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

February 2, 2006

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket No. 50-259
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1 - RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES - SUBMITTAL OF BROWNS FERRY NUCLEAR PLANT UNIT 1 SEISMIC AND INTERNAL FIRES IPEEE REPORTS (TAC NO. MC5729)

By letter dated October 26, 2005 (Reference 1), the NRC requested additional information concerning TVA's responses to NRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination Of External Events (IPEEE) For Severe Accident Vulnerabilities - 10CFR 50.54(f)."

TVA's responses are contained in the Enclosures to this letter. Enclosure 1 contains the specific NRC questions and TVA responses; Enclosures 2 through 5 provide specific documents requested by the NRC in Seismic Questions 1 and 8.

A011

There are no new regulatory commitments associated with this submittal. If you have any questions about this submittal, please contact me at (256) 729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 2nd day of February, 2005.

Sincerely,

William D. Crouch

William D. Crouch
Manager of Licensing
and Industry Affairs

Reference:

1. NRC Letter from M. H. Chernoff to K. W. Singer (TVA), "Browns Ferry Nuclear Plant, Unit 1 - Request for Additional Information Regarding Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Browns Ferry Nuclear Plant Unit 1 Seismic and Internal Fires IPEEE Reports (TAC No. MC5729)" dated October 26, 2005.

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS

The following provides TVA's response to the NRC's October 26, 2005 request for additional information regarding TVA's responses to NRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination Of External Events (IPEEE) For Severe Accident Vulnerabilities."

IPEEE FIRE REQUESTS FOR INFORMATION

NRC Fire Question 1:

The submittal provides very little discussion regarding the treatment of hot short cable failures and spurious operation circuit faults. The potential impact of fire-induced spurious actuations on the ability to achieve post-fire safe shutdown is recognized as an important fire risk issue. Describe how hot short cable failures and spurious actuation circuit faults were treated in the original analysis. Please provide such an analysis and discuss the impact of any resulting fire risk scenarios on the conclusions regarding fire vulnerabilities and potential plant improvements.

It is expected that the Individual Plant Examination for External Events (IPEEE) fire analysis will, as a minimum, include treatment of those hot short and spurious actuation circuit configurations identified as the "Bin 1" items in Regulatory Issue Summary 2004-03.

TVA Reply to Fire Question 1:

Hot shorts and spurious actuations have not specifically been considered in the analyses. Fires were assumed to occur at specific locations or compartments resulting in either an engulfing fire (initial screening) causing damage to all power and control cables and components in the area or the fire damage is confined within a zone of influence (detailed screening) based on the fire size. Damage to cables translates to incapacitation of the associated equipment (i.e., core injection pumps, balance of plant systems, etc.). The plant PSA model is modified accordingly and the core melt frequency is calculated based on specific initiating events likely to have occurred due to the fire. The methodology is focused on the effects of fires on control and power cables and determines the impact of their failure to operate (i.e., functional failure) but not their spurious operation. This evaluation approach is consistent with the FIVE methodology, and consistent with the approach used in the BFN Units 2 and 3 IPEEE fire analyses, reviewed and accepted by the NRC in its June 22, 2000 Staff Evaluation Report for BFN Units 2 and 3 (Reference 1).

However, spurious actuations as a result of fire damage are considered in fire scenarios if appropriate. For example, cable damage within 480V RMOV Boards (Fire areas 4, 5, 6, 7, etc.) results in closure of MSIVs; plant transients such as "Turbine Trip" have been conservatively assumed for a majority of the fire areas; recovery from fire related damage was specifically not considered for those scenarios that could impact the operability of a electrical boards for which the plant risk model allows recovery (i.e., top events SDREC, RFRHW, macros RCOK, RQOK, etc.). These top events (macros) were set to disallow recovery of the affected board.

While the potential exists for other than functional failures, in most cases these forms of failures would not be detrimental to safe plant shutdown. Following are some examples:

- ADS Accumulators (MSRV air supply): The accumulators are mechanical devices located inside the inert drywell. They would not experience spurious actuations during a fire.
- ADS/MSRV: Spurious operation of one MSRV can be mitigated by a minimum set of safe shutdown equipment, e.g., one RHR pump, one RHRSW pump, two EECW pumps, etc. The minimum set is available from the diverse set of equipment in the plant.
- MSIV: The MSIVs are fail safe design and will thus close when the circuits are subjected to the effects of fire.
- HPCI: Spurious operation of HPCI is mitigated by operator action in the control room if the high water trip does not occur automatically. Evaluations demonstrate that adequate time is available for the associated operator actions.
- RCIC/CRD: Spurious operation of these systems is not an issue because of their relatively low flow rate to the reactor vessel. Plant procedures allow and evaluations demonstrate that adequate time is available for operator actions to prevent water intrusion into the main steam lines.

It should be noted that only a fraction of the fires that occur will be in the appropriate location, and only a fraction of these fires will have the appropriate severity to cause damage. Consider a reactor building fire source (240V Lighting Transformer 1A) which results in the highest Conditional Core

Damage Probability (CCDP). The probabilistic formula used for this is as follows:

Δ**CDF** = $F_f * P_E * P_{SA} * P_{AS} * P_{DM} * \Delta P_{CCD}$ (per reactor-year)

(Reference NEI 00-01)

Table 1.1 ΔCDF for Reactor Building Fire with 240V Lighting Transformer 1A Fire Source		
F_f	=	fire frequency; frequency of fires of any size anywhere within the fire area
	=	8.77E-04
P_E	=	fire size parameter; fraction of fires in the area capable of reaching damaging combinations of time and temperature
	=	0.075 (Table 6-1c, IPEEE-Fire Submittal, based on EPRI FEDB)
P_{SA}	=	probability of spurious actuations of a component combination given cable damage
	=	0.08 (Based on M/C Thermoplastic, Inter-cable interaction; Probability of cable damage = 0.4 and probability of spurious actuation = 0.2 given cable damage, Reference EPRI Report 1006961)
P_{AS}	=	probability that automatic suppression will fail to control the fire before damage to the cable(s) is such that spurious actuation could occur
	=	0.1 (Reference Fire protection SDP Appendix F, based on a difference of time to damage to time to suppress of 9 minutes)
P_{DM}	=	probability that detection and manual suppression will fail to control the fire before damage to the cable(s) is such that spurious actuation could occur
	=	1 (Neglect)
P_{CCD}	=	conditional probability of core damage given fire-induced failures including spurious actuations of a component combination
	=	6.58E-04 (Table 6-2.1 (c) (13), IPEEE-Fire Submittal based on loss of HPCI, RCIC, CAD, Loop II drywell Spray, and 480V SD Board 1A, 2A, Turbine Trip initiator)
ΔCDF	=	3.46E-10

The change in core damage frequency is well below 1E-07; therefore, the fire scenario is of low risk significance. If it is assumed that spurious actuations will result in additional loss of mitigating capability and the CCDP is increased by two orders of magnitude, the resulting risk still remains low. Note from the following table that if the CCDP increases by approximately one order of magnitude, the ranking moves up by 25 and an increase of 2 orders of magnitude will bring the 50th ranked sequence to the top position. Therefore, this is a very conservative assumption.

Sequence Rank	Initiator	Frequency	CDF	CCDP
Top Ranked Sequence	LCV (loss of condenser vacuum)	9.70E-02	5.09E-08	5.25E-07
25th Ranked Sequence	LCV (loss of condenser vacuum)	9.70E-02	6.12E-09	6.31E-08
50th Ranked Sequence	TT (Turbine trip with bypass)	5.50E-01	4.23E-09	7.70E-09

Also note that spurious actuations of HPCI, RCIC, CAD, etc. are of no consequence or can be mitigated as explained above.

NRC Fire Question 2:

Related to the preceding question, no discussion is provided on the possibility of experiencing a loss-of-coolant accident (LOCA) from a fire event. Such an event may occur from spurious actuation of the automatic depressurization system or other high/low pressure interfaces. Please discuss your analysis of LOCAs caused by the postulated fire scenarios, the frequency of such events and core damage frequency (CDF) associated with them. If such an analysis was not performed, assess the impact of those scenarios on the fire area and compartment CDFs.

TVA Reply to Fire Question 2:

Automatic depressurization at BFN is provided by main steam safety/relief valves (MSRV). The current fire safe shutdown analysis assumes that no more than one MSRV will spuriously open and remain open for any given fire analysis area. TVA is currently in the process of identifying analysis areas where more than one MSRV could potentially spuriously operate due to fire damage to associated cables. Appropriate mitigation strategies will be implemented if necessary. This information can then be used to assess the detailed risk impact.

However, LOCA as a result of fire scenario can be evaluated similar to the response to Question 1. A fire scenario resulting in multiple stuck open relief valves can be considered as Other Large Break LOCA (Initiator LLO). The CCDP for LLO is (= CDF/IE = 8.56E-10/8.39E-07) 1.02E-03. Therefore, by considering a challenging fire (similar to the example in Question # 1) and a CCDP of 1.02E-03, the resulting change in CDF is calculated as follows:

FIRE DAMAGE SCENARIOS	FIRE FREQUENCY (ff)	FIRE SIZE PARAMETER (PE) (Severity Factor)	PROBABILITY OF SPURIOUS ACTUATION (PSA)	PROBABILITY OF NON-SUPPRESSION	CCDP	ΔCDF = Ff * PE * PSA * PAS * PDM * Δ PCCD (per reactor-year)
240V Lighting Transformer 1A	8.77E-04	0.075	0.08	0.10	1.02E-03	5.37E-10

Therefore, this scenario will be of low risk significance. It can be seen from the above example that for a fire scenario of a higher fire frequency, a higher CCDP will still remain low risk.

High-Low pressure interface valves are typically two valves in series. Review of RIS 2004-03, Rev. 1 shows that under Bin 1 (B), for any two thermoplastic cables, concurrent failure is postulated to occur due to inter-cable shorting. We are currently in the process of identifying analysis areas where such occurrences are possible. However, as evaluated above, a LOCA as a result of such a scenario is likely to be of low significance.

Residual Heat Removal System (RHR) high-low pressure valves have specifically been included in Bin 1 in RIS 2004-03, Rev. 1. At BFN, there are two normally closed valves in series (for shutdown cooling suction to the RHR pumps). Motive power is removed and the breaker for the outboard valve is tagged out during normal power operation. In addition, the power cable for this valve is routed in dedicated conduits to prevent hot shorts. Therefore, spurious operation is highly unlikely in this scenario. If a break were to occur in this line, it will be similar to Break Outside Containment (BOC). Only the Shutdown Cooling Mode of RHR is affected by this break. For BOC initiator the CCDP (= 3.12E-08/6.67E-4) is 4.68E-05. This scenario is therefore, bounded by LOCA events.

Spurious operation of High-low pressure interfaces may also result in containment bypass. Per EPRI Fire PRA guidance (EPRI TR-105928), all areas/components with fire induced CDF above 1E-07 were evaluated for containment bypass potential. The IPEEE fire submittal, Sections 5.3 and 6.4 describes the method. Since each of the identified areas/components has fire induced LERF that is below the cut-off of 1E-07, it can be concluded that these fires do not result in, or cause, containment breach concerns beyond those already addressed in the plant risk model.

NRC Fire Question 3:

No discussion is provided regarding the impact of fire on human error probabilities. Generally there are human error related basic events integrated into the probabilistic risk assessment (PRA) model for conditional core damage frequency (CCDP) quantification. These may derive either from the internal events analysis, but should also include fire-specific manual actions as specified in the plant's post-fire safe shutdown procedures. Describe how these manual actions were treated in the post-fire plant safe shutdown response model and in the Human Reliability Analysis portions of the IPEEE fire analysis. Note that all credited human actions should be supported by an assessment of the corresponding human error probability (HEP) including consideration of available staffing, scenario timing, and any conditions associated with the fire, which might contribute to an increase in the HEP value (e.g., smoke, blocked access routes, general confusion, etc.). If the human error probabilities were not adjusted to account for fire impact, either revisit and revise those probabilities according to the conditions posed by the fire scenario and recalculate the CDFs for the affected areas and compartments, or provide a basis for assuming that fire will not impact each of the credited human actions.

TVA Reply to Fire Question 3:

The basic premise of the BFN FIVE PRA model is that all equipment modeled in PRA remains available unless damaged by fire. Therefore, all operator actions described in the model can be performed unless affected by the fire. The majority of the operator actions are within the control bay. Actions outside the control bay generally require equipment recovery. These actions are failed in the fire model if affected by the fire and are not adjusted for human error probabilities (HEP).

Fire specific manual actions as described in the Safe Shutdown Instructions (SSI) are based on a limited set of equipment (Appendix R Equipment). Whereas, the IPEEE fire analysis relies on all PRA equipment unless shown to be damaged by fire. This is consistent with the Units 2 and 3 approach. Therefore, Safe

Shutdown Instruction (SSI) described manual actions are not applicable to the IPEEE fire analysis.

NRC Fire Question 4:

Accepted practice for fire PRAs includes the use of a range of heat release rates (HRRs) representing an uncertainty distribution for this parameter. In the Browns Ferry Nuclear Plant (BFN) Unit 1 IPEEE, a single value has been used. For control cabinets, 480Vac Motor Operated Valve (MOV) Boards, 4kV Boards and other electrical panels, a 190 Btu/sec (200kW) peak HRR has been used. This value coincides with the 95th percentile peak HRRs recommended in the Fire Protection Significance Determination Process (SDP) (Inspection Manual Chapter (IMC) 0609, Appendix F) for motor control centers (MCCs) and Switchers. Selection of 95th percentile for peak HRRs is a conservative approach, minimizing the need to evaluate other HRRs from the distribution. However, the submittal has missed two important points of Appendix F:

a) For control panels, the 95th percentile peak HRR is 650kW.

b) For MCCs and Switchers, the analysis should include the possibility of high-energy faults.

Please either provide the basis for not considering 650kW peak HRRs for control panels and not addressing high energy faults in the MOV and 4kV boards or reassess the fire propagation analysis results and re-quantify unscreened fire scenarios using an approach that explicitly treats peak HRRs consistently and includes the impact of high-energy faults. Assess the impact of any analysis changes on the study conclusions regarding fire vulnerabilities and potential plant improvements.

TVA Reply to Fire Question 4:

BFN used the EPRI FIVE methodology (including the EPRI Fire PRA Implementation Guide TR-105928) for Unit 1 consistent with the approach used for the Unit 2/3 analysis. The heat release rates (HRR) provided in these documents typically depict the mean value and not a range of values representing an uncertainty distribution. The following bounding analysis is however, provided to assess the fire vulnerabilities due to higher HRR (95th percentile) in control panels and energetic faults in MCC and switchgears in accordance with Fire Protection Significance Determination Process (SDP), 0609 Appendix F.

Most of the fire sources analyzed in Unit 1 reactor building are MCC and switchgears. 250V RMOV Board 1C and 480V RMOV Board 1C are chosen to be analyzed for energetic faults. A fire in 250V RMOV Board 1C results in the largest "Damage Height" (See Table C.2-1) due to its location close to a wall (location factor of 2) and has a relatively high CCDP and a fire in 480V RMOV Board 1C results in the largest CCDP.

Table 1 - Fire Source Evaluation

FIRE SOURCES	FIRE FREQUENCY (f_i)	SEVERITY FACTOR (SF_I)	$SF_i * f_i$	REMARKS
250V RMOV Board 1C Energetic faults	5.18E-03	1.00	5.18E-03	For energetic faults, the cabinet fire will continue to burn consistent with the "small electric fire" using the 95th percentile fire intensity (200kW) and a severity factor of 1. A 50th percentile small electric fire will not be considered.
480V RMOV Board 1C Energetic Faults	5.18E-03	1.00	5.18E-03	For energetic faults, the cabinet fire will continue to burn consistent with the "small electric fire" using the 95th percentile fire intensity (200kW) and a severity factor of 1. A 50th percentile small electric fire will not be considered.

Table 2 - Conditional (given fire induced damage) Core Damage Probability (CCDP)						
Fire Sources	Mitigating System Impact	Initiating Event (IE)	Initiating Event Freq.	CDF	CCDP (CDF/IE Freq.)	Remarks
250V RMOV Board 1C Energetic faults	DG-A, RHRSW Pump A3, RCIC SP Valves	Turbine Trip	5.09E-01	1.43E-07	2.81E-07	IE and CDF information from table 6-2.1 (c) (3). Zone of Influence based on 190 Btu/sec.
480V RMOV Board 1C Energetic Faults	RHR Loop I, CS Loop I, RCIC suppress. Pool valves, 250V MOV Bd. 1C, RHRSW Pump A2	Turbine Trip	5.09E-01	3.49E-06	6.86E-06	IE and CDF information from table 6-2.1 (c) (1). Zone of Influence based on 190 Btu/sec.

Table 3 - Given fire induced damage, the likelihood that plant will fail to achieve safe shutdown				
FIRE DAMAGE SCENARIOS (FDS)	SF_i*f_i	* PROBABILITY OF NON SUPP (PNS)	CCDP	$\Delta CDF = \sum f_i * SF_i * PNS_i * CCDP_i$
250V RMOV Board 1C Energetic faults	5.18E-03	1.00E+00	2.81E-07	1.46E-09
480V RMOV Board 1C Energetic Faults	5.18E-03	1.00E+00	6.86E-06	3.55E-08

* Probability of Non-Suppression assumed to be 1.

Most of the control panels located in the reactor building have little or no fire impact (low CCDP). Panel 1-LPNL-925-340 ES Division I was chosen to be evaluated for 95th percentile peak HRR due to its high fire impact or high CCDP. The analysis has been simplified by assuming probability of non-suppression equal to 0.1 for 95th percentile peak HRR fire scenario. No credit is taken for suppression for energetic fires. To account for the larger zone of influence and potentially damaging additional mitigating systems for high HRR fires, the CDF and CCDP were conservatively increased by two orders of magnitude.

a) Control panel 1-LPNL-925-340 ES Division I (95th percentile HRR)

Table 1 - Fire Source Evaluation					
FIRE SOURCES	**FIRE FREQUENCY (f_i)**	**SEVERITY FACTOR (SF_T)**	**SF_i*f_i**	**CHALLENGING FIRES IN AREA f = SUM SF_i*f_i**	**REMARKS**
1-LPNL-925-340 (L) Expected Fire severity	5.18E-03	0.90	4.66E-03		The lower HRR value reflects the expected fire severity or 50th percentile fire, and is associated with 90% of fires.
1-LPNL-925-340 (H) High Confidence Fire severity	5.18E-03	0.10	5.18E-04		The higher HRR value reflects the high confidence fire severity or 95th percentile fire, and is associated with 10% of fires.
				5.18E-03	

Table 2 - Conditional (given fire induced damage)Core Damage Probability (CCDP)						
Fire Sources	Mitigating System Impact	Initiating Event (IE)	Initiating Event Freq.	CDF	CCDP (CDF/IE Freq.)	Remarks
1-LPNL-925-340 (L) Low HRR, Expected Fire severity	CS-I, RHR-I, RCIC, ADS, Recirc. Pump Speed	Turbine Trip	5.09E-01	6.91E-06	1.36E-05	IE and CDF information from table 6-2.1 (c) (8). Zone of Influence based on 190 Btu/sec.
1-LPNL-925-340 (H) High HRR, High Confidence Fire severity	CS-I, RHR-I, RCIC, ADS, Recirc. Pump Speed	Turbine Trip	5.09E-01	6.91E-04	1.36E-03	Assume that a higher HRR of 650 kW results in an increase in CDF by 2 orders of magnitude.

Table 3 - Given fire induced damage, the likelihood that plant will fail to achieve safe shutdown				
FIRE DAMAGE SCENARIOS (FDS)	SF_i*f_i	*PROBABILITY OF NON SUPP (PNS)	CCDP	Δ $CDF=\sum SF_i*f_i*PNS_i*CCDP_i$
1-LPNL-925-340 (L) Expected Fire severity	4.66E-03	1.00E-01	1.36E-05	6.33E-09
1-LPNL-925-340 (H) High Confidence Fire severity	5.18E-04	1.00E-01	1.36E-03	7.03E-08
			SUM =	7.67E-08

* Probability of Non-Suppression assumed to be 0.1

The change in CDF is less than 1E-07. Therefore, the fire source can be screened from further consideration.

b) 250V RMOV Board 1C (Energetic Faults)

Table 1 - Fire Source Evaluation				
FIRE SOURCES	**FIRE FREQUENCY (f_i)**	**SEVERITY FACTOR (SF_I)**	**SF_i*f_i**	**REMARKS**
250V RMOV Board 1C Energetic faults	5.18E-03	1.00	5.18E-03	For energetic faults, the cabinet fire will continue to burn consistent with the "small electric fire" using the 95th percentile fire intensity (200kW) and a severity factor of 1. A 50th percentile small electric fire will not be considered. (Reference SDP 0609, Appendix F, Attachment 5).

Table 2 - Conditional (given fire induced damage) Core Damage Probability (CCDP)						
Fire Sources	**Mitigating System Impact**	**Initiating Event (IE)**	**Initiating Event Freq.**	**CDF**	**CCDP (CDF/IE Freq.)**	**Remarks**
250V RMOV Board 1C Energetic faults	DG-A, RHRSW Pump A3, RCIC SP Valves	Turbine Trip	5.09E-01	1.43E-07	2.81E-07	IE and CDF information from table 6-2.1 (c)(3). Zone of Influence based on 190 Btu/sec.

Table 3 - Given fire induced damage, the likelihood that plant will fail to achieve safe shutdown				
FIRE DAMAGE SCENARIOS (FDS)	**SF_i*f_i**	***PROBABILITY OF NON SUPP (PNS)**	**CCDP**	**Δ $CDF=\sum SF_i*f_i*PNS_i*CCDP_i$**
250V RMOV Board 1C Energetic faults	5.18E-03	1.00E+00	2.81E-07	1.46E-09

* Probability of Non-Suppression assumed to be 1.

The change in CDF is less than 1E-07. Therefore, the fire source can be screened from further consideration.

NRC Fire Question 5:

In Section 3.3.1, page 15, it is noted that penetrations exist in the slab separating compartments 16-1 and 16-2 that may not be sealed. It is also stated that "while these penetrations present a minimal potential for fire propagation to the Cable Spreading Room, the potential for this fire is, conservatively, being considered." No discussions could be found in the balance of the submittal addressing the noted consideration. Describe the basis for concluding that there are no vulnerabilities associated with the penetrations.

TVA Reply to Fire Question 5:

The quantitative evaluation (fire hazard assessment) is provided in Table 6-2.8.1 after the risk evaluation. The evaluation shows that even for small fires (50 kW), the smoke detector activation and alarm provides sufficient time for the site fire brigade to arrive at the fire location before the fire spreads to adjacent cabinet (i.e., before 15 minutes). Due to the timely response of the fire brigade, smoke propagation to the floor above in the cable spreading room will be minimal if any. The evaluation is re-performed using NUREG 1805 spreadsheets. The Method of Milke (similar to the one used in the IPEEE fire submittal) is shown below along with results of other methods.

Smoke Detector Response Time Calculation

References:

- *Milke, J., "Smoke Management for Covered Malls and Atria," Fire Technology, August 1990, p. 223.*
- *NFPA 92B, "Guide for Smoke Management Systems in Mall, Atria, and Large Areas," 2000 Edition, Section A.3.4.*

Heat Release Rate of the Fire (Q) (Steady State)	50.00	kW
Radial Distance to the Detector (r)	15.00	ft
Height of Ceiling above Top of Fuel (H)	12.00	ft
Activation Temperature of the Smoke Detector ($T_{activation}$)	86.00	°F
Smoke Detector Response Time Index (RTI)	5.00	$(m\text{-}sec)^{1/2}$
Ambient Air Temperature (T_a)	78.00	°F
Convective Heat Release Rate Fraction (χ_c)	0.70	
Plume Leg Time Constant (C_{pl}) (Experimentally Determined)	0.67	
Ceiling Jet Lag Time Constant (C_{cj}) (Experimentally Determined)	1.2	

Temperature Rise of Gases Under the Ceiling (ΔT_c)	18.00	°F
for Smoke Detector to Activate **r/H =**	**1.25**	

$t_{activation}$ = detector activation time (sec) = X $H^{4/3}/Q^{1/3}$; where

- X = $4.6\ 10^{-4}\ Y^2 + 2.7\ 10^{-15}\ Y^6$
- H = height of ceiling above top of fuel (ft)
- Q = heat release rate from steady fire (Btu/sec)
- Y = $\Delta T_c\ H^{5/3}\ /\ Q^{2/3}$

Before estimating smoke detector response time, stratification effects can be calculated. NFPA 92B, 2000 Edition, Section A.3.4 provides the following correlation to estimate smoke stratification in a compartment:

H_{max} = the maximum ceiling clearance to which a plume can rise (ft)

= $74\ Q_c^{2/5}\ /\ \Delta T_{f\text{->}c}^{3/5}$; where

- Q_c = convective portion of the heat release rate (Btu/sec)
- $\Delta T_{f\text{->}c}$ = difference in temperature due to fire between the fuel location and ceiling level (°F)

Convective Heat Release Rate Calculation

$Q_c = Q\ \chi_c$; where

- Q_c = convective portion of the heat release rate (Btu/sec)
- Q = heat release rate of the fire (Btu/sec)
- χ_c = convective heat release rate fraction

Q_c = **33.17** Btu/sec

Difference in Temperature Due to Fire Between the Fuel Location and Ceiling Level

$\Delta T_{f\text{->}c}$ = 1300 $Q_c^{2/3}$ / $H^{5/3}$; where

- Q_c = convective portion of the heat release rate (Btu/sec)
- H = ceiling height above the fire source (ft)

$\Delta T_{f\text{->}c}$ = **213.39** °F

Smoke Stratification Effects

H_{max} = 74 $Q_c^{2/5}$ / $\Delta T_{f\text{->}c}^{3/5}$

H_{max} = **12.02** ft

In this case the highest point of smoke rise is estimated to be **12.02** ft. Thus, the smoke would be expected to reach the ceiling mounted smoke detector.

$\mathbf{Y = \Delta T_c\ H^{5/3} / Q^{2/3}}$

Y = **86.45**

$\mathbf{X = 4.6\ 10^{-4}\ Y^2 + 2.7\ 10^{-15}\ Y^6}$

X = **3.44**

Smoke Detector Response Time Calculation

$t_{activation}$ = X $H^{4/3}/Q^{1/3}$

$t_{activation}$ (in sec)=	**26.11**

Calculation Method	Smoke Detector Response Time (sec)
METHOD OF ALPERT	12.09
METHOD OF MOWRER	2.94
METHOD OF MILKE	26.11

Based on an smoke detector alarm in 30 seconds and allowing approximately 10 minutes for the site fire brigade to respond, smoke propagation to the floor above in the cable spreading room is not expected.

NRC Fire Question 6:

In Section 3.3.2, page 17, it is indicated that there is an opening between the Cable Tunnel and the Turbine Building. A discussion is provided about the possibility of fire spreading from the Turbine Building into the Cable Tunnel. The discussion is focused on cable fires and there is no mention of the possibility of turbine oil fire or generator hydrogen fire affecting the Cable Tunnel. Please provide an assessment of the risk contribution of fires involving turbine oil (including splattering of burning oil due to a breach in oil piping) and generator hydrogen fires that might spread into the Cable Tunnel.

TVA Reply to Fire Question 6:

As described in Section 3.3.2, the Cable Tunnel (Compartment 25-1) is located below the 565 floor elevation of the Turbine Building (Compartment 25-3). The cable tunnel extends up into the 565 elevation and terminates approximately 8 feet above the floor (572.5 ft) with a horizontal rectangular opening. The cable tunnel opening is approximately 10′ x 4′ protected with a locked grated steel door. The 8 feet tall concrete shaft is designed as a flood protection wall. The rising hot gasses and products of combustion on 565 feet elevation of the turbine building would have to descend back down through the opening to get inside the cable tunnel. This is not likely as the hot gasses will be rising up. Also, the slight chimney effect of the shaft will be pushing the air out from the cable tunnel and prevent any hot gasses coming in. The flood wall will protect any oil spills getting inside the cable tunnel. Potential for hydrogen explosion is mainly on the turbine floor (exciter cabinet) and will not affect this location.

If a fire did propagate into the cable tunnel, it will only affect Division II of the RHRSW system (Pumps B2 and D2). A fire scenario resulting in turbine trip (TT) and damage to Division II RHRSW pumps was modeled. The CDF was calculated to be 4.5E-05 and CCDP is (=4.5/5.5E-01) 8.20E-05. Assuming a turbine generator (TG) lube oil fire and probability of fire propagation into the cable tunnel as 0.1, the change in risk is calculated as follows:

FIRE SOURCE	*FIRE FREQUENCY (FF)	*PROBABILITY OF FIRE PROPAGATION (PF)	CCDP	Δ CDF=FF*PF*CCDP
TG LUBE OIL FIRE	1.70E-03	1.00E-01	8.20E-05	1.39E-08

* Reference: Fire Protection SDP Appendix F, Attachment 4

As shown above, the risk contribution of this fire scenario is not significant.

NRC Fire Question 7:

In Section 6.2.1, page 72, a discussion is provided about the impact of nonqualified cables in the Reactor Building. The CDF associated with these cables is estimated as 1.5E-07, which is based on a CCDP of 3.37E-05. From the discussions provided it can be inferred that only loss of offsite power is postulated to occur as a result of fires involving the nonqualified cables and it appears that all other systems are assumed to remain unaffected by the fire. Provide a list of system components postulated to fail as a result of the nonqualified cable fire scenarios considered. Identify any other potentially risk-important components that might be impacted by cable failure due to other cables in the same general area as the assumed fire ignition point. Requantify the CDF associated with the nonqualified cable fires if other PRA equipment/component related cables are present within the estimated zone of influence such that potentially risk-important components other than offsite power might be failed.

TVA Reply to Fire Question 7:

The non-qualified cable quantities and the equipment they serve also vary. It is not possible to provide a specific list of components affected in each and every postulated fire scenario throughout the plant. Since the list of affected components is not known due to unspecified number of fire scenarios, a bounding analysis was performed. It was assumed that a worst case fire will result in a loss of offsite power (LOSP). (Please note a minor error in the CCDP value which is 3.42E-05 and not 3.37E-05 based on a CDF of 2.70E-07 and an LOSP initiating event frequency of 7.87E-03. This does not affect the results.)

The following analysis provides additional insights on the impact of loss of mitigating capability. The top five accident sequences in BFN Unit 1 involve three initiators: LCV, IMSIV and LOSP (described below). It was assumed that the fire results in equipment failure and one of these initiators. Computations also assume that the CCDP will increase by 1 order of magnitude if additional mitigating systems are affected.

Fire Scenario	Mitigating System Impact	Initiating Event (IE)	Initiating Event Frequency	CDF	CCDP	Ignition Frequency	Fire Induced CDF	Remarks
Electrical Cable Fire	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator.	Loss of Condenser Vacuum (LCV)	9.70E-02	NA	1.59E-05	4.40E-03	7.00E-08	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LCV.
Electrical Cable Fire	None, other than equipment failure subsumed in the initiator.	Inadvertent MSIV Closure (IMSIV)	5.52E-02	8.69E-08	1.57E-06	4.40E-03	6.93E-09	Assume fire results in inadvertent MSIV closure. Initiator (IMSIV) is involved in the top 3rd and 5th ranked accident sequences

Fire Scenario	Mitigating System Impact	Initiating Event (IE)	Initiating Event Frequency	CDF	CCDP	Ignition Frequency	Fire Induced CDF	Remarks
Electrical Cable Fire	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator.	Inadvertent MSIV Closure (IMSIV)	5.52E-02	NA	1.57E-05	4.40E-03	6.91E-08	Assume that the CCDP is increased by 1 order of magnitude. Initiator is IMSIV.
Electrical Cable Fire	None, other than equipment failure subsumed in the initiator.	Loss of Offsite Power (LOSP)	7.87E-03	2.69E-07	3.42E-05	4.40E-03	1.50E-07	Assume fire results in loss of offsite power. Initiator (LOSP) is involved in the top 4th ranked accident sequence.
Electrical Cable Fire	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator.	Loss of Offsite Power (LOSP)	7.87E-03	NA	3.42E-04	4.40E-03	1.50E-06	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LOSP.

Fire Scenario	Mitigating System Impact	Initiating Event (IE)	Initiating Event Frequency	CDF	CCDP	Ignition Frequency	Fire Induced CDF	Remarks
Electrical Cable Fire (Medium Loading, self Ignited)	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator.	Loss of Offsite Power (LOSP)	7.87E-03	NA	3.42E-04	4.80E-04	1.64E-07	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LOSP. Fire frequency is for Thermoplastic or non-qualified cables per unit per reactor year (Ref. SDP 0609 App F, Att 4).
Electrical Cable Fire (High Loading, self ignited)	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator.	Loss of Offsite Power (LOSP)	7.87E-03	NA	3.42E-04	1.40E-03	4.79E-07	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LOSP. Fire frequency is for Thermoplastic or non-qualified cables per unit per reactor year (Ref. SDP 0609 App F, Att 4).

Fire Scenario	Mitigating System Impact	Initiating Event (IE)	Initiating Event Frequency	CDF	CCDP	Ignition Frequency	Fire Induced CDF	Remarks
Electrical Cable Fire (caused by welding and cutting)	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator.	Loss of Offsite Power (LOSP)	7.87E-03	NA	3.42E-04	1.60E-03	5.47E-07	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LOSP. Fire frequency is for cable fires caused by welding or cutting (Ref. EPRI Fire PRA Table C-3).

The above analysis shows that for all fire scenarios, the fire induced CDF is less than 1E-06 except for the cable fire resulting in LOSP and loss of additional mitigation capability. This case was further evaluated for cable fires caused by self ignition (both medium and high loading cables) and by welding and cutting. It can be seen that the combined CDF due to self ignited high loading cable fire and cable fire caused by welding and cutting is approximately (= 4.79E-07 + 5.47E-07) 1.0E-6. Also note that no weighting factor (WF) was applied as a multiplier to the fire frequency. Thus, self ignited cable fires are not plausible due to proper current limiting provisions (fuses and/or breakers) for all cables.

NRC Fire Question 8:

In Section 6.2.1, page 73, a bounding analysis is presented for CDF contribution of nonqualified cables in the Unit 1 Reactor Building. In that analysis, loss of offsite power is assumed as a bounding event for the postulated fire scenarios. However, no discussion is provided for the basis of the CCDP in terms of assumed failed system trains and components by the fire. Please (a) provide a discussion about what PRA components could potentially be affected by nonqualified cable damage in the Unit 1 Reactor Building; and (b) given that information, provide the basis for the CCDP used in the bounding analysis.

TVA Reply to Fire Question 8:

See response to Question number 7. A review of the top accident sequences was performed to determine the dominant initiators. It was postulated that a cable fire can result in these initiators. Conservative assumptions were made to modify CCDP values as a result of loss of additional mitigation capability. The results show that cable fires will not be a significant contributor to fire induced CDF.

NRC Fire Question 9:

In Section 6.2.8.1, page 75, it is noted that there are no combustibles located in the corridor area of Fire Compartment 16-1. From this statement, it is inferred that transient fires have been either screened out or are assumed to be very unlikely. Furthermore, the corridor is not explicitly addressed in any other parts of the analysis, and therefore no discussion is provided regarding the presence of any cables in this corridor. Identify the cables and components present in the corridor. Provide either the results of the risk quantification or the basis for dismissing the corridor from analysis. If PRA equipment/component related cables are present in the corridor, provide an estimate of the fire-induced CDF associated with corridor fire scenarios that includes transient fires. (The likelihood of transient fires may be quantified using, for example, the Fire Protection SDP procedures (IMC 06.09, Appendix F).)

<u>TVA Reply to Fire Question 9</u>:

As stated on page 75, there are no significant combustibles located in the corridor. The combustible loads in the area include pipe insulation, fiberglass ladders, FP panels, etc. Cables are located in conduits and are not considered combustible loads. The total fire severity is only 6 minutes. A conservative estimate of the fire induced CDF associated with corridor fire scenarios that include transient fires is provided below:

Corridor Ignition Frequency (Reference EPRI/NRC Fire PRA Methodology & SDP Appendix F)						
Components	Generic Fire Frequency (λ_{IS})	Ignition Source Weighting Factor (W_{IS})	Ignition Source Fire Frequency ($\lambda_{IS}*W_{IS}$)	Severity Factor (SF)	Likelihood of a Challenging Fire	Remarks
Transient Fire-Welding	9.70E-03	0.1	9.70E-04			Reference EPRI Fire PRA, Table C-3.
Transients	3.90E-03	0.1	3.90E-04			Reference EPRI Fire PRA, Table C-3.
		Total	1.36E-03	0.1	1.36E-04	SF retained for 95th percentile, i.e., 10% of the fires (Ref. SDP, Appendix F).

Estimate of Fire Induced CDF Associated with Corridor Fire Scenarios								
Fire Scenario	**Mitigating System Impact**	**Initiating Event (IE)**	**Initiating Event Frequency**	**CDF**	**CCDP**	**Ignition Frequency**	**Fire Induced CDF**	**Remarks**
Transient Fire	None, other than equipment failure subsumed in the initiator	Loss of Condenser Vacuum (LCV)	9.70E-02	1.54E-07	1.59E-06	1.36E-04	2.16E-10	Assume fire results in loss of condenser vacuum. Initiator (LCV) is involved in the top 2 ranked accident sequences
Transient Fire	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator	Loss of Condenser Vacuum (LCV)	9.70E-02	NA	1.59E-05	1.36E-04	2.16E-09	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LCV.
Transient Fire	None, other than equipment failure subsumed in the initiator	Inadvertent MSIV Closure (IMSIV)	5.52E-02	8.69E-08	1.57E-06	1.36E-04	2.14E-10	Assume fire results in inadvertent MSIV closure. Initiator IMSIV is involved in the top 3rd and 5th ranked accident sequences
Transient Fire	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator	Inadvertent MSIV Closure (IMSIV)	5.52E-02	NA	1.57E-05	1.36E-04	2.14E-09	Assume that the CCDP is increased by 1 order of magnitude. Initiator is IMSIV

Estimate of Fire Induced CDF Associated with Corridor Fire Scenarios								
Fire Scenario	**Mitigating System Impact**	**Initiating Event (IE)**	**Initiating Event Frequency**	**CDF**	**CCDP**	**Ignition Frequency**	**Fire Induced CDF**	**Remarks**
Transient Fire	None, other than equipment failure subsumed in the initiator	Loss of Offsite Power (LOSP)	7.87E-03	2.69E-07	3.42E-05	1.36E-04	4.65E-09	Assume fire results in loss of offsite power. Initiator LOSP is involved in the top 4th ranked accident sequence.
Transient Fire	Other unspecified equipment loss in addition to equipment failure subsumed in the initiator	Loss of Offsite Power (LOSP)	7.87E-03	NA	3.42E-04	1.36E-04	4.65E-08	Assume that the CCDP is increased by 1 order of magnitude. Initiator is LOSP

Note: No credit taken for suppression capability

The results show that corridor fires associated with transients will not be a significant contributor to fire induced CDF.

NRC Fire Question 10:

In Section 6.2.8.1, page 77, and related analysis of the Auxiliary Instrument Room in Table 6-2.8.1 (pages 126 through 128), it is concluded that the fire risk contribution is insignificant. No description is provided for fire ignition frequencies (i.e., 8.71E-03, and 7.27E-04) and the CCDPs. Please (a) provide the basis for the ignition frequencies, (b) identify what other PRA components could potentially be affected by a fire in the Auxiliary Instrument Room, and discuss how failure of these components was modeled in the estimation of CDF. If the failure of any mitigating equipment not initially assumed to fail in the analysis is possible, please reassess the risk importance of the Auxiliary Instrument Room.

TVA Reply to Fire Question 10:

The fire ignition frequencies for Unit 1 Aux. Instrument Room or Computer room (Fire Compartment 16-1) are calculated in Table 6-1 (c). A severe fire ignition frequency is calculated as 7.27E-04 and a minor fire ignition frequency is calculated as 8.71E-03. The event tree shown on page 77 also shows the ignition frequency for the 3 cases analyzed.

A list of risk significant panels located in the Auxiliary Instrument Room is provided on page 75. A severe fire in Unit 1 Auxiliary Instrument Room or computer room results in loss of all Feedwater (TLFW), MSIV closure (IMSIV) and HPCI failure. This is based on the loss of electrical panels housing these systems. The CCDP is therefore calculated for initiator TLFW which models guaranteed failure of Feedwater and closure of MSIV. In addition, HPCI is also failed in event tree HPGTET. For minor fires, the loss is limited to the electrical panel of origin. Total loss of Feedwater is assumed for this case.

NRC Fire Question 11:

The contribution of transient fires has been dismissed for Fire Areas 4, 5 and 7, the Cable Spreading Room, and Pipe Tunnel based on the assumption that plant control procedures would eliminate the possibility. Please provide a revised CDF estimate for the identified fire areas that includes the impact of transient fuel fires and any other omitted ignition sources (i.e., nonqualified cables and nonqualified junction boxes). (The likelihood of transient fires may be quantified using, for example, the Fire Protection Significance Determination Process procedures (IMC 06.09 Appendix F).

TVA Reply to Fire Question 11:

The following evaluation provides the contribution of the transient and electrical cables to the specified rooms (fire area 4, 5, 7, cable spreading room and pipe tunnel)

Table 1 - Fire Source Evaluation

FIRE SOURCES	FIRE FREQUENCY (f_i)	SEVERITY FACTOR (SF_I)	SF_i*f_i	LIKELIHOOD OF A CHALLENGING FIRE	REMARKS
Transient (low solids and transient combustibles)	5.50E-04	0.10	5.50E-05		Ref. SDP 0609 App. F, Table A4.1. SF retained for 95th percentile, i.e., 10% of the fires. 50th percentile (90%) fires are neglected.
Electrical Cable Fire (caused by cutting and welding)	2.30E-05	0.1	2.30E-06		Self ignited cable fires are not considered plausible. Fire frequency based on low likelihood of welding and cutting fires. SF retained for 95th percentile, i.e., 10% of the fires. 50th percentile (90%) fires are neglected.
			Sum	5.73E-05	

Table 2a - Probability of Non-Suppression Given a Challenging Fire (PNS)

FIRE SOURCES	TIME FOR DETECTION (min)	TIME FOR FIXED SUPPRESSION (min)	TIME TO DAMAGE (min)	PNS FIXED SUPPRESSION (a)	PNS MANUAL SUPPRESSION (b)	REMARKS
Transient	2	N/A	10	N/A	0.3	PNS_{manual} @ (t_{damage} - $t_{detection}$) = 8 for Transient fire

Table 2b - Probability of Non-Suppression Given a Challenging Fire (PNS)				
FIRE SOURCES	**AUTOMATIC SUPPRESSION RELIABILITY**	**MANUAL SUPPRESSION RELIABILITY**	**PROBABILITY OF NON SUPPRESSION $PNS_{scenario}$**	**REMARKS**
Transient	N/A	N/A	0.30	$PNS_{Scenario} = (PNS_{manual})$

Table 3 - Given fire induced damage, the likelihood that plant will fail to achieve safe shutdown					
FIRE DAMAGE SCENARIOS (FDS)	**SF_i*f_i**	**PROBABILITY OF NON SUPP (PNS)**	**CCDP**	**CDF = $SF_i*f_i*PNS*CCDP$**	**REMARKS**
Transient (Fire Area 4)	5.73E-05	0.30	1.27E-07	2.18E-12	Assume transient fire results in damage to 4kV Board B (from Table 6-2.4).
Transient (Fire Area 5)	5.73E-05	0.30	7.53E-06	1.29E-10	Assume transient fire results in damage to 4kV Board A (from Table 6-2.5).
Transient (Fire Area 7)	5.73E-05	0.30	1.05E-04	1.80E-09	Assume transient fire results in damage to 480V Shutdown Board 1B and other components in the room (from Table 6-2.6).
Transient (Cable Spreading Room 16-2)	5.73E-05	0.10	1.73E-06	9.91E-12	Assume transient fire results in damage to single tray and loss of feedwater (from Table 6-2.8.2).
Transient (Pipe Tunnel 25-2)	5.73E-05	0.30	3.65E-07	6.27E-12	Assume transient fire results in turbine trip (from Table 6-2.9.2).

The above analysis shows that the CDF contribution of the transients and electrical cable fires to the overall area CDF is negligible.

NRC Fire Question 12:

On pages 189 (Fire Area 7), 201 (Fire Area 17), and 203 (Fire Area 19), it is noted that there is no combustible loading associated with non-qualified cables. However, in the lower part of these pages, a frequency is estimated for "cable fire-welding" category ignition source. It seems that there is an inconsistency in the method used to estimate the frequencies associated with these fire areas. Clarify the discrepancy between the two ignition source categories for these fire areas. If the overall fire frequency increases, or if a discrepancy in the analysis is noted, provide the CDF associated with the new frequencies.

TVA Reply to Fire Question 12:

Review of BFN combustible load calculation shows that no exposed cables are located in fire areas 7, 17 and 19. Cables are either in conduits or present in negligible amounts. Therefore, "cable fire welding" ignition frequency is not applicable to these areas. The "cable fire welding" ignition frequency is 1.15E-04 which is approximately 2 orders of magnitude less than the overall fire area ignition frequency. Therefore, there will be negligible impact on the CDF of these areas.

IPEEE SEISMIC REQUESTS FOR INFORMATION

NRC Seismic Question 1:

The Submittal does not provide sufficient detail to complete the review. Please provide copies of the following documents that were referenced in the Submittal:

a) Calculation of Basic Parameters for A-46 and IPEEE Seismic Program, Rev. 0 (Reference 9 in the Submittal).

b) Browns Ferry Nuclear Plant Unit 1 USI [Unresolved Safety Issue] A-46 Seismic Evaluation Report, Rev. 0, September 2004 (Reference 15 in the Submittal).

c) Unresolved Safety Issue (USI) A-46/Seismic IPEEE Relay Evaluation Browns Ferry Nuclear Plant Unit 1, Rev. 0, January 2004 (Reference 16 in the Submittal).

d) Seismic-Induced II/I Spray Evaluations at Browns Ferry Nuclear Plant Unit 1, Rev. 0, March 2004 (Reference 19 in the Submittal).

TVA Reply to Seismic Question 1:

a) Enclosure 2 of this submittal contains a copy of TVA Calculation No. CD-Q0000-940339, "Calculation of Basic Parameters for A-46 and Individual Plant Examination of External Events (IPEEE) Seismic Program," Rev. 1, June 14, 1996.

b) The report "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report," Rev. 0, dated September 23, 2004, was previously transmitted to the NRC (Enclosure 2, TVA letter to NRC dated October 7, 2004).

c) The report "Unresolved Safety Issue (USI) A-46/Seismic IPEEE Relay Evaluation Browns Ferry Nuclear Plant Unit 1," Rev. 0, dated January 2004 was previously transmitted to the NRC (Enclosure 3, TVA letter to NRC dated October 7, 2004).

d) Enclosure 3 of this submittal contains a copy of "Seismic-Induced II/I Spray Evaluations at Browns Ferry Nuclear Plant Unit 1," Rev. 0, March 2004.

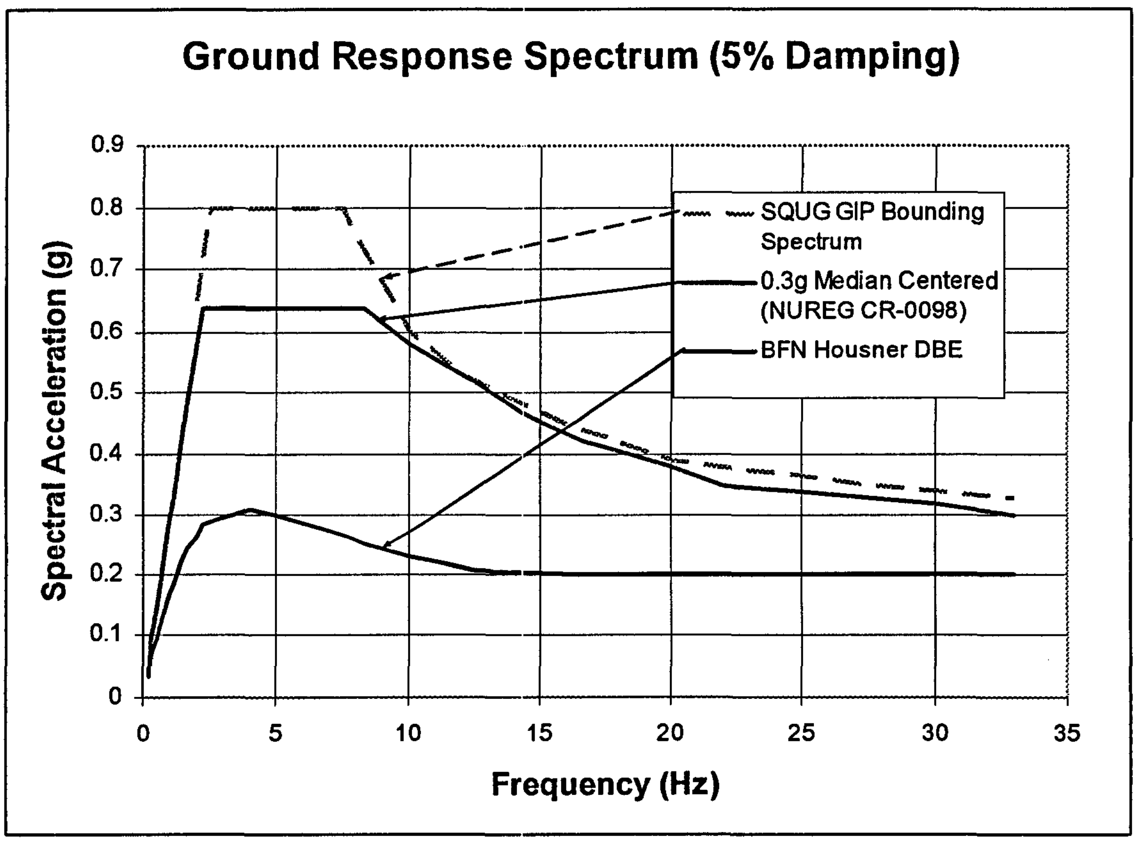
NRC Seismic Question 2:

Please provide a graph of the Review Level Earthquake (RLE) spectra used for the seismic margin assessment (SMA). On the same graph also provide the site Design Basis Earthquake (DBE) ground spectra (Housner spectra with 0.2g peak ground acceleration), and the USI A-46 spectra.

Discuss whether there is any exceedance of DBE or USI A-46 spectra over the RLE spectra in the frequency range of interest for the BFN Unit 1 systems, structures, and components (SSCs).

TVA Reply to Seismic Question 2:

Browns Ferry is a 0.3g focused scope plant for seismic IPEEE. The RLE ground motion response spectrum is based on 0.3g NUREG CR-0098 median spectral shape. The DBE is 0.2g Housner shaped spectrum. The DBE spectra were used for the USI A-46 evaluations. These spectra were defined as median centered for the A-46 evaluations, per the SQUG GIP. A comparison of the DBE and IPEEE ground motion response spectra is shown below. The IPEEE ground motion response spectrum fully envelopes the DBE ground motion response spectrum. The SQUG GIP Bounding Spectrum is also shown for information. The SQUG GIP Bounding Spectrum fully envelopes both the DBE and the IPEEE spectra.



Seismic IPEEE floor response spectra were determined by scaling up the DBE floor response spectra by a factor of 1.88 for the Reactor Building (see TVA Calculation No. CD-Q0000-940339, "Calculation of Basic Parameters for A-46 and Individual Plant Examination of External Events (IPEEE) Seismic Program," Rev. 1, sheet 49; Enclosure 2 of this submittal).

NRC Seismic Question 3:

Please describe the scope and the kind of seismic spectra used for the seismic review of the BFN Unit 1 Restart Project. How is the Restart Project seismic review coordinated with the seismic IPEEE/USI A-46 review?

TVA Reply to Seismic Question 3:

Sections 3 and 4 of Chapter III of the Browns Ferry Nuclear Performance Plan (NPP) (Reference 2) describe the seismic review programs to be conducted prior to restart. The programs utilize the Browns Ferry Design Basis Earthquake floor response spectra as described in Section 12.2 and Appendix C of the Browns Ferry Final Safety Analysis Report (FSAR) and in accordance with the respective TVA Browns Ferry Nuclear Plant seismic design criteria.

The USI A-46 and Seismic IPEEE reviews were conducted in parallel with the other ongoing design basis verification programs. As a result of performing the USI A-46 and Seismic IPEEE reviews in parallel with these other programs, final verification for certain items of equipment was not possible due to ongoing work associated with open Design Change Notice (DCN) packages. To enable final verification, a Punch List was developed to track each item of equipment requiring final verification. The Punch List is documented in Appendix H of "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report," Rev. 0, September 2004.

In addition, as documented in TVA letter to the NRC dated October 7, 2004, there are two regulatory commitments associated with the USI A-46 reviews:

1. BFN Unit 1 USI A-46 outliers will be resolved prior to restart of BFN Unit 1.
2. TVA will complete the operations review of the BFN Unit 1 A-46 verification following BFN Unit 1 procedural development and approval, and notify the NRC of the results of that review prior to restart of BFN Unit 1.

NRC Seismic Question 4:

BFN Unit 1 has been out of service since March 1985. Considering that safety-related SSCs in BFN Unit1 have been idle for 20 years, how does Tennessee Valley Authority (TVA) ensure that these SSCs will be in working order and will perform their designed safety functions properly, especially under the seismic DBE conditions? Preoperational tests (if to be performed) and limited IPEEE seismic walkdown performed may not uncover all the potential seismic problems due to age-related degradation of SSCs. Are all these addressed in the BFN Unit 1 Restart Project?

TVA Reply to Seismic Question 4:

Controls implemented during the extended shutdown of BFN Unit 1 to preserve systems and components in conjunction with the inspections, modifications and replacements, and other programs within the scope of the recovery effort will collectively provide assurance of the capability of the BFN Unit 1 structures, systems, and components to function properly, including under DBE conditions.

Plant Layup and Preservation Program

To protect the plant during the extended BFN Unit 1 shutdown, many of the BFN Unit 1 systems were placed in the BFN Unit 1 Plant Lay-up and Equipment Preservation Program (Plant Layup Program), while some systems remained in operation to maintain Unit 1 in its defueled condition or to provide necessary support of the operation of Units 2 and 3.

Systems that remained in operation were maintained in accordance with plant procedures; i.e., the internal operating conditions (e.g., water chemistry, flow rate, temperature, etc.) for these systems were maintained consistent with the operating units. In addition, the Unit 1, 2 and 3 reactor buildings are one continuous structure; accordingly, the external surfaces of BFN Unit 1 systems have been exposed to the same overall environmental conditions as the operating units. The normal BFN Unit 1 ventilation systems remained in service and equipment was maintained to prevent system leakage so that the equipment was not subjected to aggressive external conditions.

Many BFN Unit 1 systems were placed in the Plant Layup Program, and internal conditions controlled and monitored in accordance with that program. TVA has recently submitted substantial information explaining, in detail, various aspects of the Plant Layup Program. In support of the BFN license renewal applications, TVA submitted a letter dated February 19, 2004 (Reference 3), that provided a system-by-system detailed evaluation of the BFN Unit 1 layup conditions and associated Aging Management Review. That submittal contained a table, identifying system-by-system, the inspection/evaluation methodologies used to verify piping system integrity, and a description of the piping system refurbishment/ replacements. TVA letter dated July 19, 2004 (Reference 4), also submitted to support the BFN license renewal application, discussed evaluation of the effects of layup on BFN Unit 1 structures and supports. That evaluation identified no adverse effects of layup on BFN Unit 1 structures or component supports.

Identification of System and Component Replacements/Refurbishments

TVA letter dated January 31, 2005 (Reference 5), provided information explaining TVA's philosophy in identifying piping and equipment to be replaced, and that piping and equipment to be inspected, evaluated, and replaced, refurbished, or repaired as necessary. In short, while TVA maintained much of the plant under the Plant Layup Program, TVA took no credit for the Plant Layup Program in identifying the scope of system inspections/replacements to support BFN Unit 1 recovery. In addition, as license renewal approval was a key assumption in the economic feasibility of Unit 1 restart, the overall management philosophy for Unit 1 restart was to return the plant to operation in a condition that would support long-term safe and reliable operation of the unit, including the anticipated 20-year period following license renewal. Therefore, for some cases, TVA decided up front to replace entire piping sections and components, rather than expend extensive engineering resources to confirm that the existing piping and equipment was acceptable. TVA also decided up front to refurbish a large population of pumps and valves not already planned for replacement.

Reference 5 provides the results of this scoping process, and also identifies, system-by-system, the inspection/evaluation methodologies employed to determine system integrity, the results of those inspections, and the piping or sections of piping identified for repair/replacement.

BFN Unit 1 Recovery Project

The ongoing BFN Unit 1 recovery project is comprehensive in scope, and systematic in design. Key aspects (relative to system seismic capability) include:

- Inspection of piping and equipment not slated for replacement or refurbishment at the outset of the project to evaluate piping and equipment condition (see discussion above);

- Replacement/refurbishment of piping and equipment to ensure system design criteria is met and reliability ensured through the end of the current license period and through the extended period of operation associated with license renewal (see discussion above);
- Modification of the plant, where appropriate, to address design and operational issues resolved previously for BFN Units 2 and 3, and to make BFN Unit 1 functionally congruent to BFN Units 2 and 3;
- Following the regulatory framework for the restart of BFN Unit 1 submitted by TVA letter dated December 13, 2002 (Reference 6), and accepted by the NRC in Reference 7 which identifies the BFN Nuclear Performance Plan special programs to be completed (see further discussion below); and
- Resolution of outstanding NRC Generic Letter, NRC Bulletin, and NUREG-0737 Action Items not previously completed for Unit 1 due to the unit's shutdown in 1985 (also identified in Reference 6).

Reference 6 identifies the overall scope of the BFN Unit 1 recovery project, including a description of the NPP special programs being completed and the NRC generic communication issues being resolved prior to restart. The NPP special programs include:

- Long Term Torus Integrity Program;
- Large Bore Piping and Supports Program (Bulletins 79-02 and 79-14);
- Small Bore Piping and Instrument Tubing Program;
- Control Rod Drive (CRD) Insert and Withdrawal Piping Seismic Qualification Program;
- Drywell Steel Platforms and Upper Drywell Platforms;
- Miscellaneous Steel Frames Program;
- Cable Tray Supports Program;
- Conduit Supports Program;
- HVAC Duct Supports Program;
- Seismic Class II Over Class I / Spatial System Interactions Program; and
- Restart Test Program.

By letter dated August 15, 2005 (Reference 8), TVA provided a description of the BFN Unit 1 Restart Test Program along with descriptions of the BFN Unit 1 modifications being performed as part of the recovery effort.

Collectively, the BFN Unit 1 recovery project represents a comprehensive effort to inspect, evaluate, repair or replace, and/or modify the plant to ensure plant integrity, capability of plant SSCs to perform their design functions, and ensure long term plant reliability. Accordingly, based on these efforts, TVA is assured that BFN Unit 1 SSCs will be capable of performing their design functions under DBE conditions.

NRC Seismic Question 5:

Questions on Section 3 of the Submittal - System Description and Success Path Selection:

a) Section 3 of the submitted report states that "The success path selection and identification of components for the BFN1 seismic IPEEE program were based on the previous BFN2 and BFN3 seismic IPEEE programs." It further stated that "Success path logic diagrams (SPLDs) were constructed for the BFN2/3 seismic IPEEE ...." and "They (SPLDs) were used as a basis for the identification of the equipment to be included on the BFN2/3 seismic safe shutdown equipment list (SSELs)." This description is rather confusing.

   For the current BFN1 seismic IPEEE, did TVA prepare separate SPLDs and SSELs for BFN Unit 1 apart from those for BFN Unit 2 or BFN Unit 3? BFN Unit 1 should have its own set of SSELs that are basically different from those for BFN Units 2 and 3. If there is common equipment that appears on BFN Unit 1 and BFN Units 2 and 3 SSELs, please identify.

b) Section 3 of the Submittal provides two lists, one for the relevant plant functions and one for the front line systems to accomplish those functions, but there is no description regarding which function is accomplished by what front line system(s).

   Please confirm whether the following function/system match-up for BFN Units 2 and 3 also applies for BFN Unit 1:

   "The frontline systems selected to achieve the four shutdown functions are: a) control rod drive system (CRD) for reactivity control, b) safety/relief valves (SRVs) for reactor pressure control, c) core spray (CS) and low pressure coolant injection (LPCI) mode of residual heat removal system (RHR) (with reactor pressure vessel depressurization using SRVs) for reactor coolant inventory control, and d) suppression pool cooling mode of RHR for decay heat removal."

TVA Reply to Seismic Question 5:

Separate SPLDs and SSELs were prepared for BFN Unit 1. The SPLDs for BFN Unit 1 are the same as those constructed for BFN Unit 2 and BFN Unit 3.

The BFN Unit 1 USI A-46 Safe Shutdown Equipment List (SSEL) was expanded to include the additional items of equipment that are evaluated for Seismic IPEEE. This includes the items of equipment necessary for primary containment isolation and small LOCA mitigation. There are items of common equipment that appear on the BFN Unit 1 and BFN Units 2 and 3 SSELs.

Appendix B of "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report," Rev. 0, dated September 2004 (Enclosure 2, TVA letter to NRC dated October 7, 2004, Docket No. 50-259) provides the composite SSEL for BFN Unit 1. Common equipment items (also on BFN Units 2 and 3 SSEL) are identified with an asterisk following the SSEL number in the 1st column of the Table. IPEEE components are identified with an "I" entry in the 8th column of the Table ("Issue").

The plant functions / systems match-up for BFN Units 2 and 3 applies to Unit 1.

Chapter 2 of "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report," Rev. 0, dated September 2004, describes the safe shutdown paths chosen to mitigate a postulated DBE, which include:

- Reactivity Control,
- Reactor Coolant System Pressure Control,
- Reactor Coolant System Inventory Control, and
- Decay Heat Removal.

The primary path used for safe shutdown at BFN Unit 1 is insertion of the control rods and depressurization of the reactor coolant system using the main steam safety/relief valves (MSRVs) for pressure control. The Core Spray (CS) or Residual Heat Removal (RHR) system is then used to maintain reactor coolant inventory. The RHR system is also used for decay heat removal.

NRC Seismic Question 6:

Section 3.2.5.8, "Nonseismic Failures and Human Actions," of NUREG-1407 stated that for Electric Power Research Institute SMA, "Success paths are chosen based on a screening criterion applied to nonseismic failures and needed human actions. It is important that the failure modes and human actions are clearly identified and have low enough probabilities to not affect the seismic margins evaluation."

Please provide information as to how this was considered in choosing the success paths and the associated equipment for BFN Unit 1 SMA.

TVA Reply to Seismic Question 6:

As described in Chapter 2 of "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report," Rev. 0, dated September 2004 was previously transmitted to the NRC (Enclosure 2, TVA letter to NRC dated October 7, 2004, Docket No. 50-259), multiple trains were evaluated for each success path, thereby resolving nonseismic failure concerns.

Human Actions are addressed in accordance with the SQUG GIP. The "desk-top" review method will be used by the Operations Department to verify that existing normal, abnormal and emergency operating procedures are adequate to mitigate the postulated transient and that operators could place and maintain the plant in a safe shutdown condition. As documented in TVA letter to the NRC dated October 7, 2004, TVA will complete the operations review of the BFN Unit 1 A-46 verification following BFN Unit 1 procedural development and approval, and notify the NRC of the results of that review prior to restart of BFN Unit 1.

The systems and equipment selected for seismic review in the BFN Unit 1 USI A-46 program are consistent with those selected in the BFN Units 2 and 3 programs, which are those for which normal, abnormal, and emergency operating procedures are available to bring the plant from a normal operating mode to a hot shutdown condition. The BFN Units 2 and 3 shutdown paths for USI A-46 and Seismic IPEEE were reviewed by the BFN Operations staff, and as a result the plant abnormal operating procedure 0-AOI-100-5, "Earthquake," was revised to enhance the operator guidance necessary to verify and ensure diesel generator and electrical board operation, as well as identify specific instrumentation with the highest reliability following a seismic event. The operations personnel reviewed the specific actions required and concluded that the actions could be performed in the required amount of time with normally available resources.

Potential challenges to the operator were explicitly reviewed during validation of the pertinent plant operating procedures related to the FSAR, Chapter 14, Accident Analysis for the LOOP transient and Appendix R evaluations which preceded the A-46 program review. In addition, the potential for local failure of architectural features and the potential for adverse interactions in the vicinity of safe shutdown equipment, where local operator action may be required, were reviewed as part of the BFN Units 2 and 3 A-46 resolution process. There were no seismic or housekeeping issues affecting the control room. Seismic interaction reviews eliminated any concerns with the plant components and structures located in the immediate vicinity of the components which had to be manipulated. Therefore, the potential for physical barriers resulting from equipment or structural earthquake damage which could inhibit operator ability to access plant equipment was considered, and the potential barrier to successful operator performance was eliminated. The BFN Unit 1 A-46 reviews similarly eliminated all seismic interaction concerns.

NRC Seismic Question 7:

Section 7 and Appendix C of NUREG-1407 states that a peer review should be conducted by individuals who are not associated with the initial evaluation, to ensure the accuracy of the documentation and to validate both the IPEEE process and its results. The Submittal has no mention of any peer review performed.

Please provide the following information for the seismic IPEEE: (1) composition of peer review team, (2) areas of peer review and major comments, and (3) resolution of comments.

TVA Reply to Seismic Question 7:

The BFN Unit1 seismic IPEEE peer review was performed by Dr. James J. Johnson in conjunction with his peer review of the BFN Unit 1 USI A-46 program as documented in Chapter 6 and Appendix G of "Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report," Rev. 0, dated September 2004 (Enclosure 2, TVA letter to NRC dated October 7, 2004, Docket No. 50-259).

The peer review included the safe shutdown equipment selection, cable tray and conduit raceways, mechanical and electrical equipment for A-46 and Seismic IPEEE as an integrated implementation program, and High Confidence Low Probability of Failure (HCLPF) capacity determination. The scope of the peer review focused on those aspects of the seismic IPEEE program not reviewed previously for the integrated program, and involved in-plant observation, walkdown documentation, and HCLPF calculations for selected components and plant features.

The Peer review concluded "The approach and result are consistent with the EPRI NP-6041-SL and acceptable in response to the US NRC Supplement 4 to Generic Letter 88-20 for BFN Unit 1." No major comments were identified in this peer review. Some clarifications were provided by the Seismic Capability Engineers as to the precise meaning of selected sections of the summary report, thereby reaching concurrence on these issues.

NRC Seismic Question 8:

Please provide copies of the following high confidence of low probability of failure (HCLPF) calculation for equipment not screened out:

a) MCC (ID No: 1-BDBB-281-0001A)

b) RHR Heat Exchanger (ID No: 1-HEX-74-900A)

Flat bottom tanks were identified in many previous seismic IPEEE reviews as components with potential low HCLPFs, but the Submittal has no discussion on this. Was the condensate storage tank included in the SSEL? If the condensate storage tank was not included, please explain why.

TVA Reply to Seismic Question 8:

Enclosed in Enclosures 4 and 5, respectively, of this submittal are copies of the following TVA calculations:

a) TVA Calculation CDQ1 999 2004 0156, "HCLPF Calculations of MCC Anchorage for Seismic IPEEE Program," Rev. 0, June 9, 2004.

b) TVA Calculation CDQ1 074 2004 0160, "HCLPF Calculations of RHR Heat Exchanger Anchorage for Seismic IPEEE Program," Rev. 0, June 9, 2004.

The condensate storage tank was not included in the SSEL. The reactor decay heat removal function is accomplished by relieving steam from the reactor via the lifting of the main steam safety/relief valves (MSRVs) at their respective set points into the suppression pool. The MSRVs could be manually operated by the control room operator to lower reactor pressure so that the low pressure coolant injection (LPCI) mode of residual heat removal (RHR) could be initiated for reactor coolant inventory control. In this mode, the LPCI takes suction from the suppression pool. The decay heat removal would be achieved by placing the RHR system in the suppression pool cooling (SPC) mode of operation. During the SPC mode of RHR, the RHR pump takes suction from and discharges to the suppression pool via the RHR heat exchangers. The service water system would provide the capability to transfer the decay heat from the RHR system to the ultimate heat sink.

NRC Seismic Question 9:

Chapter 6 and Table C.1, Item 3.2 of NUREG-1407 requires that coordination with ongoing programs and other seismic issues (such as USI A-17, 40, 45, eastern U.S. seismicity issue and other seismic safety issues such as Generic Safety Issue (GSI)-156, "Systematic Evaluation Program", GSI-172, "Multiple System Response Program (MSRP)") be described in the IPEEE submittal. Other than seismic induced fire/flood evaluation and USI A-46, the Submittal did not provide information on the coordination with ongoing programs and other seismic issues.

Please provide the missing information.

TVA Reply to Seismic Question 9:

USI A-17, "System Interactions in Nuclear Power Plants," addresses NRC's concerns regarding the interaction of various systems with regard to whether actions of consequences could adversely affect the redundancy and independence of safety systems. The seismic systems interaction concerns consist of seismic spatial interactions (failure and falling; proximity and impact; and flexibility of attached lines) and seismic induced flooding and fire. Per Section 2.3.6 of Part 1 of the SQUG GIP 2A, successful completion of USI A-46 fully addresses, without any other actions, USI A-17 as it applies to seismic spatial interactions. Seismic induced flooding and fire are addressed in Chapter 8 of the submittal seismic IPEEE report.

USI A-40, "Seismic Design Criteria," investigates selected areas of the seismic design process. Required action is limited to the seismic adequacy of safety related above ground tanks. Per Section 2.3.6 of Part 1 of the SQUG GIP 2A, successful completion of USI A-46 fully addresses, without any other actions, USI A-40 as it applies to seismic adequacy of tanks and heat exchangers. Note that there are no large above ground flat bottom tanks on the BFN Unit 1 SSEL (see response to second part of RAI # 8).

USI A-45, "Shutdown Decay Heat Removal Requirements," has the objective of determining whether the decay heat removal function at operating plants is adequate and if cost-beneficial improvement could be identified. As required, the seismic adequacy of the decay heat removal system is included in the BFN Unit 1 seismic IPEEE program. No significant or unique seismic vulnerabilities were identified in the decay heat removal function.

The "Eastern U.S. Seismicity Issue" addresses the likelihood of earthquake events exceeding the seismic design basis for plants. The NRC identified eight plants at five Eastern U.S. sites as outliers. BFN is not one of the five sites. Furthermore, per Section 6.3.3.2 of NUREG-1407, the IPEEE provides resolution for this issue without requiring additional analysis or documentation.

GSI-156, "Systematic Evaluation Program," included reviews of 11 older operating nuclear power plants. BFN was not a Systematic Evaluation Program (SEP) plant. During the course of the SEP and NRC Seismic Qualification Review Team (SQRT) reviews, the staff identified a concern with the anchorage and supports for electrical equipment in the SEP plants. An information notice concerning this issue was sent to all operating plants by the NRC in May 1980. Eventually, the NRC staff issued USI A-46 to address the seismic adequacy of mechanical and electrical equipment in all older operating plants, including BFN Unit 1. No additional evaluation is required, and NUREG-1407 does not mention the SEP program.

GSI-172, "Multiple System Responses Program" (MSRP), was raised by the ACRS during the review of several other generic safety issues such as USI A-17 and USI A-46, because of a concern that due to scope limitations on each issue, and a possible lack of coordination between issues, there was a possibility that some potentially significant safety concerns were not being addressed. The MSRP program identified 21 potential safety issues. Following the completion of the IPE and IPEEE programs, the issue was closed. No additional evaluation is required, and NUREG-1407 does not mention the MSRP program.

NRC Seismic Question 10:

The evaluation of seismic induced flooding in the Submittal does not discuss the failure potential of dams upstream of BFN Unit 1 and its consequences to BFN Unit 1. Generic Letter 88-20, Supplement 5, does not exclude review of dams, levees or dikes and consequences of their failures.

Please provide this information.

TVA Reply to Seismic Question 10:

The external flood evaluation for BFN Unit 1 was included in the report "Browns Ferry Nuclear Plant, Individual Plant Examination for External Events (IPEEE), Internal Fires, High Winds, Floods, Transportation and Nearby Facility Accidents," July 1995, transmitted to the NRC as enclosure to letter dated July 24, 1995. TVA provided further information concerning the probability and consequences of flooding due to failure of upstream dams in a submittal document dated January 29, 1999. This issue was resolved for all three BFN Units in Reference 1.

The BFN Unit 1 Reactor Building is integral with the BFN Units 2 and 3 Reactor Buildings, and is at the same elevation above sea level as BFN Units 2 and 3 for which TVA has satisfactorily addressed IPEEE. No further evaluation is required.

REFERENCES

1. NRC Letter to TVA, "Browns Ferry Units 1, 2, and 3, Individual Plant Examination of External Events (IPEEE) and Related Generic Safety Events, Staff Evaluation (TAC Nos. M83595, M83596, M83697)," dated June 22, 2000

2. TVA letter to NRC dated August 28, 1986

3. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - January 28, 2004 Meeting Follow-Up - Additional Information," dated February 19, 2004

4. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN), Units 1, 2 And 3 (TAC NOS. MC1768, MC1769, and MC1770) License Renewal Application: Response To Request For Additional Information (RAI) Regarding Lay-Up Effects of Unit 1 Structures and Component Supports," dated July 19, 2004

5. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application (LRA) - Relating to Section 3.0 Unit 1 Lay Up Questions - Response to Aging of Mechanical Systems During the Extended Outage of Browns Ferry Nuclear Plant Unit 1 - NRC Request for Additional Information (RAI) (TAC Nos. MC1704, MC1705, and MC1706)," dated January 31, 2005

6. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Regulatory Framework for the Restart of Unit 1," dated December 13, 2002

7. NRC letter to TVA, "Regulatory Framework for the Restart of Browns Ferry Nuclear Plant, Unit 1 (TAC MB7679)," dated August 14, 2003

8. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Response to NRC Request for Additional Information Regarding the Restart Testing Program (TAC NO. MC7208)," dated August 15, 2005