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SUBJECT: Submits response to NRC 980603 RAI re Browns Ferry Nuclear Plant 971001 proposed TS change for power uprate operation.

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TVA

Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609

June 17, 1998

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-260
Tennessee Valley Authority	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE NO. TS-384 - POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

This letter provides additional information requested by NRC in support of TS-384. On October 1, 1997, TVA provided TS-384, an amendment to Operating Licenses DPR-52 and DPR-68 that will allow Units 2 and 3 to operate at an uprated power level of 3458 MWt.

The enclosure provides TVA's response to the June 3, 1998, NRC RAI for the October 1, 1997, proposed TS change. This letter includes replies to each of the NRC requests.

There are no new commitments made in this letter. If you have any questions, please telephone me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

200109

cc: See Page 3

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REFERENCES

1. TVA letter to NRC dated October 1, 1997, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request For License Amendment for Power Uprate Operation

2. TVA letter to NRC dated March 16, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 Technical Specification (TS) No. 384 Supplement 1 - Request for License Amendment for Power Uprate Operation

3. NRC letter to TVA dated June 3, 1998, Browns Ferry Nuclear Plant, Units 2 and 3: Request for Additional Information Relating to Technical Specification Change No. TS-384 - Power Uprate Operation (TAC Nos. M99711 and M99712)

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ENCLOSURE
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 2 AND 3

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE NO. TS-384 - POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

This enclosure provides the TVA response to the June 3, 1998, NRC request for additional information.

NRC REQUEST 1

In Section 4.1.1.1, Containment long term pool temperature response, it is indicated that a pre-uprate SHEX code benchmark evaluation was performed using a heat exchanger K-factor of 223 BTU/sec-°F, 92° F Residual heat removal (RHR) service water (SW) temperature, 6500 gpm RHR flow rate, and 4000 gpm RHR SW flow rate. The analysis resulted in a peak suppression pool temperature of 1750° F [sic]. These values are not consistent with the existing BFN analysis which used a heat exchanger K-factor of 228 BTU/sec-°F, 95° F RHR service water temperature, 6500 gpm RHR flow rate, and 4500 gpm RHR SW flow rate and reported a peak pool temperature of 1770° F [sic]. The existing analysis was performed using the May-Witt decay heat curve while the uprate was performed using the ANS/ANSI 5.1 (2σ uncertainty). The comparative analysis should use the same input assumptions. Please justify this inconsistency discrepancy. Also, please provide the containment temperature and pressure profiles.

TVA REPLY 1

A plant-specific SHEX benchmark case using inputs consistent with the Updated Final Safety Analysis Report (UFSAR) basis (i.e., heat exchanger K-factor of 228 BTU/sec-°F, RHR service water temperature of 95° F, 6500 gpm RHR flow rate, 4500 gpm RHR SW flow rate and May-Witt decay heat curve) was performed for BFN as part of the power uprate containment analyses. The resulting maximum suppression pool temperature is 176.7° F. This result is in agreement with the UFSAR analysis which was prepared using the licensing basis computer code. Thus, it is concluded that the change in computer codes does not affect the results. As requested by this question, the suppression pool temperature and drywell and wetwell vapor space pressure profiles for this benchmark case are attached as Figures 1-1 and 1-2.

Given that the peak suppression pool temperature obtained from the SHEX UFSAR benchmark case is very close to the 177° F limit for the BFN long-term torus integrity program (LTTIP), it is clear that with a direct incorporation of the power uprate input assumptions, the peak suppression pool temperature would exceed this containment design criteria. To resolve this issue, revised containment input parameters were imposed such that the net effect of power uprate will result in a peak suppression pool temperature of 177° F:

- 92° F RHR service water temperature
- 223 Btu/sec-°F
- 4000 gpm RHR service water flow rate
- 6500 gpm RHR flow rate.

The 92° F RHR service water temperature is a BFN technical specification change item (included in the October 1, 1997 submittal as Surveillance Requirement 3.7.1.2 and associated Figure 3.7.1-1). Operating in accordance with proposed Figure 3.7.1-1 will allow the plant to remain within its LTTIP limits (177° F) up to the current Ultimate Heat Sink limit of 95° F. The RHR service water flow rate is conservatively assumed at a lower value of 4000 gpm to more accurately reflect system performance with two RHRSW pumps supplying one loop of RHR heat exchangers. The lower RHR heat exchanger k-factor is the result of the change to the RHR service water temperature and the RHR service water flow rate. Additionally, the ANSI/ANS 5.1-1979 plus 2σ uncertainty decay heat curve was utilized as discussed in the letter from Ashok Thadani (NRC) to Gary Sozzi (GE), "Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993. The results of this revised set of input parameters showed a peak suppression pool temperature of 175° F and 177° F for the pre-uprate and the power uprate condition, respectively. The UFSAR will be revised to reflect these conditions upon Power Uprate implementation.

Please see the TVA Reply to NRC Request 2 for the uprated containment temperature response profiles.

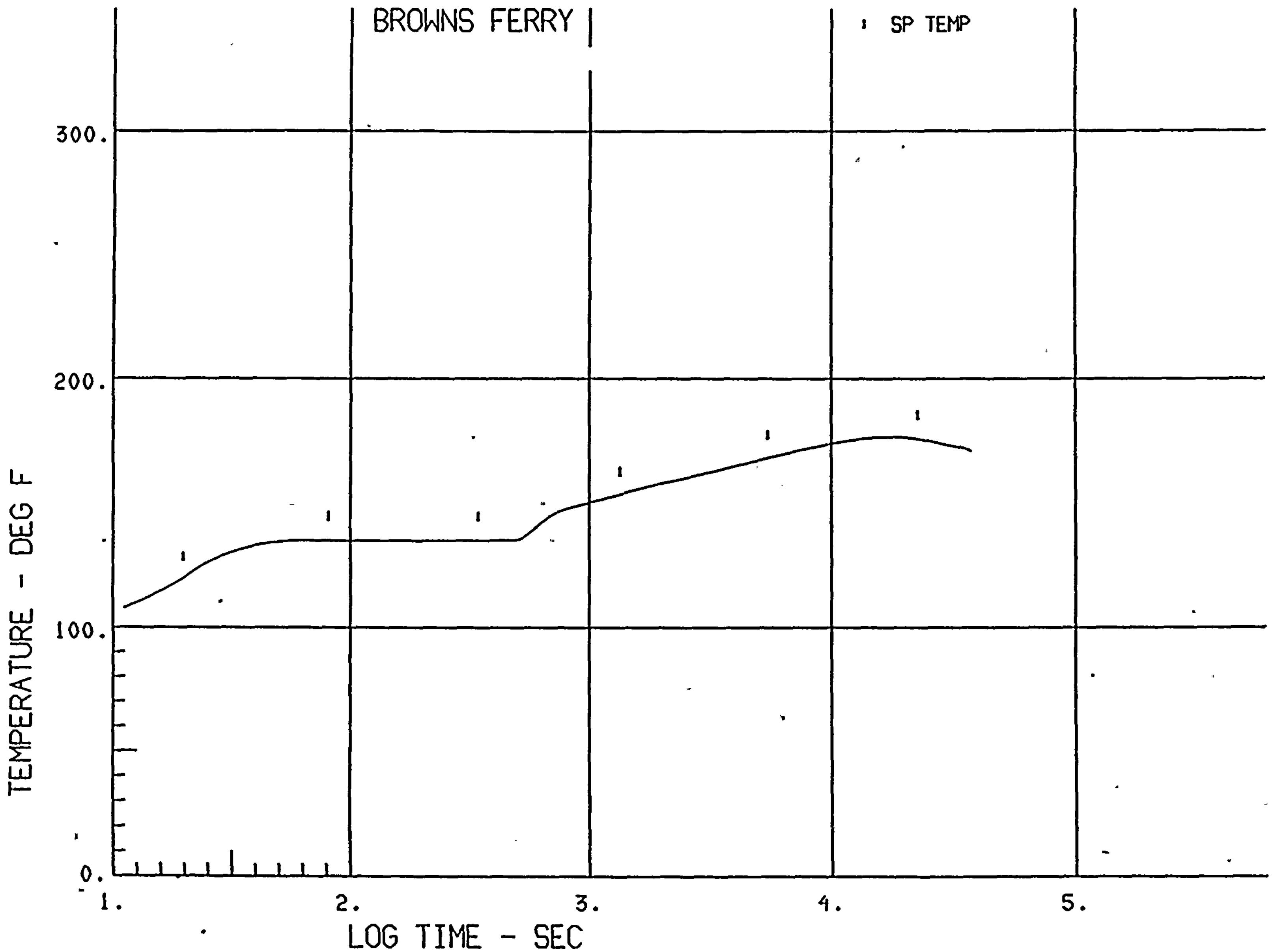

Figure 1-1: FSAR Benchmark Case - Suppression Pool Temperature Response at Pre-Uprate Power

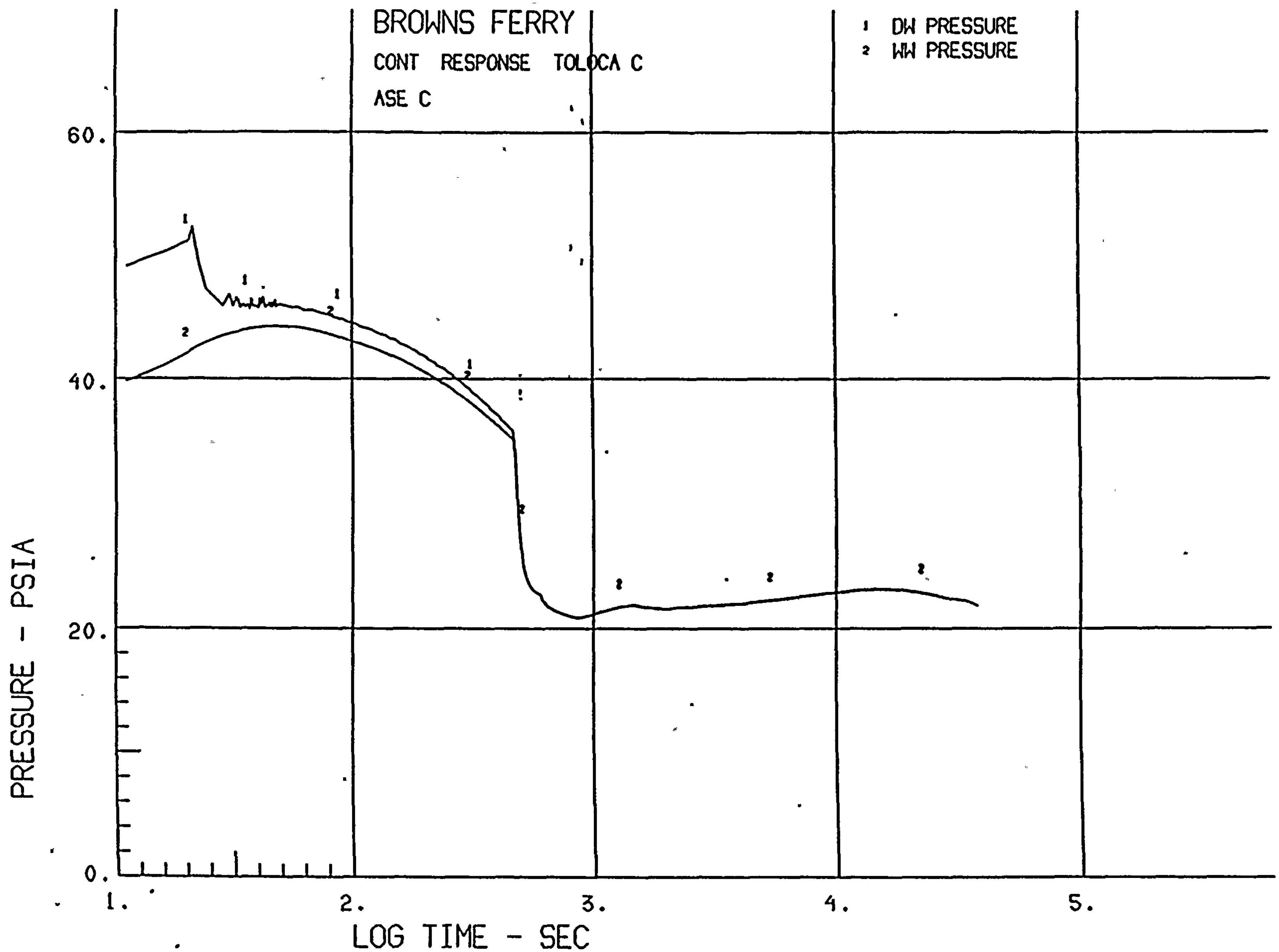

Figure 1-2: FSAR Benchmark Case - Drywell and Wetwell Pressure Response at Pre-Uprate Power

NRC REQUEST 2

Section 4.1.1.2, Containment Gas Temperature Response indicates that at uprated power, the calculated peak drywell airspace temperature exceeds the drywell shell design temperature, but only at the beginning of the accident and for a short period of time. This is not considered a threat to the drywell shell structure due to relative long drywell shell heat-up time. Please provide the containment temperature profiles after a loss-of-coolant-accident.

TVA REPLY 2

The containment drywell and wetwell vapor space temperature profiles for Power Uprate conditions are provided in Figures 2-1 and 2-2.

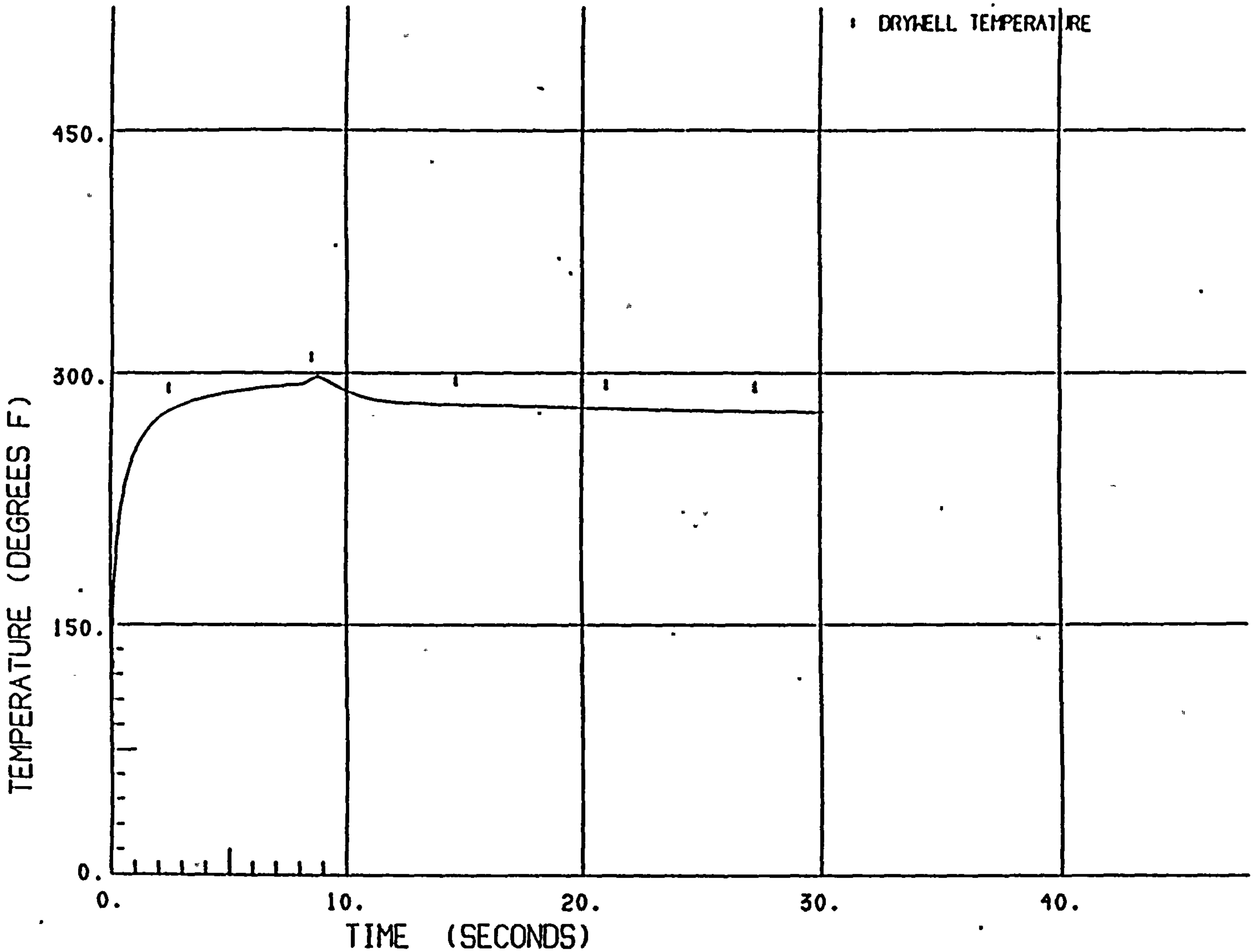

Figure 2-1: DBA-LOCA Containment Temperature Response at Power Uprate Condition

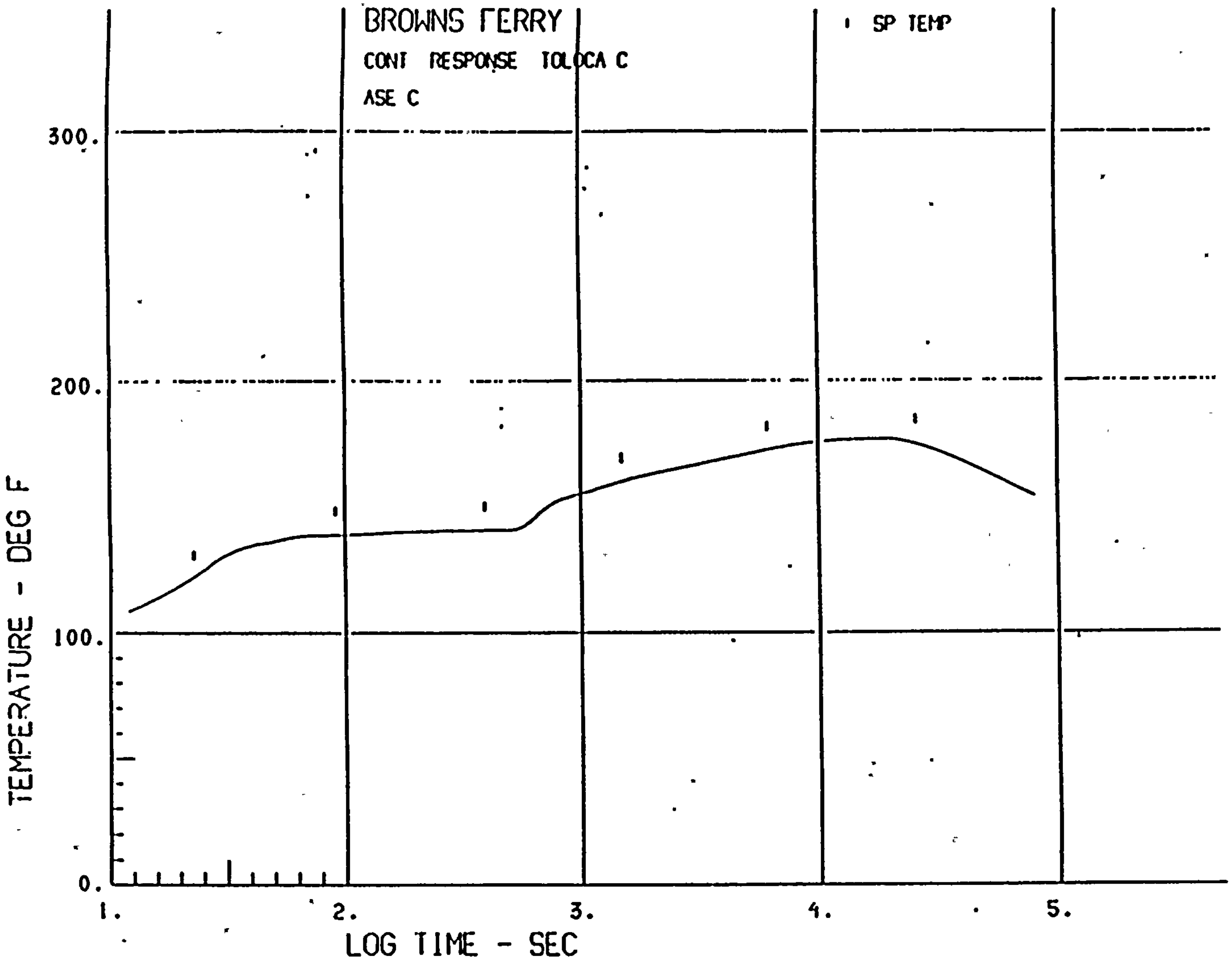

Figure 2-2: DBA-LOCA Suppression Pool Temperature Response at Power Uprate Condition

NRC REQUEST 3

Section 4.1.2.3, Subcompartment Pressurization, indicates that the sacrificial shield wall design remains adequate because the peak pressure in the annulus region increases slightly due to the power uprate. Please provide details of the results to justify your conclusion.

TVA REPLY 3

The structural evaluation in BFN UFSAR Section 12.2.2.6 demonstrates that the shield wall structure can withstand 19 psi pressurization, which is a differential pressure across the shield wall from the annulus to the drywell space. The largest line which has the safe-end located in the annulus is the 4-inch jet pump instrument line nozzle. For all larger lines, the double-ended line break results in the flow being directed into the drywell and not into the annulus. Effects of a postulated loss of coolant accident (LOCA) occurring within the sacrificial shield area have been investigated utilizing uprated conditions.

Method of Evaluation

It is assumed that the design basis accident occurs and that the fluid is saturated through the break. For conservatism, Appendix K conditions (Moody's slip flow model) are also assumed. The following power/flow conditions will be used:

	Dome Pressure[1] (psia)	Power (MWt)	Flow (% Flow)
Pre-Uprate	1023	3358	100
Uprated	1053	3527	100
Uprated (MELLLA)	1053	3527	81

[1]Dome pressure increases 3 psi above nominal due to 2% power increase for Appendix K conditions. Nominal conditions: 100% of rated power and 1020 psia dome pressure at pre-uprate conditions, and 100% of rated power and 1050 psia at uprated conditions.

where:

3293 MWt	=	100% of pre-uprate power
3358 MWt	=	102% of pre-uprate power
3458 MWt	=	100% of uprated power
3527 Mwt	=	102% of uprated power
102.5 Mlb/hr	=	100% of core flow (pre-uprate & uprate)

The methods used are SAFER/GESTR-LOCA and engineering calculation.

The SAFER/GESTR-LOCA model is a thermal-hydraulic transient code developed by GE and approved by the NRC for long term inventory analysis of BWR LOCAs as well as other off-normal reactor transients where the neutron kinetics are of no consequence. The SAFER/GESTR-LOCA model simulates all the BWR major vessel regions: lower plenum, guide tubes, core, core bypass, upper plenum, the initially subcooled region outside the core shroud, the saturated region outside the core shroud and the steam dome. In addition, flow, inventory, and heat transfer calculations are performed for a high power fuel assembly. The program solves the mass and energy balances for each region and the momentum equation for the flow loops. A single pressure is used for the thermodynamic property calculations. The SAFER/GESTR-LOCA model can simulate all BWR water makeup systems.

The pressure differential across the shield wall, ΔP, can be characterized based on the velocity in the annulus:

$$\Delta P = k\left(\rho V^2\right)$$

where:
- k is the loss coefficient in the annulus
- ρ is the flow density
- V is the local velocity

For conservatism, the break flow velocity V_b is equal to the local velocity V and the break flow density ρ_b is equal to the flow density ρ.

In terms of the free stream velocity, V_∞,

$$\Delta P = k\rho_b V_\infty^2\left(\frac{V_b}{V_\infty}\right)^2$$

By the continuity equation,

$$V_\infty = \frac{m_a}{\rho_a A_a}$$

$$V_b = \frac{m_b}{\rho_b A_B}$$

where:

m_a is the annulus flow rate
A_a is the flow area between the annulus and drywell
ρ_a is the steam density in the annulus
m_b is the break flow rate
A_b is the break area
ρ_b is the break flow density

Combining the equations above for ΔP

$$\Delta P = k\rho_b V_\infty^2\left(\frac{V_b}{V_\infty}\right)^2 = k\rho_b\left(\frac{m_a}{\rho_a A_a}\right)^2\left(\frac{m_b}{\rho_b A_b}\right)^2\left(\frac{\rho_a A_a}{m_a}\right)^2 = k\frac{1}{\rho_b}\left(\frac{m_b}{A_b}\right)^2 = k\frac{1}{\rho_b}G^2$$

where:

G is the break flow mass flux
k is a constant.

The SAFER/GESTR-LOCA outputs were used to obtain the break mass flux and break enthalpy. The previous equation was used to ratio the pressure differential for the different power/flow conditions,

$$\Delta P_2 = \Delta P_1\left(\frac{G_2}{G_1}\right)^2\left(\frac{\rho_1}{\rho_2}\right)$$

For pre-uprate conditions, the pressure differential is less than 2 psi, as specified in BFN UFSAR Section 12.2.2.6. The uprated analyses are based upon an assumed pre-uprated value of 2 psi.

<u>Summary of Results</u>

The comparison of the annulus pressurization loads at pre-uprate and uprate conditions are summarized as follows:

Power/Flow (%Power/%Flow)	Annulus Pressurization (psi)	Increase Above Pre-uprate (Δpsi)
Pre-uprate (102/100)	2	
Uprate (102/100)	2.09	0.09
Uprated (102/81)	2.41	0.41

The results show that the annulus pressurization load on the shield wall is increased by approximately 0.4 psi due to the power uprate at the MELLLA point. The MELLLA point gives the largest annulus pressurization. The results demonstrate that the annulus pressurization loads at uprated conditions are still well below the limit of the BFN design basis value of 19 psi.

NRC REQUEST 4

Section 4.1.4, Combustible Gas Control In Containment, predicts an earlier startup of the containment atmospheric dilution system at uprated power due to the increase in hydrogen and oxygen generation rates and that the predicted startup time will not result in system operation outside established design or operational restraints. Please provide the time responses.

TVA REPLY 4

Containment Atmospheric Dilution (CAD) system operation is performed to maintain containment oxygen levels below 5% volume following a LOCA. The response of the containment and the CAD system following a LOCA is expected to be consistent with that shown in UFSAR Figures 5.2-14 and 15. These original, design basis evaluations indicate that CAD initiation will be required between 1 and 2 days after the LOCA based on the suppression chamber oxygen level.

A 5% power uprate will increase the oxygen generation rate by approximately 5% since oxygen generation is a function of radiolysis which is approximately a direct function of core power.

Based on a 5% increase in oxygen generation rate, CAD initiation will be required one to two hours earlier. This slight reduction in time is not critical to manual operator actions during a LOCA.

REFERENCES:

1. TVA letter to NRC dated October 1, 1997, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request For License Amendment for Power Uprate Operation

2. TVA letter to NRC dated March 16, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 Technical Specification (TS) No. 384 Supplement 1 - Request for License Amendment for Power Uprate Operation

3. TVA letter to NRC dated March 20, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request for License Amendment for Power Uprate Operation

4. NRC letter to TVA dated June 3, 1998, Browns Ferry Nuclear Plant, Units 2 and 3: Request for Additional Information

Relating to Technical Specification Change No. TS-384 - Power Uprate Operation (TAC Nos. M99711 and M99712)