January 31, 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 – CIVIL SECTION RELATING TO AGING MANAGEMENT REVIEWS AND PROGRAMS FOR CONTAINMENTS, STRUCTURES, AND COMPONENET SUPPORTS FOR SECTIONS 3.5, 4.7.4, AND B.2.1.32 – RESPONSE TO NRC REQUEST FOR ADDITONAL INFORMATION (RAI) (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 license renewal application, the NRC staff, by letter dated December 10, 2004, identified an area where additional information is needed to complete its review.

The specific area requiring a request for additional information (RAI) is related to the aging management reviews and programs for containments, structures, and component supports for Sections 3.5, 4.7.4, and B.2.1.32 respectively.

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

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 31st day of January, 2005.

Sincerely,

Original signed by:

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Manager of Licensing
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s:/Licensing/Lic/BFN LR Civil Sections 3.5, 4.7, and B.2.1.32 RAI response

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), RELATED TO THE LRA SECTIONS 3.5, 4.7.4, AND B.2.1.32

(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), RELATED TO THE LRA SECTIONS 3.5, 4.7.4, AND B.2.1.32

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The specific area requiring a request for additional information (RAI) is related to the aging management reviews and programs for containments, structures, and component supports for Sections 3.5, 4.7.4, and B.2.1.32 respectively.

Listed below are the specific NRC requests for additional information and the corresponding TVA responses.

NRC RAI 3.5-1:

Item Numbers 3.5.1-3 and 3.5.1-17 (Table 3.5.1) of the LRA, states that the BFN aging management review (AMR) results are consistent with NUREG-1801 with the exceptions described in aging management program (AMP) B.2.1.32. NUREG-1801 under Item B.1.1.1-d (page II B1.5) recommends further evaluation regarding the stress corrosion cracking of containment bellows. In discussion of these items in Section 3.5.2.2.1.7 of the LRA, the applicant asserts that Appendix J, Type B testing is effective in detecting leakages through the vent line bellows, as well as, through other pressure boundary bellows. Please provide additional information regarding the frequency of Type B testing (performance based interval, in accordance with Option B, Appendix J) of containment pressure boundary bellows at BFN Units 2 and 3, and status of these bellows for BFN Unit 1.

TVA Response to NRC RAI 3.5-1:

Section 3.5.2.2.1.7 of the LRA states:

> "The type of bellows used on the BFN containment penetrations are not the type described in NRC IN 92-20 (Quad Cities Station Unit 1). The vent line bellows at BFN are a single ply bellow design. Pipe penetration bellows for high energy lines are two ply bellows with a mesh. The design of BFN penetration bellows allows full pressure to be transmitted to all portions of the bellows during Appendix J testing. The BFN containment penetrations bellows are not susceptible to a failure of the 10 CFR 50, Appendix J LLRT test to detect cracking as described in NRC IN 92-20."

BFN pipe penetration bellows are 10 CFR 50, Appendix J, Type B tested. BFN vent line bellows are 10 CFR 50, Appendix J, Type A tested.

Type B and C tests are performed prior to initial reactor operation. Subsequent Type B and C tests are performed at a frequency of at least once per 30 months until performance data are collected for evaluation for extended test interval in accordance with RG 1.163. Type B tests may use an extended interval of up to 120 months (excluding airlocks). Unit 2 and 3 bellows are tested at a 60-month test interval. There have been no bellows failures on either Unit 2 or 3 bellows. Prior to the restart of Unit 1, Appendix J, Type B testing of containment pipe penetration bellows will be performed. Unit 1 bellows will be tested at least once per 30 months until test performance data is available to justify an extended test interval under Option B.

NRC RAI 3.5-2:

For seals and gaskets related to containment penetrations, in Item Number 3.5.1-6 of Table 3.5.1, and Component Type, "Compressible Joints and Seals" in Table 3.5.2.1; the containment inservice inspection (ISI) (AMP B.2.1.32) and the containment leak rate testing (AMP B.2.1.34) have been identified as the aging management programs. Based on Exception 1 in AMP B.2.1.32, the AMP will not be applicable for aging management of containment seals and gaskets. For equipment hatches and air-locks at BFN, the approach is that the leak rate testing program will monitor aging degradation of seals and gaskets, as they are leak rate tested after each opening. Clarify whether these assumptions are correct. For other penetrations (mechanical and electrical) with seals and gaskets, provide information regarding the adequacy of Type B leak rate testing frequency to monitor aging degradation of seals and gaskets at BFN containment drywell. Please provide the status of seals and gaskets of these penetrations at BFN Unit 1.

TVA Response to NRC RAI 3.5-2:

ASME Section XI, 1992 Edition, 1992 Addenda, Category E-D, Item Numbers E5.10 (Seals), and E5.20 (Gaskets) requires a visual examination, VT-3, of containment seals and gaskets. Examination of most seals and gaskets requires the joints to be disassembled. When the airlocks, hatches, electrical penetrations, and flanged connections are tested in accordance with 10 CFR 50, Appendix J, degradation of the seal or gasket material would be revealed by an increase in the leakage rate. Corrective measures would be applied and the component re-tested.

For Units 1, 2 and 3, Relief Request CISI-1 was granted to perform Appendix J test in lieu of the visual examination, VT-3, on the containment seals and gaskets. The moisture barriers continue to receive a visual VT-3 examination in accordance with Category E-D for Units 1, 2, and 3.

The scope of the 10 CFR 50, Appendix J program includes all pressure-retaining components, the containment shell (drywell and torus) and penetrations. The following components are included in the scope of the program:

- containment penetration seals on airlocks, hatches, spare penetrations with flange connections, electrical penetrations and other devices required to assure containment leak-tight integrity;
- containment penetration gaskets on airlocks, spare penetrations with flange connections, and other devices required to assure containment leak-tight integrity;
- pressure retaining bolted connections;
- containment penetration bellows; and
- airlocks.

Units 2 and 3 O-ring seals (flanges, hatches, etc.) are tested on either a 30 or 60-month interval. Seal failures have occurred sporadically since restart. The Unit 2 and Unit 3 drywell heads have experienced failures and are currently classified as Maintenance Rule (a)(1) for corrective actions.

There are currently no electrical penetration performance problems on Unit 2. All electrical penetrations on Unit 2 are currently on a 120-month test interval. Testing has identified only minor problems such as gauge, tubing, and root valve leaks. Unit 3 electrical penetrations are on 30, 60, or 72-month test intervals. In general, testing has identified only minor problems such as gauge, tubing, and root valve leaks. However, one electrical penetration (3-EPEN-100-0101C) on Unit 3 experienced a failure, was repaired, and is being tested on a 30-month test interval. Other electrical penetrations are being tested at a 60-month interval. The remainder of the Unit 3 electrical penetrations are on a 72-month interval.

Type B testing will be performed as part of the Unit 1 restart effort and will continue at least once per 30 months until test performance data is available to justify an extended test interval under Option B.

NRC RAI 3.5-3:

Containment drywell-head to drywell joint consists of a pressure unseating containment boundary with pre-loaded bolts. Loosened bolts and deteriorated gasket and/or seal can breach containment pressure boundary. Exceptions 1 and 2, taken in the containment ISI program (AMP B.2.1.32) will preclude examinations of seals and bolts of this joint. Only Type A leak rate testing and associated visual examination requirements of Appendix J program (AMP B.2.1.34) can be relied upon to detect defects and degradation of this joint. The test interval for Type A leak rate testing can be 10 to 15 years. Provide information regarding the plans and programs that are used to ensure the integrity of this joint for each containment. Please provide the status of the components (O-rings and bolts) at this joint for BFN Unit 1.

TVA Response to NRC RAI 3.5-3:

These containment pressure boundary components will continue to be inspected consistent with the Browns Ferry CLB for 10 CFR Part 50, Appendix J requirements. On Units 2 and Unit 3 the Type A test frequency is currently on a 10-year interval. There have been no performance based Type A test failures on Unit 2 or Unit 3. A Type A Integrated Leak Rate Test will be performed as part of the Unit 1 restart effort.

Type B testing is also performed on the drywell-head seal every refueling outage for all three units. Therefore, in combination of the Type A tests and Type B tests, integrity for this joint for each containment is assured.

Exception 2 pertains to bolt torque or tension testing. Pressure retaining bolting associated with the Containment drywell-head to drywell joint is examined in accordance with ASME Section XI, Subsection IWE.

NRC RAI 3.5-4:

Item 3.5.1-12 in Section 3.5.2.2.1.4, water leakages from the sand drains have been found in BFN Units 2 and 3, and the results of the UT examinations performed from the accessible areas of the drywells indicated that the condition of the drywell shells are good, and these areas did not require augmented examination. Provide the following additional information related to the drywell shell corrosion in this area for each BFN containment drywell:

a. In other Mark 1 containments, the cause of water leakage from the sand bed drains has been found as the water leaking from refueling cavity (see IN 86-99, "Degradation of Steel Containments)." As no water leakage has been indicated from BFN Unit 1 (having no refueling activities during its long lay up), it would appear that the cause of the water leakage in Units 2 and 3 could be the same as that described in the information notice. Please provide a discussion of the root cause in this context.

b. If the water leakage is related to refueling operation, please provide information regarding the corrosion susceptibility of the cylindrical part of the drywell shell on the insulation (inaccessible) side.

c. Item No. E4.12 of Examination Category E-C of Subsection IWE requires the owner to establish grid and measurement locations in the suspect areas identified for augmented examinations. Please provide information regarding the methods used to establish confidence level that no drywell shell corrosion exists in the sand-pocket areas.

d. Unless preventive actions are taken and conditions verified, that no leakage and shell corrosion exists in the suspect areas, IWE will require continuation of UT measurements in the augmented examination areas. Please provide justification for excluding the suspect areas from augmented examinations.

e. Based on the results of the UT examinations performed from the accessible areas of the drywells, BFN asserted that the condition of the drywell shells is good. Provide a discussion of BFN's criteria for judging that the condition of the drywell steel liner plate is good and the rationale for the criteria.

f. Provide a discussion of any degradation observed and/or repair work implemented as a result of past general visual inspection of the moisture barrier located at the junction of the steel drywell and the concrete floor.

TVA Response to NRC RAI 3.5-4:

a. See response to item "b."

b. A postulated failure of the drywell-to-reactor building refueling seal can result in water intrusion into the annulus space around the drywell. This leakage can occur only during refueling outages when the reactor cavity is flooded to allow movement of fuel between the reactor and the fuel pool. However, water intrusion does not cause failure of the drywell's intended function. Any water leakage resulting from a postulated failure of the drywell-to-reactor building refueling seal could not remain suspended in the annulus region for an indefinite period of time and would eventually be routed to the sandpocket area drains or would evaporate due to the heat generated in the drywell during operation.

In TVA's response to NRC Generic Letter 87-05 dated August 30, 1988, which addressed the potential for corrosion of boiling water reactor (BWR) Mark I steel drywells in the "sand pocket region," TVA provided the NRC with the results of the ultrasonic testing for corrosion degradation of drywell liner plate. The results of the ultrasonic testing states: Each unit's drywell was ultrasonically tested near the sand cushion area during 1987. The results from these tests showed that the nominal thickness was maintained on each drywell. Below are the results of each unit's drywell ultrasonic testing:

- Unit 1 - No reading below the nominal thickness of one inch was measured indicating that the integrity of the drywell liner plate is maintained. Periodic leakage from the sand cushion area has been observed. Corrosive species in the drainage are bases to suspect a higher rate of corrosion on Unit 1 drywell liner plate than on Unit 2 and 3. However, objective evidence of serious corrosion damage was not noted.

- Unit 2 - No reading below the nominal thickness of one inch was measured indicating that no damage to the integrity of the drywell liner plate has occurred.

- Unit 3 - No reading below the nominal thickness of one inch was measured indicating that no damage to the integrity of the drywell liner plate has occurred.

c. In response to NRC Generic Letter 87-05, TVA provided the NRC with the results of the ultrasonic testing for corrosion degradation of BFN Units 1, 2, and 3 drywell liner plates near the sand cushion area during 1987. The results from these tests showed that the nominal thickness was maintained on each drywell. Paragraph IWE-1242 of ASME Section XI requires the Owner to determine containment surface areas requiring augmented examination, in accordance with Paragraph IWE-1241. UT thickness measurements of this area were obtained during the U2C10 and U3C8 refueling outages for Units 2 and 3 respectively and in 1999 and 2002 for Unit 1 (0-TI-376 Appendix 9.7 page 4). The data indicates that the condition of the drywell steel liner plate in this area meets code requirements, and that this area should not be categorized for augmented examination.

d. See response to Item c.

e. See response to Item c.

f. The internal drywell steel containment vessel (SCV) embedment zone is subject to corrosion if the drywell floor-to-containment vessel moisture barrier fails, allowing moisture intrusion, or if the concrete floor of the drywell cracks, allowing moisture seepage through to the steel liner. During the Unit 2 Cycle 9 outage, a portion of the moisture barrier was replaced (Problem Evaluation Report (PER) BFPER971516). Engineering personnel performed an examination of the exposed drywell SCV area below the moisture seal. This inspection indicated some minor pitting and localized rust, but nothing approximating a challenge to nominal wall thickness. No propagation of iron oxide to the concrete surface was noted, which would be indicative of steel containment vessel corrosion below the concrete. Inspections conducted by the Containment ISI program during Unit 2 Cycle 10 refueling outage and Unit 3 Cycle 9 refueling outage also identified some damaged areas of the moisture barrier (gaps, cracks, low areas/spots, or other surface irregularities) that were evaluated by engineering and replaced or repaired. (PER 99-005254-000 for Unit 2 Drywell moisture seal barrier and PER 00-004163-000 for Unit 3 Drywell moisture seal barrier).

In Unit 1, the moisture barrier in areas that would be made inaccessible due to ductwork installation have been replaced. Visual examination of exposed drywell SCV area below the moisture barrier identified some minor pitting. Ultrasonic thickness and pit depth measurements were taken and evaluated by engineering which confirmed nominal wall thickness was not encroached. The entire Unit 1 moisture barrier will be replaced before restart.

The Structures Monitoring Program also monitors the concrete to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell. Research of plant history did not reveal any instances of water spills and water ponding on the containment concrete floor. A general visual inspection of the moisture barrier at the junction of the steel drywell shell and the concrete floor is performed once each inspection interval in accordance with the ASME Section XI, Subsection IWE aging management program.

NRC RAI 3.5-5:

A number of load bearing reinforced concrete structures within the drywell shell are subjected to temperatures higher than the established threshold of 150 degrees F, as discussed in Item 8 of Section 3.5.2.2.2.1 of the LRA. The effectiveness of the closed cooling ventilation system is paramount in preventing large temperature excursions in the drywells. Provide the following information related to the concrete structures within the drywells of each Unit:

a. Provide a summary of the operating experience related to the reliability of the closed cooling ventilation system.

b. Provide a summary of the results of the last inspections performed on (1) reactor pressure vessel (RPV) pedestal supports, (2) the foundation and floor slab, and (3) the sacrificial shield wall under the existing Structural Monitoring Program.

c. Item 8 of Section 3.5.2.2.2.1 of the LRA (page 3.5-44) states that the main steam tunnels in the Reactor Building at BFN Units 1, 2, and 3 have a maximum normal space ambient temperature of 160 degrees F. Provide a discussion of BFN's basis, including a summary of the results of the engineering analysis performed, to support BFN's conclusion that the conditions identified in NUREG-1801 are satisfied and that aging management of reduction of strength and modulus due to elevated temperature for the affected concrete components is not required.

TVA Response to NRC 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°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.

NRC RAI 3.5-6:

Table 3.5.2.26, "Structures and Component Supports," is silent on the AMR related to Class MC supports. AMP B.1.33 of the LRA takes exception to NUREG-1801, Section XI.S3, and states that the aging effects for supports of MC components will be managed by the Structures Monitoring Program (B.2.1.36), or Water Chemistry Program (B.2.1.5) with associated One-Time Inspection Program (B.2.1.29) for submerged supports during the extended period of operation. Provide the following information related to the aging management of Class MC supports:

a. Please provide the results of the aging management review for (1) MC component supports within the BFN containments, (2) MC component supports outside the containments, and (3) supports for piping penetrating through the containments and designated as MC piping (if any). Also, summarize the program (sample size, inspection frequency, personnel qualification, etc.) used to arrive at the AMR results.

b. 10 CFR 50.55a(g)(4) requires the inservice inspections of Class MC pressure retaining components, and their integral attachments in accordance with the requirements of Section XI of the ASME Code. Subsection IWF of Section XI requires the inservice inspection requirements for Class 1, Class 2, Class 3, and Class MC supports. Figure IWF-1300-1 provides illustrations of typical examination boundaries. Item F1.40 of Examination Category F-A of Subsection IWF, sets the examination requirements for Class MC supports, other than those for the MC piping supports. Please provide justification for the exception taken in AMP B.2.1.33 regarding the aging management of Class MC component supports.

c. As Subsections IWE and IWF do not incorporate explicit requirements for inservice inspection of supports of pipes designated as Class MC, please provide a description of a proposed aging management program (could be part of the Structural Monitoring Program), including sample size, the extent of examination, frequency of examination, and qualification of personnel who perform and evaluate the inspection results.

TVA Response to NRC RAI 3.5-6:

This RAI response is addressed in the response to the following RAIs: 2.4-2, 2.4-13(a) & (b) and B.2.1.33. Refer to the TVA response to NRC dated January 24, 2005.

NRC RAI 3.5-7:

Under "Buried" environment of Table 3.0.2, "External Service Environments" of the LRA states that ground water at BFN is non-aggressive. Provide historical BFN site ground water chemistry test results together with a discussion of the extent of past ground water sampling and testing frequency as well as the extent of fluctuation of the test results to support the above assertion.

TVA Response to NRC RAI 3.5-7:

Since BFN did not have data available from the construction period or since plant start-up, baseline sampling was performed over the past year of groundwater and the Wheeler Reservoir. The baseline sampling was to establish if BFN had aggressive or non-aggressive water as defined by the following criteria: pH <5.5, Chlorides > 500 ppm and Sulfates > 1500 ppm. The samples were taken at intervals to take into consideration seasonal variations. The samples were taken from the existing site radiological monitoring wells and from the Wheeler Reservoir in close proximity to the Intake Pumping Station structure. Samples were taken at various depths in the monitoring well and the Reservoir by the site environment staff and analyzed by an off-site laboratory for the site environment group.

Results of Browns Ferry groundwater and Wheeler Reservoir water sampling are as follows:

a. Groundwater:

- pH ranges from 6.33 to 8.77 which are well above <5.5 (Note in the well that the value 6.33 was obtained, the remaining pH readings ranged from 7.16 to 7.60 during the time period of sampling. Only one other well had a pH value below 7 and its pH was 6.92 with the remaining readings ranging between 7.12 and 7.6.)

- Chlorides – maximum reading of 18.3 ppm which is well below the threshold of 500 ppm

- Sulfates – maximum reading of 30.3 ppm which is well below the threshold of 1500 ppm

b. Wheeler Reservoir:

- pH ranges from 7.28 to 8.64 which are well above <5.5.

- Chlorides – maximum reading of 13.9 ppm which is well below the threshold of 500 ppm.

- Sulfates – maximum reading of 15.5 ppm which is well below the threshold of 1500 ppm.

Browns Ferry groundwater water and Wheeler Reservoir sample measurements have confirmed that parameters are well below threshold limits that could cause concrete degradation (an aggressive environment does not exist).

NRC RAI 3.5-8:

The AMR discussion provided in Section 3.5.2.2.2.2, aging management of inaccessible areas of the LRA (page 3.5-45) is rather general and brief, and requires more detailed elaboration to support BFN's conclusion that the conditions identified in

NUREG-1801 as revised by ISG-03 are satisfied and no aging management for below grade inaccessible BFN concrete is needed. Provide additional BFN specific information to justify the above conclusion including: (1) concrete quality and test data for BFN inaccessible concrete, (2) past operating experience regarding exposure of BFN's inaccessible concrete to aggressive chemical/fluid environment, and (3) past inaccessible concrete inspection finding and data related to concrete degradation and repairs.

TVA Response to NRC RAI 3.5-8:

1. The BFN concrete structures and concrete components are designed in accordance with ACI 318-63 and 71 and constructed using ingredients conforming to ACI and ASTM standards, which provide for a good quality, dense, well cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing bars.

   Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77.

2. As noted in the response to RAI 3.5-7, Browns Ferry groundwater water and Wheeler Reservoir sample measurements have confirmed that parameters are well below threshold limits that could cause concrete degradation (an aggressive environment does not exist).

3. A review of Browns Ferry operating history, the 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 when below grade inaccessible concrete was excavated for other reasons.

NRC RAI 3.5-9:

With respect to the first and last items of Table 3.5.2.2 of the LRA (page 3.5-63), no aging effect requiring management and aging management program are identified for hatches/plugs, and electrical and I&C penetrations made of carbon and low alloy steel, respectively, that are embedded or encased in concrete, whereas, Item III.A2.2-a (page III A2-10) of NUREG-1801 calls for designation of a structures monitoring program to manage the loss of material and corrosion aging effects for steel components exposed to various environments. Additionally, the mechanical penetrations listed in the fourth item of Table 2.5.2.2 of the LRA (page 3.5-64) and the structural steel beams, columns, plates, trusses listed in the last item of Table 3.5.2.2 of the LRA (page 3.5-66), that are embedded or encased in concrete, are also identified as having no applicable aging effect that requires aging management, and therefore, no AMP is designated for the commoners. This same BFN position is shown through out the remainder of Table 3.5.2 of the LRA. BFN is requested to discuss past operating experience and inspection results related to aging degradation of embedded or encased hatches, plugs, duct banks, manholes, mechanical penetrations, and electrical and I&C penetrations in order to provide an operating experience based rationale to justify its assertion that these components require no AMP to manage their aging.

TVA Response to NRC RAI 3.5-9:

The BFN concrete structures and concrete components are designed in accordance with ACI 318-63 and 71 and constructed using materials conforming to ACI and ASTM standards, which provide for a good quality, dense, well cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing bars.

Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77.

As a minimum, all exposed portions of embedded carbon steel structural components are inspected for the following aging effects:

- Outside Air Environments: Loss of material due to general and pitting corrosion
- Inside Air Environments: Loss of material due to general corrosion
- Containment Air Environments: Loss of material due to general corrosion

A review of Browns Ferry operating history, the 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 for carbon steel components embedded/encased in concrete.

NRC RAI 3.5-10:

Non-ferrous aluminum electrical and I&C penetrations embedded or encased in concrete are listed in the second item of Table 3.5.2.2 (page 3.5-64) of the LRA as components requiring no AMP to manage any aging effect. Provide a discussion of past BFN and applicable industry operating experience to justify this BFN AMR finding. Referring to embedded or encased stainless steel spent fuel pool liners listed in the fourth item of Table 3.5.2.2 (page 3.5-66); BFN is requested to discuss applicable BFN operating experience of these liners to justify its AMR results that no AMP is needed to manage any aging effect.

TVA Response to NRC RAI 3.5-10:

The BFN concrete structures and concrete components are designed in accordance with ACI 318-63 and 71 and constructed using materials conforming to ACI and ASTM standards, which provide for a good quality, dense, well cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing bars.

Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77.

Embedded or Encased Aluminum Response:
Aluminum is a reactive metal, but it develops an aluminum oxide film that protects it from further corrosion in an indoor environment. The specific aluminum alloy (6063-T42) used at BFN for conduit and raceways is resistant to general corrosion, pitting, and SCC during testing in outdoor, and saltwater environments.

For the aluminum that is embedded/encased within the concrete, corrosion is not considered an applicable aging mechanism. The concrete must first be degraded by other aging mechanisms, which reduce the protective cover and potentially allow for the intrusion of aggressive ions causing a reduction in concrete pH. Aging management of concrete aging effects will manage the corrosion of the embedded/encased aluminum's concrete protective cover.

A review of Browns Ferry operating history, the 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 for aluminum components embedded/encased in concrete.

Embedded or Encased Stainless Steel Response:
For the stainless steel that is embedded/encased within the concrete, corrosion is similarly not considered an applicable aging mechanism. The concrete must first be degraded by other aging mechanisms, which reduce the protective cover and allow for the intrusion of aggressive ions causing a reduction in concrete pH. Adequate management of other concrete aging effects will in effect manage the aging of the embedded/encased stainless steel.

After a review of the Browns Ferry operating history, the 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 for stainless steel that is embedded/encased within concrete.

Operating history did show a small leak in the Unit 1 fuel pool liner. The Unit 1 fuel pool has remained in service during the extended outage since spent fuel is stored in the pool. This leak in the Unit 1 fuel pool was documented in accordance with the site's corrective action program, SPP-3.1, TVAN Standard Program and Processes, "Corrective Action Program" as PER 00-011982-000 (electronic corrective action program number 35486. This leak is contained within the leak channel beneath the fuel pool liner. The fuel pool liners are monitored on a monthly basis per operation instruction 1-OI-78. The leak is small (~0.06 gpm) and has been steady over time without an increasing trend over the last ten years.

NRC RAI 3.5-11:

With respect to the fire barriers consisting of ceramic fiber listed in Table 3.5.2.5 (page 3.5-74) of the LRA, BFN's AMR identified neither aging effect requiring management nor AMP for the ceramic fiber fire barriers. Discuss BFN's past plant specific inspection results of these fire barriers in order to provide an operating experience based justification for the above AMR finding.

TVA Response to NRC RAI 3.5-11:

This same RAI was asked as RAI 3.3-2 for the Reactor Building. In the response to that RAI, the same material was also addressed for the Diesel Generator Building (Table 3.5.2.5, item number 10 on page 3.5-74). Refer to the TVA response to RAI 3.3-2 (TVA letter to NRC dated September 30, 2004).

NRC RAI 3.5-12:

Non-ferrous aluminum conduit and supports, that are exposed to outside air, are listed in Table 3.5.2.26 (page 3.5-140) as components having no applicable aging effect requiring management; thus, no AMP is designated to manage their aging. Depending on the severity of the outside air environment to which the components are consistently exposed, some aluminum conduit and supports may experience loss of material aging effect. Discuss BFN's past plant specific inspection results of these supports in order to provide an operating experience based justification for the above AMR finding.

TVA Response to NRC RAI 3.5-12:

The following aluminum components in an outside air environment are identified:

- electrical and I&C penetrations,
- conduits and supports, and
- non-ASME equivalent supports.

Aluminum alloys containing zinc are susceptible to corrosion in wetted aggressive environments. However, the outside air environment does not contain contaminants that would cause an aggressive environment. In addition, the aluminum conduit and conduit supports at BFN are also constructed of 6063-T42 alloy that is resistant to pitting, crevice corrosion, and SCC [Ref. Metals Handbook, Ninth Edition, Volume 13, "Corrosion," ASM International, 1987]. Since the potential for concentration of contaminates is not significant, and the specific aluminum grade used at BFN in an outside air environment is more resistant to corrosion, loss of function due to corrosion is not considered plausible.

A review of Browns Ferry operating history, the 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 for the following aluminum components:

- electrical and I&C penetrations,
- conduits and supports, and
- non-ASME equivalent supports

NRC RAI 3.5-13:

Table 3.5.2.26 (page 3.5-141) of the LRA lists equipment supports and foundations made of non-ferrous lubrite that are exposed to inside air environment as components having no aging effect requiring management and therefore, no AMP is designated for the components. Table III.B1.1.3-a (page III B1-4) of NUREG-1801 identifies loss of mechanical function, corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads, and elastomer hardening as potentially applicable aging effects for the lubrite components, and designates ASME Section XI, Subsection IWF as the AMP to manage the listed aging effects. Discuss BFN's past plant specific inspection and maintenance results of these lubrite supports in order to provide an operating experience based justification for the LRA assessment.

TVA Response to NRC RAI 3.5-13:

The Table 3.5.2.26 entry applies to the lubrite plates used for the Core Spray and RHR pump equipment support plates. EPRI report 1002950, "Aging Effects for Structures and Structural Components (Structural Tools), Revision 1", states that Lubrite material resists deformation, has a low coefficient of friction, resists softening at elevated temperatures, absorbs grit and abrasive particles, is not susceptible to corrosion, withstands high intensities of radiation, and will not score or mar. Lubrite products are solid, permanent, completely self lubricating, and require no maintenance. The Browns Ferry reactor building environment at the location of the Core Spray and RHR pump equipment support plates is not an aggressive or wetted environment.

A search of Browns Ferry and industry operating experience did not identify any instances of Lubrite plate degradation or failure to perform its intended function due to aging effects.

NUREG-1759, "Safety Evaluation Report Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4" and NUREG-1769, "Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3", concur that there are no aging effects for Lubrite plate that require aging management.

NRC RAI 3.5-14

With respect to the neutron absorbing sheets in spent fuel storage racks discussed in Item 3.3.1.10 of Table 3.3.1 of the LRA, BFN referred to Section 3.3.2.2.10 of the LRA for its further evaluation. Section 3.3.2.2.10 states that the Chemistry Control Program manages general corrosion and a one-time inspection of boral coupon test specimens was performed at BFN that confirmed no significant aging degradation had occurred and the neutron absorbing capacity of the boral had not been reduced. Since it is implied that some boral aging degradations had occurred at the time of inspection of the test specimens, please discuss the basis for BFN's above assertion that the neutron absorbing capacity of the boral will be maintained at an adequate level during the extended period of plant operation.

TVA Response to NRC RAI 3.5-14:

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.

NRC RAI 4.7.4-1:

Table 3.5.2.2 of the LRA lists the AMR results of expansion joint (elastomer, polyurethane foam) as TLAA and refers the TLAA to Section 4.7 of the LRA. Section 4.7.4, Radiation Degradation of Drywell Expansion Gap Foam states that an analysis of the effect of dose on the foam showed the material properties will remain within the limits assumed by the original design analysis for the additional 20 years of extended operation. Provide a more detailed discussion of the analysis including a discussion of the method and assumptions adopted in the analysis, the type of data extrapolation applied and the quantitative results obtained to justify the applicant's assertion that the requirements of 10 CFR 54.21(c)(1)(i) are fully met.

TVA Response to NRC RAI 4.7.4-1:

The TLAA analysis determines that the total dose to the polyurethane foam located between the drywell steel and the reactor building concrete will result in a 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.

The analysis model consists of the standard geometry sphere with a steel clad of 0.825 inches (drywell steel thickness). The radius of the sphere is 33.5 feet. Computer code QAD-P5Z, which is a point kernel variation of QAD-P5F, was used to determine dose and/or exposure rates. The computer code PARINT integrated the dose rates over time. The principle gamma source from normal operation is N-16; therefore the photon spectrum for normal operation is for N-16 with an arbitrary 1 Ci activity as input. The resultant dose rate was then scaled to the appropriate power level. The STP computer code determined the time dependent photon spectra. STP is the standard TVAN computer code for source term development. Gamma and neutron attenuation are considered.

Actual power conditions are utilized in the TLAA analysis. This applies for roughly the first 25% of plant life during which time each unit was down for a significant amount of time. For conservatism, it is assumed that EPU starts October 24, 2003, even though Unit 1 has yet to be restarted. Prior to October 24, 2003, Units 2 and 3 are at 105% (uprate) conditions. For an additional conservatism, Permali neutron shielding has not been included in the TLAA analysis.

The foam will only receive the significant dose from the drywell. The drywell is surrounded by a minimum of 5 feet of concrete. It is clear that the drywell sources will have a greater impact than any sources in the reactor building. The reactor building source impact will be negligible compared to the drywell.

The maximum dose for 60 year operation at EPU conditions without Permali neutron shielding occurs for Unit 2 and is 9.92E+07 which is less than a total dose of 1.0E08 rads used in the original analysis. Therefore, the material properties of the polyurethane foam will remain within the limits assumed by the original analysis.

NRC RAI B.2.1.32-1:

NRC Information Notice 88-82, "Torus Shells with Corrosion and Degraded Coatings in BWR Containments," describes and discusses the problems associated with corrosion of torus shells. Provide information regarding the operating experience and inspection of torus shells at the three Units of BFN. As the quality of torus water in the BFN Unit 1 torus may not have been monitored during its long lay up period, provide additional discussion of the condition of the torus for this Unit.

TVA Response to NRC RAI B.2.1.32-1:

The torus interior surfaces, at the waterline, are subject to corrosion due to moisture and repeated wetting and drying in the waterline region. Accessible portions of the torus inside surface are inspected each refueling outage. UT thickness measurements taken in torus underwater areas of both Units 2 and 3, revealed no evidence of excessive degradation (all readings were within 10% of nominal wall thickness). Previous inspections have documented evidence of minor coating degradation at the waterline region. Based on the above, it is concluded that the underwater region of the torus has not been subjected to accelerated degradation. Since evidence of repeated loss of coatings has been documented in the waterline region, augmented examination of this area is warranted, as a conservative measure on Units 2 and 3(0-TI-376 Rev. 4, Appendix 9.7).

Unit 1

During its lay-up period, the water in the Unit 1 torus (pressure suppression pool) was maintained by CI-13.1 "Chemistry Program" (Appendix A Table 20). Sampling frequency was quarterly. The torus was drained in the summer of 2003 for coating repair to be completed as a part of the Unit 1 recovery effort.

A VT-3, visual examination, was performed on the Unit 1 torus in August 2003. This examination included 100% of the Code Class MC boundary inside the torus (shell, ring girders, etc.) and both sides of the vent system (main vent line, vent header and downcomers). The visual examination found light to medium rust or discoloration in several areas and heavy rust in smaller, less frequent areas. There were also some instances of base metal encroachment, such as gouges, scratches, and tool marks. Engineering evaluation of the examination results determined the torus structural condition was acceptable as is with no base metal repairs required.

The requirements of ASME Section XI Inservice Inspection Subsection IWE, 1992 Edition with the 1992 Addenda will be implemented on Unit 1. Type A, B, and C leak rate testing required by 10 CFR 50, Appendix J will also be performed prior to Unit 1 restart.