November 20, 2006

10 CFR 50.55a

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 No. 50-296
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 3 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI, INSERVICE INSPECTION PROGRAM FOR THE THIRD TEN-YEAR INSPECTION INTERVAL - REQUEST FOR RELIEF 3-ISI-21, RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) NUMBER 2 (TAC NO. MC8795)

TVA submitted, by letter dated October 19, 2005, its Third Ten-Year Inservice Inspection (ISI) and System Pressure Test (SPT) Programs for Unit 3 of the Browns Ferry Nuclear Plant. The Code of record for the Third Ten-Year Interval ISI and SPT Programs is the 2001 Edition, 2003 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI. The Third Ten-Year Interval began on November 19, 2005.

During its review of the BFN Unit 3, Third Ten-Year Interval, Inservice Inspection Program, the NRC staff identified questions, by letter dated September 15, 2006, regarding BFN Unit 3 request for relief 3-ISI-21. Request for relief 3-ISI-21 addresses TVA’s proposed risk informed inservice inspection (RI-ISI) program. TVA provided its response to the staff’s request for additional information by letter dated October 11, 2006. The NRC staff identified follow up questions regarding TVA’s request. These questions were discussed in a teleconference on November 2, 2006. As a result, TVA is providing responses to the NRC questions in the enclosure to this letter.

If you have any questions, please contact me at (256) 729-2636.

Sincerely,

Original signed by:

William D. Crouch
Manager of Licensing
and Industry Affairs

Enclosure
cc: See Page 3

Enclosure
cc(Enclosure):

Mr. Malcolm T. Widmann, 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 Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

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

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

DTL:JWD:BAB
Enclosure
cc (Enclosure):

B. M. Aukland, POB 2C-BFN
M. Bajestani, NAB, 1A-BFN
A. S. Bhatnagar, LP 6A-C
R. H. Bryan, BR 4X-C
Samuel Flood, CMB 1B-BFN
R. G. Jones, POB 2C-BFN
G. V. Little, NAB 1D-BFN
R. A. DeLong, SAB 1A-BFN
B. J. O'Grady, PAB 1E-BFN
K. W. Singer, LP 6A-C
P. D. Swafford, LP 6A-C
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)
UNIT 3
AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME)
SECTION XI, INSERVICE INSPECTION (ISI) PROGRAM,
THIRD TEN-YEAR INSPECTION INTERVAL

REQUEST FOR RELIEF 3-ISI-21,
RISK INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI), NO. 2

(SEE ATTACHED)

ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 3
AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME)
SECTION XI, INSERVICE INSPECTION (ISI) PROGRAM,
THIRD TEN-YEAR INSPECTION INTERVAL

REQUEST FOR RELIEF 3-ISI-21,
RISK INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM,

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI), NO. 2

During its review of the BFN Unit 3, Third Ten-Year Interval Program, the NRC staff identified questions regarding request for relief 3-ISI-21. Request for relief 3-ISI-21 addresses TVA's proposed risk informed inservice inspection (RI-ISI) program. These questions were transmitted to TVA by NRC letter dated September 15, 2006. TVA provided its response to the staff's request for additional information by letter dated October 11, 2006. Subsequently, the NRC staff identified follow up questions. These questions were discussed in a teleconference on November 2, 2006. As a result, TVA is providing responses to the NRC follow up questions in this enclosure. Listed below are the specific NRC requests and the corresponding TVA responses.

NRC Request 9A

In reference to TVA's response to NRC question 3 in the September 15, 2006 RAI: Provide an accurate description of the TVA risk-ranking process associated with the placement of HSS segments into Regions 1 and 2, such that the apparent inconsistencies discussed in the November 2, 2006, conference call are resolved.

(1) If, in fact, some segments containing elements with a quantified failure rate (QFR) are placed in Region 2, please expand your previous RAI response to explain the process for risk ranking segments between Regions 1 and 2.

(2) If the NRC safety evaluation description (apparently confirmed in the November 2, 2006, conference call) is accurate, please revise your previous RAI response accordingly.

TVA Response to NRC Request 9A

As referenced in the last paragraph on page 15 of the February 11, 2000, Safety Evaluation Report, "The TVA method does not further divide HSS segments into two regions." Attachment B to the October 11, 2006 RAI response contained an erroneous reference on page E1-15 under Structural Element Selection. The third sentence should simply state: "At TVA, two methods were used to select elements." The words "in Region 2" should not have been included, and were an internal reminder of the practice in the WCAP for which the alternative was being presented.

NRC Request 9B

Specifically state the inspection selection criteria for elements, both for those with a QFR and for those with a zero failure rate (ZFR), and clarify your previous RAI response to remove any inconsistencies. This explanation should cover the following cases:

(1) All elements in the segment have a QFR.
(2) All elements in the segment have a ZFR.
(3) Some elements in the segment have a QFR and some have a ZFR.

TVA Response to NRC Request 9B

TVA utilizes a Quantified Element Selection Process. To perform Quantified Element Selection, risk is calculated by determining the failure rate for each individual element (which could be calculated to be zero) and multiplying that value by the conditional core damage probability CCDP (or CLERP) for the entire segment. RRW for each element with respect to the total CDF (or LERF) for segments is determined. Any element with RRW > 1.001 is selected for examination. This process will select all elements considered to have High Failure Importance.

Segments selected for defense-in-depth may not have any elements with quantified failure rates that result in RRW > 1.001. For these segments, a number of elements equivalent to current Section XI inspection requirements are selected. These elements are selected based on stress levels, failure rate, consistency with elements selected in similar segments, or previous examination history as a final consideration. At least one element in each HSS segment shall be examined.

Both the RRW HSS segments and the Defense-in-Depth HSS segments can contain a combination of elements with Quantified Failure

Rates and elements with Failure Rates quantified to be zero. The process described above and approved in the February 11, 2000 Safety Evaluation Report assures that appropriate elements are selected for examination even if the Failure Rate is quantified as zero. This selection is then confirmed by the Change in Risk Evaluation.

NRC Request 9C

The licensee's submittal dated October 19, 2005, on page 60 of 192, Section 7.12.2 and other locations states "Procedure 3-SI-4.6.G-A of this Program outlines an acceptable alternative approach to the existing Section XI requirements for the scope and frequency of inspection of the ISI Program..." Provide Procedure 3-SI-4.6.G-A or clarify this reference.

Section 7.12.4 states "The piping segments and inspection strategy (i.e., frequency, number of inspections, methods, or all three) are defined in Part 6 of Table R-A and 3-SI-4.6.G-A." Provide Part 6 of Table R-A and 3-SI-4.6.G-A or clarify these references.

TVA Response to NRC Request 9C

3-SI-4.6.G-A is the internal TVA procedure which documents the entire RI-ISI process. The appropriate portions of this procedure were extracted or summarized in the original template submittal to provide the information the NRC had decided was necessary to review the application.

The "Part 6 of Table R-A" is a typographical error and should read "Section 8.1 Part 5." Section 8.1 Part 5 is located on page 82 of 192 of TVA's October 19, 2005, Unit 3, Third Ten-Year Interval, ISI Program submittal and provides the RI-ISI examination schedule. This item has been documented in the BFN corrective action program.

NRC Request 10

In reference to TVA's response to NRC question 6: There appear to be several discrepancies within Attachments C, D and E of Enclosure 1 to your previous RAI response, as well as those tables relative to the table in Attachment 1 to your Relief Request 3-ISI-21, based on the process description discussed at the conference call on November 2, 2006. Also, there appear to be discrepancies between the tabular data from your original RI-ISI program submittal and the renewal. Explain what each table represents and correct or otherwise resolve all apparent discrepancies.

<u>TVA Response to NRC Request 10</u>

Specific NRC questions relative to the general question above were discussed in a teleconference between BFN and the NRC staff on November 2, 2006. The NRC questions are provided below with the corresponding TVA response. The NRC questions are shown in bold text followed by the TVA response in plain text.

Please confirm that, in the transition from the 2nd to the 3rd interval, no pipe segments were re-categorized from HSS to LSS.

<u>TVA RESPONSE</u>

No pipe segments were re-categorized from HSS to LSS.

It is not obvious why some of the HSS segments from the original Table 3.8-1 (and shown on Attachment C) were left off the Attachment D list (notably Recirc system segments 3-068-003 through 3-068-013, and 3-068-015), even if all of the contained elements' failure probabilities are 0.0. It is observed that segment 3-073-001, which ostensibly has at least one element with a quantified FP (due to non-zero CDF per Attachment C) is also not shown on Attachment D. For completeness it is suggested that all of the HSS segments be included in this list.

<u>TVA RESPONSE</u>

See response below.

There are elements from segment 3-068-001 and many other segments with FP = 0.0. Yet these were not included in the Attachment E list (i.e., the "ASME Section XI pool" from which 25% are selected). Referring to the reviewer's comment under "TVA Response to NRC Request 3", does this mean that the only elements from segments in which all elements have a FP = 0.0 go into the Attachment E list? If this is the case, why are 10 elements from segment 3-068-014 and 13 elements from segment 3-068-016, which do contain some elements with non-zero FPs, also included in the Attachment E list? (It is also noted that the elements from these segments in the Attachment E list are shown on Attachment D as "not selected")

<u>TVA RESPONSE</u>

See response below.

If it was intended to include all elements with FP = 0.0 from segments with RRW >1.001, but <1.005 into Attachment E, then why aren't those elements from segments 3-068-001 and 3-069-001 with FP = 0.0 listed in that attachment?

TVA RESPONSE

As explained in the response to 9B above, elements are selected either by quantified element RRW or by the Section XI criteria. The referenced attachments are divided on a segment basis, or divided by the method by which elements within the segment were selected, not by the failure rate of the individual element. The Quantified RRW process in Attachment D applies to those segments that contained elements with RRW ≥ 1.001. The process in Attachment E applies to those segments deemed HSS where no element RRW is above the qualification threshold. As such, Attachment E should have been titled "Element Selection (Segments with no Element RRW ≥ 1.001)," not "Element Selection (Zero Failure Rate)," since as can be seen there can be elements with a failure rate quantified as zero in either category. Based on these categorizations, each segment appears in the appropriate Attachment with its corresponding selected elements.

Without any knowledge of your weld identification nomenclature, it is observed that the same weld identifier is listed in Attachment E for segments 3-068-003 and 3-068-015. Is this the same weld, or can two different segments have the same weld identifier?

TVA RESPONSE

This was a typographical error. The element selected for 3-068-015 should be weld identifier RWR-3-002-G019. Another typographical error has been detected. The element selected for 3-068-006 should be weld identifier RWR-3-001-G015. These items have been documented in the BFN corrective action program.

Element RWCU-3-001-G011 of Segment 3-069-001, previously in Table 3.8-1 of TVA's original submittal, is not shown in Attachment D. Please explain the reason for not including it. If it still exists, please re-include into the table with its current status.

TVA RESPONSE

During development of the BFN Unit 3 Third Ten-Year Interval RI-ISI Program, re-evaluation of the piping isometrics indicated that weld RWCU-3-001-G011 was more appropriately assigned to segment 3-069-002. Even if it had been assigned to segment 3-069-001, it had a failure rate quantified as zero under current conditions and would have had no effect on the element selection process.

According to Attachment 1 of your Request for Relief 3-ISI-1, in addition to the 15 locations associated with FAC in the MS and FW systems, there are now 17 examinations for thermal fatigue and 39 for IGSCC within the scope of the RI-ISI programs. This represents a total of 71 examinations. (The list of examinations in Attachment A also contains 71 of them.) Attachment D herein shows a total of 33 elements selected for examination. Attachment E adds another 21 elements. This indicates a total of 54 examinations. Please explain and resolve the apparent discrepancy between Attachment D/E and the other sources showing the total count of examinations.

TVA RESPONSE

It is assumed that the above reference is to Request for Relief 3-ISI-21 rather than 3-ISI-1. There is one typographical error in Attachment 1. For system 074 (RHR) there should be 7 elements, rather than 8 elements, selected for IGSCC Category C inspection, resulting in a total of 38 IGSCC examinations. This issue has been documented in the BFN corrective action program. This leaves a difference of 16 elements. In TVA's January 18, 2000, RAI response for the original (i.e., Second Ten-Year RI-ISI Interval) program submittal, TVA agreed to add an additional 16 elements to provide additional coverage for defense-in-depth outside of the HSS segments. As such, these welds would not appear in the element selection tables presented in Attachments D and E, but are included in Attachment 1 of 3-ISI-21.

Pertinent to the licensee's statement "In most cases for segments influenced by the hydrogen water chemistry/noble metal injection program, the failure rates went to zero and the segments were no longer HSS based on RRW." Please explain why there are many segments (mostly main steam and feedwater) with a quantified CDF, but a LERF value of 0.0. (Presumably this is not due to a "zero failure rate"; perhaps there is a LERF truncation value?).

TVA RESPONSE

If a pipe failure rate went to "zero," the computed CDF and LERF values will always be "Zero" since the "pipe failure rate" is a multiplication factor in both cases. For some initiators (e.g. LLO - Other Large LOCA) the Conditional Large Early Release Probability (CLERP) is computed to be "zero" since the LERF is "zero" (the LERF value has been truncated to zero for "LLO" for Unit 3). However, the conditional core damage probability (CCDP) for the same initiator (LLO) is not zero, since the initiator does result in core damage. Examples of these cases are Main Steam segments 36, 37, 38, 39 and Feedwater segments 6 and 7.

NRC Request 11

In reference to TVA's response to NRC question 7: During the conference call on November 2, 2006, reference was made to the "2002 Unit 3 PRA model." Provide the date, revision designator, base CDF, and LERF of the PRA model used to perform the analysis (e.g. - CCDP calculations) for the proposed third interval RI-ISI program.

TVA Response to NRC Request 11

The PRA model used to perform the analysis (e.g. - CCDP calculations) for the proposed BFN Unit 3, Third Ten-Year Interval RI-ISI program is as follows.

BFN Unit 3 Baseline Model: U3011702 dated January 17, 2002
CDF = 1.91E-6
LERF = 2.69E-7

NRC Request 12

In reference to TVA's response to NRC question 8: Address the impact of the "non-EPU" findings from the NRC January 23-26, 2006, PRA audit on the results of the third interval risk-informed inservice inspection program as discussed during the November 2, 2006 conference call.

TVA Response to NRC Request 12

Specific NRC questions relative to the general question above were discussed in a teleconference between BFN and the NRC staff on November 2, 2006. The NRC questions are related to Section 3.8 of the above referenced Unit 1 NRC January 23-26, 2006, PRA audit report. The NRC questions are shown below in bold text followed by the TVA response in plain text.

Section 3.8, Miscellaneous modeling issues that could impact CDF and LERF estimates.

Although only a few items were noted that fall into this category, they reinforce the need for a peer review of the models. Also, the number seems high considering the relatively short review period and the focused nature of the PRA audit.

TVA RESPONSE

A peer review has since been performed on the Unit 1 Model of Record. The single A level fact and observation has been resolved and the B level facts and observations are under review for path forward for each.

There were examples where CCF modeling was not correct; for example, the 12 identical service water pumps were grouped into smaller groups. CCF of the SRVs was also not carried out properly. The explanation given by the licensee was that the code could not handle these combinations; however, there are ways to consider CCF of large groups even given the code limitations (e.g. use of a global CCF event for the failure of a group of 4 or more).

TVA RESPONSE

Top event SWC models the global common cause failure of all service water pumps. SRV failure to open is modeled with the Multiple Greek Letter (MGL) methodology. Beta, Gamma, Delta, Epsilon and Zeta factors are used due to the large number of components in the group.

The licensee stated that they reduced high pressure coolant injection/reactor core isolation cooling CCF in order to lower the CDF to offset the risk increase from not crediting the CRD. This is not an appropriate approach for determining a change in risk, because a methodology or assumption change is being used to offset any real risk increase. The change in risk should be estimated with consistent assumptions and methodologies in the base case and modified case to provide an accurate estimate of the risk impact of the application itself.

TVA RESPONSE

While the industry has progressed over the years in developing more accurate common cause data (i.e., the INEEL CCF database, etc.), incorporation of this data to reduce or offset the increase in CDF and LERF due to the inability to utilize CRD in an enhanced flow mode was not the motivation for this update.

The reduction in HPCI/RCIC common cause factors was based on accepted industry practice and use of the INEEL CCF database.

Fault tree for plant air did not include the 480 volt ac power dependencies for the B and C compressors.

TVA RESPONSE

The B and C compressors are powered from 480V Unit boards, which are supplied from offsite power (not supplied by diesel generators during loss of offsite power). Therefore, this dependency is modeled by subsuming it into the relevant LOOP scenarios.

The licensee discovered logic errors in the loss of instrument bus A and loss of instrument bus B initiating event fault trees while trying to explain the logic to the audit team.

TVA RESPONSE

These errors were resolved through the TVA corrective action program process and incorporated into the Unit 1 model that has since been reviewed by the peer certification team. While the observation of the loss of instrument bus A and loss of instrument bus B initiating event fault trees was an error in model logic, it is not expected to result in any changes to the segments risk ranking. This conclusion also applies to Unit 3.

The licensee stated that some CDF is not accounted for because of truncation effects by solving directly for LERF and then figuring out CDF. This inappropriately understates risk by some unknown amount.

TVA RESPONSE

Due to the use of the Riskman Code, truncation (total frequency of all unresolved scenarios) is retained and reported. In an effort to resolve the issue that there could potentially be an entire family of scenarios “just below the surface,” a separate sensitivity evaluation is performed to ensure that CDF and LERF are, in fact, converting at the quantification cutoff selected for model use and reporting to the regulator. In the case of the recently peer-reviewed Unit 1 model, the following values are generated:

The accepted truncation guidelines are 5 decades below CDF or less than 5% change per decade reduction, whichever results in a lower truncation value for model quantification.

	1.00E-11	**1.00E-12**	**1.00E-13**
CDF	4.4029E-06	4.6733E-06	4.8081E-06
LERF	5.0682E-07	5.5043E-07	5.7257E-07
Percent Change in CDF		6.1%	2.9%
Percent change in LERF		8.6%	4.0%
Unaccounted Frequency	1.41E-04	4.45E-05	1.36E-05
Change in Unaccounted Frequency		3.17	3.27
Unaccounted Frequency as Factor of CDF	32.07	9.52	2.83

Based on this evaluation, the Unit 1 model can be quantified at 1E-12 for all initiating events.