TVA

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

April 29, 2009

TVA-BFN-TS-418
TVA-BFN-TS-431

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) – UNITS 1, 2, AND 3 – TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418 – EXTENDED POWER UPRATE (EPU) – RESPONSE TO REQUEST FOR INFORMATION REGARDING CONTAINMENT PARAMETERS FOR NRC CONFIRMATORY ANALYSIS (TAC NOS. MD5262, MD5263, AND MD5264)

By letters dated June 28, 2004 and June 25, 2004 (ADAMS Accession Nos. ML041840109 and ML041840301), TVA submitted license amendment requests to the NRC for the EPU operation of BFN Unit 1 and BFN Units 2 and 3, respectively. The proposed amendments would change the operating licenses to increase the maximum authorized core thermal power level of each reactor by approximately 14 percent to 3952 megawatts.

To support the NRC performance of a confirmatory containment analysis using the Gothic code, NRC verbally requested that TVA provide a copy of Form OPL-4a. Form OPL-4a is used by TVA and General Electric-Hitachi to document containment analysis input parameters. A draft copy of OPL-4a was e-mailed to NRC on March 18, 2009, which was followed by a TVA submittal of a modified version of Form OPL-4a on April 10, 2009 (ML091060381). By e-mail on April 6, 2009, TVA received questions on the draft OPL-4a transmittal and several requests for plant physical data related to primary containment components. The enclosure to this letter provides a response to the April 6, 2009, information request.

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TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

No new regulatory commitments are made in this submittal. If you have any questions regarding this letter, please contact J. D. Wolcott at (256) 729-2495.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 29th day of April, 2009.

Sincerely,

R. G. West
Site Vice President

Enclosure:

Response to Request for Information Regarding Containment Parameters for NRC Confirmatory Analysis

Enclosure:

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418
EXTENDED POWER UPRATE (EPU)

RESPONSE TO REQUEST FOR INFORMATION REGARDING CONTAINMENT PARAMETERS FOR NRC CONFIRMATORY ANALYSIS

On March 18, 2009, the Tennessee Valley Authority provided some draft information by e-mail regarding the containment analysis for the Browns Ferry EPU. The following questions are related to the draft document.

NRC SCVB-75/73

Provide the following information:

a. Liquid vapor interface area in reactor vessel;
b. Data for heat sinks (material, mass, and surface area) in drywell (DW), wetwell (WW) airspace, and suppression pool);
c. Size of each vacuum breaker pipe and its length between WW and DW;
d. Reactor pressure vessel (RPV) metal thickness (average if it varies);
e. Elevations of suppression pool low and high water level;
f. Elevations of vacuum breaker line inlet (in WW) and outlet (in DW) area center;
g. Elevations of core spray (CS) suction line, and residual heat removal (RHR) suction line at the suppression pool;
h. For Appendix R event analysis, provide the heat removal rate by each drywell fan cooler.

TVA Response to SCVB-75/73

a. The vessel fluid (liquid water and vapor) in the General Electric - Hitachi (GEH) containment codes (M3CPT and SHEX) is modeled as a single fluid node. It is assumed at all times that the vessel liquid and vessel vapor are in thermodynamic equilibrium with each other at a saturation condition corresponding to the vessel pressure and temperature. Since thermodynamic equilibrium between the vessel vapor and liquid is imposed in the calculation, heat and mass transfer between the vessel liquid and vapor is not mechanistically modeled and a liquid-vapor interface area in the reactor vessel is not used in the GEH containment analysis.

b. Tables 1, 2, and 3 provide the heat sink data compiled for use in the EPU containment analyses for BFN Units 1, 2, and 3.

Table 1 - Drywell Heat Sink Data

Sink No.	Sink[1] Description	Total Exposed Surface Area (square feet)	Material	Shell Thickness (feet)	Exposure[2]	Mass[3] (pounds mass (lbm))
1	Zone I	3230	carbon steel	0.125	1	197838
2	Zone II	2770	carbon steel	0.062	1	84152.6
3	Zone III	5750	carbon steel	0.067	1	188773
4	Zone IV	5150	carbon steel	0.083	1	209451
5	LOCA Vent System	17130	carbon steel	0.021	1, 2	176268

Notes:

1. *See Figure 1 for drywell zones identification.*
2. Exposure –
   1 = Drywell atmosphere
   2 = Wetwell atmosphere
3. Mass calculated as the Total Surface Area * Shell Thickness * Density (Table 3).

Table 2 - Wetwell Heat Sink Data

Sink No.	Sink Description	Total Exposed Surface Area (square feet)	Material	Total Thickness (feet)	Exposure	Mass (lbm)[1]
1	Upper Torus Shell	17057	carbon steel	0.063	wetwell airspace	526554
2	Lower Torus Shell	17057	carbon steel	0.063	suppression pool	526554

Notes:

1. Mass calculated as the Total Surface Area * Thickness * Density (Table 3).

Table 3 - Thermo-physical Properties of Passive Heat Sink Materials Assumed in Analysis

Material	Density (lbm/feet^3)	Specific Heat (British Thermal Units (BTU) /lbm-°F)	Thermal Conductivity (BTU/lbm-ft-°F)
carbon steel	490	0.11	26

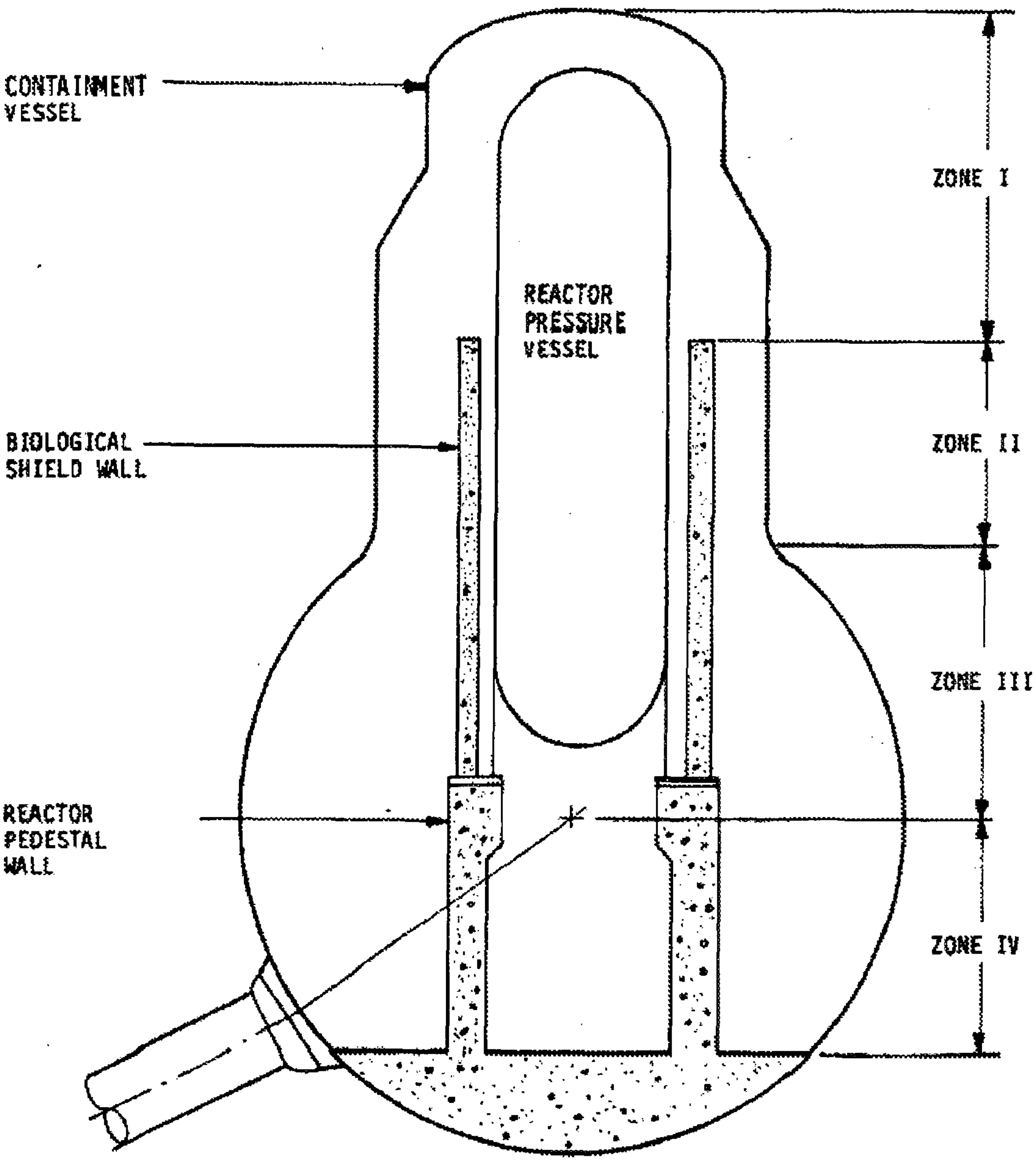

Figure 1

c. See response to item f.

d. An average RPV metal thickness of 6.56 inches (0.547 feet) was used to develop the vessel metal nodal model inputs for the BFN Unit 1, 2 and 3 EPU containment analyses.

e. TS require that suppression pool water level be ≥ -6.25 inches with and ≥ -7.25 inches without differential pressure control, and ≤ -1.0 inches. Normal operation is with differential pressure established. The low and high suppression pool levels in terms of building elevation are shown below.

Parameter	TS Value	Building Elevation
Suppression Pool Low Level	- 6.25 inches	536 feet 7 inches
Suppression Pool High Level	- 1.0 inches	536 feet 1¾ inches

f. The suppression pool-to-drywell vacuum breakers are 18 inch diameter. The centerline on the vacuum breaker inlet in the wetwell airspace is at building elevation 541 feet 5 inches. The vacuum breaker is connected to an 18 inch diameter pipe that penetrates the drywell vent header and extends into the drywell header air space. The length of the pipe is approximately 2 feet and the centerline of the end of the pipe inside the drywell header air space is at building elevation 542 feet 10½ inches.

g. The RHR and CS pumps take suction from a ring header that attaches to the suppression pool. The centerline of the ring header is at building elevation 525 feet 4 inches.

h. The drywell cooler inputs developed for the Appendix R containment response analyses for the net positive suction head (NPSH) evaluation assumed a heat removal rate of 176.7 BTU/seconds for each of 10 available drywell fan coolers. The Appendix R containment analysis assumes this is a base cooling rate corresponding to a drywell temperature of 135°F and a Reactor Building Closed Cooling Water (RBCCW) system inlet temperature of 100°F. The Appendix R containment calculation uses this base heat removal to calculate a time varying heat removal rate with the assumption that the drywell cooler heat removal rate is approximately linearly proportional to the drywell to RBCCW temperature difference assuming a constant RBCCW inlet temperature of 100°F.

NRC SCVB-76/74

FW flow rate at 102 percent EPU, page 17 of Draft OPL 4a gives 28Mlb/hr, Figure 1-2 of Unit 1 PUSAR gives 16781000 lb/hr for FW flow. Explain the difference.

TVA Response to SCVB-76/74

The entry "28" in the draft OPL-4a is a referral to Reference 28, which is listed in the references section of the OPL-4.a. The feedwater flow rate from Reference 28 is 16.781 million pounds per hour, which is the same as that shown in the Unit 1 Power Uprate Safety Analysis Report (PUSAR) containment analyses figure.

NRC SCVB-77/75

On page 5 of the Draft OPL 4a, the LOCA break area for long term response for the design basis accident (DBA) is 4.2 ft2 and for NPSH is 1.94 ft2. Specify the type of breaks, and explain why different type of breaks were considered for these cases.

TVA Response to SCVB-77/75

The short-term (10 minute duration) SHEX calculation for NPSH evaluation assumes a break in the discharge side of the recirculation loop (with a corresponding break area of 1.94 square feet (ft^2) because this represents the limiting break configuration for the 10-minute short-term containment response for NPSH evaluation. The long-term design basis accident (DBA) Loss-of-Coolant Accident (LOCA) analyses and the long-term DBA LOCA NPSH evaluation both use a break area of 4.2 ft^2, which corresponds to the DBA LOCA recirculation loop suction side break area. Form OPL-4a does not correctly reflect this difference.

NRC SCVB-78/76

Provide reference 2 given in Draft OPL 4a page 23.

TVA Response to SCVB-78/76

Reference 2 is Task Report 0400, Containment System Response, which was prepared for TVA by GEH in support of EPU. Enclosure 7 of TVA's March 23, 2006, submittal provided the inputs, assumptions, methodology, and results from the task report. The task report has been since superseded by later calculations and submittals, and the task report itself is not updated to reflect these subsequent changes. Therefore, the report is not suitable for submittal. TVA will, however, make the task report available for inspection.

NRC SCVB-79/77

The vacuum breaker loss coefficient per draft OPL-4a page 18 is 0.45, address whether this includes entrance and exit loss coefficients.

TVA Response to SCVB-79/77

The wetwell-to-drywell vacuum breaker modeling inputs to the BFN EPU containment analysis applied the loss coefficient shown in the OPL-4a as the complete vacuum breaker system loss coefficient (including exit and entrance losses).

NRC SCVB-80/78

On the last line of Page 7, discuss what is meant by drywell holdup volume and why it is not modeled. On Page 8, discuss why the drywell pool surface area not modeled.

TVA Response to SCVB-80/78

Drywell holdup volume is that volume, usually at the bottom of the drywell, where water falling to the drywell floor is "held up" before spilling over to the wetwell via the vent lines. It is typically more of a consideration for a Mark III containment, which has a large drywell pool volume. The effect of the holdup volume is relatively small due to the small holdup volume associated with the Mark I drywell such as BFN and can be neglected. Therefore, the volume was not modeled in the GEH containment analysis for BFN. If the drywell holdup volume is not modeled, a surface area corresponding to this drywell holdup volume is likewise not modeled. Although the holdup volume is not used in the GEH containment analysis, it is accounted for in the TVA NPSH calculations for event analyses that spray water in the drywell and in the LOCA analysis. The drywell holdup volume is approximately 28,600 gallons and is included in the TVA NPSH calculation as a reduction in suppression pool level.

NRC SCVB-81/79

The safe shutdown analysis document for Units 2 and 3, page 277 of 559, states

> When reactor pressure drops below the shutoff head of the RHR system, the RHR system operating in the LPCI mode, would inject subcooled water into the reactor.

According to draft OPL-4a Page 14, the low pressure coolant injection (LPCI) pump shutoff head is 319.5 psig. In the Appendix R analysis, address whether LPCI is assumed to be initiated automatically based on its shutoff head setpoint after the operator depressurizes the reactor using 3 safety relief valves (SRVs). In case LPCI is assumed to be operator initiated in the analysis, provide the initiation time.

TVA Response to SCVB-81/79

The Appendix R Safe Shutdown Analysis is maintained in the BFN Fire Protection Report (FPR) and the NRC reference is from a previous Units 2 and 3 FPR. With Unit 1 in service, the FPR is now written for all three units and has the same stipulation that is quoted in the question.

In the Appendix R Safe Shutdown Procedures, which are called Safe Shutdown Instructions (SSIs), the operator starts a RHR pump and opens SRVs within 20 minutes into the limiting Appendix R event. Opening the SRVs results in depressurizing the reactor. When reactor pressure decreases below 450 pounds per square inch gauge (psig), the SSIs instruct the operator to open the inboard LPCI inboard injection valve, which completes the LPCI injection path to the reactor. The operating RHR pump would then begin to inject (automatically) to the vessel when reactor pressure decreased below the RHR pump shutoff head (319.5 psig). The Appendix R analysis assumes the depressurization starts at 25 minutes.