May 24, 2005

10 CFR 54

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

Gentlemen:

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 - LICENSE RENEWAL APPLICATION (LRA) – RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION CONCERNING FOLLOW UP TO RAI 2.4-2, FOLLOW UP TO RAI 3.5-5, FOLLOW UP TO RAI 3.5-14, AND FOLLOW UP TO RAI 4.7.4-1 (TAC NOS. MC1704, MC1705, AND MC1706)

By letter dated December 31, 2003, TVA submitted, for NRC review, an application pursuant to 10 CFR 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2, and 3. As part of its review of TVA's LRA, the NRC staff, through a series of informal requests beginning on April 5, 2005, requested additional information. This letter addresses concerns in the following areas: follow up to RAI 2.4-2, follow up to RAI 3.5-5, follow up to RAI 3.5-14, and follow up to RAI 4.7.4-1. The remainder of the concerns will be addressed in separate correspondence currently scheduled to be submitted by May 27, 2005.

The enclosure to this letter contains the specific NRC requests for additional information and the corresponding TVA responses.

If you have any questions regarding this information, please contact Ken Brune, Browns Ferry License Renewal Project Manager, at (423) 751-8421.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 24th day of May, 2005.

Sincerely,

Original signed by:

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosure:
cc: See page 3

Enclosure
cc (Enclosure):

State Health Officer
Alabama Department of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, Alabama 36130-3017

Chairman
Limestone County Commission
310 West Washington Street
Athens, Alabama 35611

(Via NRC Electronic Distribution)
Enclosure
cc (Enclosure):

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. Stephen J. Cahill, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

NRC Unit 1 Restart Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

cc: continued page 4

cc: (Enclosure)

Margaret Chernoff, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Eva A. Brown, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Yoira K. Diaz-Sanabria, Project Manager
U.S. Nuclear Regulatory Commission
(MS 011F1)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Ramachandran Subbaratnam, Project Manager
U.S. Nuclear Regulatory Commission
(MS 011F1)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

JEM:TLE:BAB
Enclosure
cc (Enclosure):

A. S. Bhatnagar, LP 6-C
K. A. Brune, LP 4F-C
J. C. Fornicola, LP 6A-C
R. G. Jones, NAB 1A-BFN
K. L. Krueger, POB 2C-BFN
R. F. Marks, Jr., PAB 1A-BFN
F. C. Mashburn, BR 4X-C
N. M. Moon, LP 6A-C
J. R. Rupert, NAB 1F-BFN
K. W. Singer, LP 6A-C
M. D. Skaggs, PAB 1E-BFN
E. J. Vigluicci, ET 11A-K
NSRB Support, LP 5M-C
EDMS, WT CA-K

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ENCLOSURE

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

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) CONCERNING FOLLOW UP TO RAI 2.4-2, FOLLOW UP TO RAI 3.5-5, FOLLOW UP TO RAI 3.5-14, AND FOLLOW UP TO RAI 4.7.4-1

(SEE ATTACHED)

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

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) CONCERNING FOLLOW UP TO RAI 2.4-2, FOLLOW UP TO RAI 3.5-5, FOLLOW UP TO RAI 3.5-14, AND FOLLOW UP TO RAI 4.7.4-1

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This enclosure contains the specific NRC request(s) for additional information and the corresponding TVA response(s).

NRC Follow up to RAI 2.4-2

Based on the response to RAI 2.4-2, the staff finds that the components identified in the RAI are covered under the scope of Section 2.4.1 of the LRA except item (f), which is covered under the scope of Section 2.3 of the LRA. However, 10 CFR 54.4(a) and (b) require identification of all in-scope structures and components and their intended functions. The staff reviewer may stretch its imagination, and accept inclusion of the drywell and suppression chamber supports [items (j) and (k)] as within the scope of license renewal. However, an absence of all the structural components internal to drywells and the suppression chambers (Items (a) to (e), and items (g) and (h)) from Table 2.4.1.1 would give an impression that they are not within the scope of license renewal. The applicant is requested to explicitly incorporate the components internal to drywell and suppression chambers within the scope of license renewal, through cross referencing, if necessary. Note: This can be done during the next update of the LRA.

TVA Response to Follow up RAI 2.4-2

The methodology used to determine the components within the scope of license renewal is described in LRA Section 2.1.4.3.3, Structural Component Scoping, and reads as follows:

> "For structures determined to be within the scope of 10 CFR 54, detailed structural drawings were reviewed to identify structural components (such as structural steel, foundations, floors, walls, ceilings, penetrations or stairways). For in-scope structures, all structural components that are required to support the intended functions of the structure were identified as in-scope of 10 CFR 54. These structural components were generally evaluated as generic structural commodities, not as individual components."

LRA Section 2.4.1.1 addresses the Primary Containment Structure and includes all component types as noted in Table 2.4.1.1. The Component Type "Reinforced Concrete Beams, Columns, Walls, and Slabs" includes the concrete of the reactor vessel support pedestal and other structural concrete located within the Primary Containment Structure. The Component Type "High Density Shielding Concrete" includes the concrete of the biological shield wall. The Component Type "Structural Steel Beams, Columns, Plates, Trusses" includes the plates that form the cylindrical shell of the biological shield wall and other structural steel components such as the steel platforms located within the Primary Containment Structure. The Component Type "Steel Containment Elements" includes the stabilizers between the biological shield wall and containment shell, reactor pressure vessel (RPV) male stabilizer bracket and RPV female stabilizer and anchor bolts, drywell, drywell steel support skirt and anchor bolts, drywell head and closure bolts, torus and torus ring girder, embedded steel, and other components that comprise the primary containment boundary of the Primary Containment Structure. The Component Type "Compressible Joints and Seals" includes the gasket material used in the drywell head seal, drywell and torus access hatch seals, personnel access doors and penetration seals located in the Primary Containment Structure.

Components identified as supports that are located within the Primary Containment Structure were addressed in Section 2.4.8.1, Structures and Component Supports Commodity Group. The Component Type "ASME Equivalent Supports and Components" includes the anchor bolts of the RPV support skirt, RPV ring girder and anchor bolts and other supports for ASME class 1 and class 2 piping within the Primary Containment Structure.

The reactor pressure vessel (RPV), even though it is located within the Primary Containment Structure, is considered as a mechanical piece of equipment and is scoped within the mechanical system section of the LRA, Section 2.3.1, Reactor Coolant Systems. This is consistent with industry practice for this component.

NRC Follow Up to RAI 3.5-5

Note that item 8 of Section 3.5.2.2.2.1 of the LRA states in part: "The upper elevations of the sacrificial shield wall may exceed 150 degree F briefly and infrequently, during abnormal operations and is not considered to affect its function." The upper elevation of the sacrificial shield wall inside the drywell shell is not a load bearing reinforced concrete structure.

a. The drywell closed cooling ventilation system is a non-safety related system and not in scope for License Renewal. This function is not required for Safe Shutdown of the plant. If this cooling system function is lost, operator action will be required when the Technical Specifications for drywell temperature limits exceeds 150°F.

b. A review of Browns Ferry Structures Monitoring Baseline inspection and the results for the first Structures Monitoring inspection period did not reveal any loss of intended function due to aging effects of the RPV pedestal supports, the foundation and floor slab, and the sacrificial shield wall.

c. Appendix A of ACI 349-85 specifies that the concrete temperature limits for normal operation or any other long term period shall not exceed 150°F except for local areas, which are allowed to have increased temperatures not to exceed 200°F. With the exception of the main steam tunnels in the Reactor Building, BFN reinforced concrete structures have general area temperatures less than 150°F during normal operation. The general area temperatures have been conservatively evaluated using maximum normal space ambient temperatures noted on the Harsh Environmental drawing series and associated calculations. The Unit 1, 2, and 3 main steam tunnels at BFN have a maximum normal space ambient temperature of 160°F as noted in the Harsh Environmental drawing series and associated calculations. Note however, that this is a maximum normal space ambient temperature. The TVA Harsh Environmental drawing series and associated calculations identify the average normal space ambient temperature as 135°F. This is judged to be acceptable because when concrete is subjected to prolonged exposure to elevated temperatures, reductions in excess of 10 percent of the compressive strength, tensile strength, and the modulus of elasticity only begin to occur in the range of 180°F to 200°F. (Reference EPRI TR-103842, July 1994).

Therefore, the conditions identified in NUREG-1801 are satisfied and aging management of reduction of strength and modulus due to elevated temperature for concrete components at BFN is not required.

The staff recognizes the temperature thresholds, and has no problem with the EPRI TR position. However, at these temperatures, the concrete structures go through additional shrinkage cracking, and spalling. The staff's basic concern is related to the degradation of pedestals supporting the reactor vessels and that of seismic restraints anchored to the sacrificial shields and the drywell. The staff had expected more description regarding the concerns in response to item "b."

TVA Response to Follow Up to RAI 3.5-5

The inspection of concrete within the drywell is conducted per BFN procedure LCEI-CI-C9, "Procedure for Walkdown of Structures for Maintenance Rule." This LCEI provides the basis for BFN monitoring/inspection tasks, examination criteria, evaluation requirements, and acceptance criteria in compliance with the Maintenance Rule [10 CFR 50.65]. A baseline inspection for BFN was established in 1997 and subsequent inspections are performed on a five-year frequency. Section 7.2 of LCEI-CI-C9 provides inspection guidelines, and visual inspections of structural conditions is used as the method to detect degradation. Visual inspection is an acceptable technique and is consistent with techniques identified in industry codes and standards such as ACI 349.3R-96. Inspection checklists (Attachment 1 of LCEI-CI-C9) are used to document inspection results/defects. Section 7.3 of LCEI-CI-C9 provides guidance for evaluation of the results documented on the inspection checklists. The acceptance criteria are defined in Section 7.3 of LCEI-CI-C9 as: (1) acceptable, (2) acceptable with deficiencies and (3) unacceptable. The latest inspection of the concrete of the reactor vessel support pedestal, biological or sacrificial shield wall and other structural concrete within the Primary Containment Structure had been completed by 2002 for Units 2 and 3. All concrete elements within the Primary Containment Structure for Units 2 and 3 were found to be acceptable.

NRC Follow Up to RAI 3.5-14

In the letter dated January 31, 2005, the applicant provided the following response:

A total of 16 boral coupons were placed in the Unit 3 spent fuel storage pool (SFSP) in October 1983. The coupons supplied by the rack manufacturer are of the same metallurgical condition as the high density fuel storage racks (HDFSR) in thickness, chemistry, finish, and temper. For the first six years of the planned fifteen year surveillance program, examination was to have taken place at two-year intervals. Accordingly, two coupons were removed in October 1985. Blisters were found upon examination, and because of this unexpected anomaly, three additional coupons were analyzed not finding any blisters. As a result of blisters found on the coupons removed in 1985, the surveillance program has been expanded to include monitoring the formation and behavior of these blisters. These boral coupons are periodically removed from the fuel pool for testing and are evaluated for corrosion or other degradation of the neutron absorber plates by comparing various physical characteristics of the test coupons to baseline measurements taken when the coupons were installed. Also, a metallurgical engineer examines the coupons for general corrosion, local pitting, and bonding. No further blisters, corrosion, or degradation has been identified in coupons evaluated through 2003.

The above response to the RAI states that these boral coupons are periodically removed from the fuel pool for testing and are evaluated for corrosion or other degradation of the neutron absorber plates by comparing various physical characteristics of the test coupons to baseline measurements taken when the coupons were installed. The response implies that a metallurgical engineer also periodically examines the coupons for general corrosion, local pitting, and bonding. And no further blisters, corrosion, or degradation has been identified in coupons evaluated through 2003. However, it is not clear to the staff if these periodic inspection activities are an extension of the 1983 one-time inspection program covering boral coupon test specimens or a separate AMP in addition to the Chemistry Control Program mentioned above. BFN is requested to clarify the key parameters of this periodic inspection program or activity including the objective, scope, frequency and inspection approach of the program.

TVA Response to Follow Up to RAI 3.5-14

The Boral coupon inspection program was initiated in 1983 to implement the inspection and testing requirements of UFSAR Section 10.3.6; this checks the long-term behavior of the material of the high density spent fuel racks. The inspection is performed per BFN Technical Instruction (TI) TI-116, “High Density Fuel Storage System Surveillance Program.” When the TI is performed, Boral coupons are removed from the spent fuel storage pool and examined by the Metallurgical Engineer in their original condition to determine if sampling of surface corrosion products is appropriate. Thickness measurements are obtained of each coupon and documented in accordance with the TI. If degradation is such that further investigation is warranted, a minimum of one coupon is selected to be unsheathed or opened. Prior to the unsheathing process, a dye penetrant test for indications on the outer surfaces of the coupon will be performed and is examined by the Metallurgical Engineer. The Metallurgical Engineer decides if further unsheathing of the coupons is required. The visual examination by the Metallurgical Engineer is documented on the appropriate forms of the TI. The current frequency for performing this TI is two years. The surveillance frequency is re-evaluated each time the surveillance is performed and can be changed based on the trend of the historical data results. The inspection of the Boral coupons will continue until such time as the trend of the historical data results collected provides a basis to discontinue the inspections.

NRC Follow up RAI 4.7.4-1

Please provide tests or other research publication based justification for making the following assertion in part of the response:

> "...total dose of less than 1.0E8 rads. The material properties of the polyurethane foam will remain within the limits assumed by the original analysis for a total dose of less than 1.0 E08 rads."

TVA Response to Follow up RAI 4.7.4-1

The basis for asserting that the polyurethane foam will maintain its material properties when exposed to radiation dosage is BFN UFSAR Section 5.2.3.2 which states in part "… Irradiation tests have shown that no change in the resilient characteristics will take place for exposures up to 10^8R." This is in accordance with BFN's current licensing basis. Additionally, this same information is presented in Section 4.7.4, "Summary Description," of the LRA.