System	MSS		MSS		MSS		MSS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0
MEDP (C)	1,178.0	1,244.0	1,178.0	1244.0	1,178.0	1,244.0	1,178.0	1,244.0
DP Load (O)	-1,527.8	-1,614.4	-1,527.8	-1,614.4	-1,527.8	-1,614.4	-1,527.8	-1,614.4
DP Load (C)	1,593.3	1,682.6	1,593.3	1,682.6	1,593.3	1,682.6	1,593.3	1,682.6
Pmax (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0
Pmax (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0
Piston Effect (O)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Piston Effect (C)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Min. Req'd Thrust (O)	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8
Min. Req'd Thrust (C)	3,215.7	3,305.0	3,215.7	3,305.0	3,215.7	3,305.0	3,215.7	3,305.0
TST Min (O)	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8
TST Min (C)	4,707.8	4,838.5	4,643.5	4,772.4	4,643.5	4,772.4	4,701.4	4,831.9
Thrust Max (O)	12,684.0	N/C	12,726.0	N/C	12,726.0	N/C	12,726.0	N/C
Thrust Max (C)	12,558.0	N/C	12,586.0	N/C	12,586.0	N/C	12,586.0	N/C
Q Min (O)	17.2	16.3	17.2	16.3	17.2	16.3	17.2	16.3
Q Min (C)	34.1	35.0	34.1	35.0	34.1	35.0	34.1	35.0
Q Max (O)	48.0	N/C	79.9	N/C	79.9	N/C	48.0	N/C
Q Max (C)	48.0	N/C	79.9	N/C	79.9	N/C	48.0	N/C
C14 Thrust	5,778.0	N/A	***7,019***	N/A	***9,452***	N/A	***8,903***	N/A
Margin								
Open Thrust	11,061.6	11,148.2	11,103.6	11,190.2	11,103.6	11,190.2	11,103.6	11,190.2
Open Torque	30.8	31.7	62.7	63.6	62.7	63.6	30.8	31.7
Close Thrust	7,850.2	7,719.5	7,942.5	7,813.6	7,942.5	7,813.6	7,884.6	7,754.1
Close Torque	13.9	13.0	45.8	44.9	45.8	44.9	13.9	13.0
Setup Margin								
TST min < C14 Thrust?	-	OK	-	OK	-	OK	-	OK

Valve Number	B21-MOVF016		B21-MOVF019		B21-MOVF085		E51-MOVF013	
System	MSS		MSS		MSS		ICS	
	Current (Rev 3)	Uprate (Rev 4)	Current (Rev 3)	Uprate (Rev 4)	Current (Rev 3)	Uprate (Rev 4)	Current	Uprate
MEDP (O)	1,178.0	1,244.0	1,178.0	1,244.0	1,178.0	1,244.0	1,306.6	1,336.6
MEDP (C)	1,178.0	1,244.0	1,178.0	1,244.0	1,178.0	1,244.0	0.0	0.0
DP Load (O)	3,953	4,174	3,953	4,174	3,953	4,174	23,196.8	23,729.4
DP Load (C)	3,967	4,188	3,967	4,188	3,967	4,188	0.0	0.0
Pmax (O)	700	700	750	750	1100	1100	1,366.6	1,396.6
Pmax (C)	700	700	750	750	1100	1100	1,366.6	1,396.6
Piston Effect (O)	-1,158.0	-1,223.6	-1,158.0	-1,223.6	-1,158.0	-1,223.6	-3,287.1	-3,359.3
Piston Effect (C)	1,158.0	1,223.6	1,158.0	1,223.6	1,158.0	1,223.6	3,287.1	3,359.3
Min. Req'd Thrust (O)	3,494	3,651	3,544	3,701	3,895	4,050	22,019.8	22,480.2
Min. Req'd Thrust (C)	5,825	6,112	5,875	6,162	6,225	6,512	5,397.2	5,469.4
TST Min (O)	3,494	3,651	3,544	3,701	3,895	4,050	22,019.8	22,480.2
TST Min (C)	8,604	9,028	8,366	8,775	8,737	9,140	8,090.4	8,198.6
Thrust Max (O)	8,561.0	N/C	8,561.0	N/C	8,561.0	N/C	27,937.2	N/C
Thrust Max (C)	13,035.5	N/C	13,035.5	N/C	13,035.5	N/C	30,206.4	N/C
Q Min (O)	50.0	52.1	50.0	52.1	51.1	53.2	367.7	375.4
Q Min (C)	77.1	80.8	77.1	80.8	78.1	81.8	90.1	91.3
Q Max (O)	185.0	N/C	155.8	N/C	145.0	N/C	319.8	N/C
Q Max (C)	185.0	N/C	155.8	N/C	145.0	N/C	319.8	N/C
C14 Thrust	11,705.4	N/A	***12,262***	N/A	***9,770***	N/A	15,207.0	N/A
Margin								
Open Thrust	5067	4251	5017	4251	4666	4156	5,917.4	5,457.0
Open Torque	135.0	132.9	105.8	103.7	93.9	91.8	Neg	Neg
Close Thrust	4860	4104	4790	4353	4299	4504	22,116.0	22,007.8
Close Torque	107.9	104.2	78.7	75.0	66.9	63.2	229.7	228.5
Setup Margin								
TST min < C14 Thrust?	-	OK	-	OK	-	OK	-	OK

Valve Number	E51-MOVF019		E51-MOVF022		E51-MOVF045		E51-MOVF059	
System	ICS		ICS		ICS		ICS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	1,363.0	1,393.0	1,351.0	1,381.0	1,165.0	1,231.0	0.0	0.0
MEDP (C)	1,363.0	1,393.0	1,290.0	1,318.1	1,165.0	1,231.0	1,165.0	1,231.0
DP Load (O)	-2,938.6	-3,003.3	-9,228.1	-9,433.0	-7,611.6	-8,042.8	0.0	0.0
DP Load (C)	2,942.7	3,007.5	8,811.4	9,003.3	7,626.6	8,058.7	9,082.0	9,596.5
Pmax (O)	1,363.0	1,393.0	1,374.5	1,404.5	1,165.0	1,231.0	0.0	0.0
Pmax (C)	1,363.0	1,393.0	1,313.5	1,341.6	1,165.0	1,231.0	1,188.5	1,254.5
Piston Effect (O)	0.0	0.0	-34.9	-35.7	0.0	0.0	0.0	0.0
Piston Effect (C)	0.0	0.0	34.9	35.6	0.0	0.0	1,764.8	1,862.8
Min. Req'd Thrust (O)	2,660.0	2,595.3	1,040.0	834.3	1,497.6	1,066.4	1,300.0	1,300.0
Min. Req'd Thrust (C)	5,602.7	5,667.5	9,886.3	10,079.0	9,124.2	9,556.3	12,146.8	12,759.3
TST Min (O)	2,660.0	2,595.3	1,040.0	834.3	1,497.6	1,066.4	1,300.0	1,300.0
TST Min (C)	8,398.4	8,495.5	14,819.6	15,108.4	13,868.8	14,525.5	14,066.0	14,775.3
Thrust Max (O)	12,726.0	N/C	21,816.0	N/C	21,816.0	N/C	21,816.0	N/C
Thrust Max (C)	12,586.0	N/C	21,576.0	N/C	21,576.0	N/C	21,576.0	N/C
Q Min (O)	38.6	37.7	12.0	9.6	15.6	11.1	10.7	10.7
Q Min (C)	81.2	82.1	113.7	115.9	94.9	99.4	99.6	104.6
Q Max (O)	62.0	N/C	213.4	N/C	141.9	N/C	120.7	N/C
Q Max (C)	62.0	N/C	213.4	N/C	141.9	N/C	120.7	N/C
C14 Thrust	10,086.7	N/A	18,626.2	N/A	14,191.0	N/A	14,575.0	N/A
Margin								
Open Thrust	10,066.0	10,130.7	20,776.0	20,981.7	20,318.4	20,749.6	20,516.0	20,516.0
Open Torque	23.4	24.3	201.4	203.8	126.3	130.8	110.0	110.0
Close Thrust	4,187.6	4,090.5	6,756.4	6,467.6	7,707.2	7,050.5	7,510.0	6,800.7
Close Torque	Neg	Neg	99.7	97.5	47.0	42.5	21.1	16.1
Setup Margin								
TST min < C14 Thrust?	-	OK	-	OK	-	No	-	No

Valve Number	E51-MOVF063		E51-MOVF064		E51-MOVF076		G33-MOVF001	
System	ICS		ICS		ICS		WCS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	0.0	0.0	0.0	0.0	1,165.0	1,231.0	0.0	0.0
MEDP (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,176.0	1,242.0
DP Load (O)	0.0	0.0	0.0	0.0	-393.2	-415.5	0.0	0.0
DP Load (C)	32,021.8	33,835.9	32,021.8	33,835.9	393.2	415.5	17,559.9	18,544.8
Pmax (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,176.0	1,242.0
Pmax (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,176.0	1,242.0
Piston Effect (O)	-2,416.1	-2,553.0	-2,416.1	-2,553.0	0.0	0.0	-2,078.2	-2,194.8
Piston Effect (C)	2,416.1	2,553.0	2,416.1	2,553.0	0.0	0.0	2,078.2	2,194.8
Min. Req'd Thrust (O)	3,000.0	2,863.1	3,000.0	2,863.1	741.7	719.4	1,743.3	1,626.7
Min. Req'd Thrust (C)	37,437.9	39,388.9	37,437.9	39,388.9	1,134.9	1,157.2	21,381.4	22,482.8
TST Min (O)	3,000.0	2,863.1	3,000.0	2,863.1	741.7	719.4	1,743.3	1,626.7
TST Min (C)	52,787.4	55,538.3	52,787.4	55,538.3	1,725.0	1,758.9	30,147.8	31,700.8
Thrust Max (O)	62,750.0	N/C	62,750.0	N/C	3,981.4	N/C	35,274.0	N/C
Thrust Max (C)	62,750.0	N/C	62,750.0	N/C	4,611.3	N/C	56,988.0	N/C
Q Min (O)	43.8	41.8	43.8	41.8	5.0	4.8	27.5	25.7
Q Min (C)	546.6	575.1	546.6	575.1	7.7	7.9	337.8	355.2
Q Max (O)	605.5	N/C	491.4	N/C	24.3	N/C	293.4	N/C
Q Max (C)	605.5	N/C	491.4	N/C	24.3	N/C	293.4	N/C
C14 Thrust	8,876.8	N/A	8,633.6	N/A	2,684.9	N/A	13,971.0	N/A
Margin								
Open Thrust	59,750.0	59,886.9	59,750.0	59,886.9	3,239.7	3,262.0	33,530.7	33,647.3
Open Torque	561.7	563.7	447.6	449.6	19.3	19.5	265.9	267.7
Close Thrust	9,962.6	7,211.7	9,962.6	7,211.7	2,886.3	2,852.4	26,840.2	25,287.2
Close Torque	58.9	30.4	Neg	Neg	16.6	16.4	Neg	Neg
Setup Margin								
TST min < C14 Thrust?	-	No	-	No	-	OK	-	No

Valve Number	G33-MOVF004		G33-MOVF039		G33-MOVF040		G33-MOVF053	
System	WCS		WCS		WCS		WCS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	0.0	0.0	0.0	0.0	0.0	0.0	1,175.1	1,241.1
MEDP (C)	1,176.0	1,242.0	0.0	0.0	0.0	0.0	1,175.1	1,241.1
DP Load (O)	0.0	0.0	0.0	0.0	0.0	0.0	9,160.7	9,674.9
DP Load (C)	17,962.4	18,969.8	0.0	0.0	0.0	0.0	9,184.9	9,700.4
Pmax (O)	1,176.0	1,242.0	0.0	0.0	0.0	0.0	1,175.1	1,241.1
Pmax (C)	1,176.0	1,242.0	1,433.8	1,499.8	1,433.8	1499.79	1,175.1	1,241.1
Piston Effect (O)	-2,078.2	-2,194.8	0.0	0.0	0.0	0.0	-1,744.9	-1,842.8
Piston Effect (C)	2,078.2	2,194.8	3,448.7	3,607.4	3,448.7	3,607.4	1,744.9	1,842.8
Min. Req'd Thrust (O)	1,743.0	1,626.4	1,923.9	1,923.9	1,923.9	1,923.9	8,215.8	8,632.0
Min. Req'd Thrust (C)	21,783.6	22,907.6	5,372.6	5,531.3	5,372.6	5,531.3	11,729.8	12,343.3
TST Min (O)	1,743.0	1,626.4	1,923.9	1,923.9	1,923.9	1,923.9	8,215.8	8,632.0
TST Min (C)	30,714.9	32,299.7	7,758.0	7,987.2	7,758.0	7,987.2	16,703.2	17,576.8
Thrust Max (O)	35,274.0	N/C	21,330.6	N/C	21,330.6	N/C	17,554.6	N/C
Thrust Max (C)	56,988.0	N/C	21,095.9	N/C	21,095.9	N/C	21,576.0	N/C
Q Min (O)	27.5	25.7	40.4	40.4	40.4	40.4	123.2	129.4
Q Min (C)	344.2	362.0	112.8	116.1	112.8	116.1	175.9	185.1
Q Max (O)	286.6	N/C	314.0	N/C	445.9	N/C	230.0	N/C
Q Max (C)	286.6	N/C	314.0	N/C	445.9	N/C	230.0	N/C
C14 Thrust	8,191.0	N/A	***17,514***	N/A	***15,385***	N/A	17,233.0	N/A
Margin								
Open Thrust	33,531.0	33,647.6	19,406.7	19,406.7	19,406.7	19,406.7	9,338.8	8,922.6
Open Torque	259.1	260.9	273.6	273.6	405.5	405.5	106.8	100.6
Close Thrust	26,273.1	24,688.3	13,337.9	13,108.7	13,337.9	13,108.7	4,872.8	3,999.2
Close Torque	Neg	Neg	201.2	197.9	333.1	329.8	54.1	44.9
Setup Margin								
TST min < C14 Thrust?	-	No	-	OK	-	OK	-	No

Valve Number	G33-MOVF054		B21-MOVFO86	
System	WCS		MSS	
	Current	Uprate	Current	Uprate
MEDP (O)	1,175.1	1,241.1	0.0	0.0
MEDP (C)	1,175.1	1,241.1	0.0	0.0
DP Load (O)	9,160.7	9,674.9	0.0	0.0
DP Load (C)	9,184.9	9,700.4	0.0	0.0
Pmax (O)	1,175.1	1,241.1	0.0	1055.0
Pmax (C)	1,175.1	1,241.1	0.0	1055.0
Piston Effect (O)	-1,744.9	-1,842.8	0.0	-1048.7
Piston Effect (C)	1,744.9	1,842.8	0.0	1048.7
Min. Req'd Thrust (O)	8,215.8	8,632.0	1005.4	500
Min. Req'd Thrust (C)	11,729.8	12,343.3	1005.5	1548.7
TST Min (O)	8,215.8	8,632.0	1005.4	500
TST Min (C)	16,703.2	17,576.8	1005.4	2256.5
Thrust Max (O)	17,554.6	N/C	11779.0	10801.3
Thrust Max (C)	21,576.0	N/C	11779.0	10683.6
Q Min (O)	123.2	129.4	11.7	5.2
Q Min (C)	175.9	185.1	11.7	16
Q Max (O)	224.4	N/C	62	62
Q Max (C)	224.4	N/C	62	62
C14 Thrust	15,862.0	N/A	3587.0	7044.5
Margin				
Open Thrust	9,338.8	8,922.6	10773.6	10301.3
Open Torque	101.2	95.0	50.3	56.8
Close Thrust	4,872.8	3,999.2	10773.6	9134.9
Close Torque	48.5	39.3	50.3	46
Setup Margin				
TST min < C14 Thrust?	-	No	OK	OK

Questions 18- on Required Testing are from D. Jaffe 6/21/00

Question 18: NEDC-31897P-A, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Section 5.11.9, "Power Uprate Testing," stipulates that uprate licensing applications will include pre-operational tests for systems or components which have revised performance requirements. The logical place for this information would be section 10.4 of LAR 99-15. Please identify the River Bend Station (RBS) systems and components that will be tested as a result of LAR 99-15.

Response 18: *To implement the flow-only phase of the uprate the power will be ascended while online from the current licensed Core Thermal Power Limit of 2894MWt, no outage is anticipated. Pre-operational tests as described in RG-1.68 will be performed by reducing power, and acquiring data at 90, 95 and 100 % of the old license limit of 2894 MWT. This data for each monitored parameter will be used to predict the values for each Uprate Power Ascension Test Condition beyond the old license limit. Testing will consist of observation of each parameter during power ascension to, and at each Test Condition, and a determination that the response of the parameter in the MWT domain is as anticipated and acceptable, based on the above-described "pre-operational tests."*

When the pressure increase phase is implemented, the testing will be continued, using the same methodology, until full power is realized. The test procedure for pressure increase will take data on the same parameters at the maximum obtainable flow-only power level, as pressure is increased in approximately 5 psi increments. Pressure control system dynamic tuning will not be required, since increased pressure results in additional control valve (control system) margin at a given power level. Dynamic testing of the Feedwater / Reactor level control system is not envisioned for pressure increase. Testing may be required to ensure that scram avoidance is maintained on loss of one feed pump coincident with a level 4 (recirc runback test). Lessons learned during the flow-only implementation will be incorporated as appropriate.

Of the systems which will be tested during Up-rate Power Ascension, only the Reactor Feedwater / Reactor Level Control System and Reactor Pressure Control System, including subsystems Pressure Regulator, Turbine By-Pass, and Turbine Control Valves, are deemed to have substantive changes in performance requirements. Therefore, these systems will be dynamically tested, as a part of the Power Ascension Test Procedure, to ensure that they will perform adequately at the new higher flow rates and power levels.

Testing will also be performed on other systems and components at River Bend Station, as the result of Power Uprate. The data will be gathered in accordance with the generic guideline, which requires the licensee to ensure that "a careful, monitored approach to increased power is achieved." The basis for selection of parameters to be monitored is inclusion of parameters which are indicative of plant stability and reliability.

Question 19: For RBS systems and components with revised performance requirements, please identify those that will not be tested in accordance with the original RBS startup testing methods and acceptance criteria. Your response should include justification for any deviations from the original pre-operational test program and/or from the applicable guidance in Regulatory Guide 1.68, "Initial Test Program for Water-Cooled Reactor Power plants."

Response 19: *Preoperational Testing, Initial Fuel Loading and Precritical Testing, Initial Criticality Testing and Low Power Testing, as described in Regulatory Guide 1.68, were completed during the initial start-up of River Bend. The only guidance in Regulatory Guide 1.68, which is applicable to the River Bend 5% Power Uprate, is that portion of the Power Ascension Testing which must be performed at approximately 100% licensed core thermal power. For the purposes of uprate testing, we have defined substantive changes in performance requirements as changes for which existing plant operating experience and the results of original start-up testing could not be used as the basis for an evaluation that could ensure adequate margin in the original design at uprated conditions. Based on NEDC-31897P-A, "General Electric Generic Guidelines for Boiling Water Reactor Power Uprate", and plant specific evaluations which were performed for River Bend Station, only the following River Bend systems experience substantive changes in performance requirements as the result of the flow-only phase of the power uprate implementation:*

Reactor Feedwater / Reactor Level Control System
Reactor Pressure Control System

These systems will be subjected to dynamic testing, in accordance with the methodology employed during the original start-up testing. The original acceptance criteria will be used, except where it has been superceded by new criteria, as the result of evaluations which were performed while dispositioning test exceptions during original start-up testing. When the pressure increase phase is implemented, the testing will be continued, using the similar methodology, until full power is realized. Once full power is achieved, reactor pressure will be raised from the old operating point of approximately 1040 psia to the new operating point of approximately 1070 psia, in small increments. At each pressure increment, data will be gathered for a list of possibly affected parameters, and the data will be evaluated at each increment, prior to raising reactor pressure further. It is anticipated that the list of monitored parameters for this phase will be substantially the same as for the flow-only phase. Lessons learned during the flow-only implementation will be incorporated in pressure increase phase testing methodology.

There are no systems, for which changes to performance requirements were evaluated to be substantive, which will not be tested in accordance with original RBS start-up test methods and acceptance criteria.

Question 20: Emergency Operating Procedure (EOP) curves and limits may be included in the Safety Parameter Display System (SPDS). Power uprate changes to these curves and limits will require an update to SPDS.

Response 20: *RBS has programmatic controls to ensure that any changes required to SPDS as a result of power uprate are incorporated. In accordance with River Bend Nuclear Procedure RBNP-027, Revision 8, the Licensing Department is required to transmit the approved License Amendment Request (LAR) to the responsible departments for implementing the change. Each responsible department manager reviews the LAR and ensures the requirements for implementation for the manager's department have been met. This would include all required EOP revisions. The EOP revision process is controlled by Operations Section Procedure OSP-0009. In accordance with OSP-0009, Revision 13, when changes are made to Technical Specifications, they will be evaluated for impact on the Emergency Operating and Severe Accident Procedures. In addition, OSP-0009, Revision 13, requires a copy of the revised EOPs/SAPs to be forwarded to the Supervisor Computer Systems to ensure the maintenance of the ERIS and DRMS (SPDS) data base.*