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 – REACTOR VESSEL AND INTERNALS MECHANICAL SYSTEMS SECTIONS 3.1, 4.2, AND B.2.1 – 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 1, 2004, identified areas where additional information is needed to complete its review.

The specific areas requiring a request for additional information (RAI) are related to the aging management of Reactor Vessel and Internals Mechanical Sections 3.1, 4.2, and B.2.1 of the License Renewal Application (LRA). Drafted forms of these RAIs were discussed with the TVA Staff on a telephone conference call held on September 16, 2004.

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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s:/Licensing/Lic/BFN LR Reactor Vessel Mech Sections 3.1, 4.2, and B.2.1

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 REACTOR VESSEL AND INTERNAL MECHANICAL SYSTEMS
SECTIONS 3.1, 4.2, AND B.2.1

(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), RELATED TO REACTOR VESSEL AND INTERNAL MECHANICAL SYSTEMS SECTIONS 3.1, 4.2, AND B.2.1

By letter dated December 31, 2003, the Tennessee Valley Authority (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 1, 2004, identified areas where additional information is needed to complete its review.

The specific areas requiring a request for additional information (RAI) are related to the aging management of Reactor Vessel and Internals Mechanical Sections 3.1, 4.2, and B.2.1 of the License Renewal Application (LRA). Drafted forms of these RAIs were discussed with the TVA Staff on a telephone conference call held on September 16, 2004.

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

Steam Dryer

NRC RAI 3.1-1

NRC Information Notice 2002-26, Supplement 2 "Additional Flow-Induced Vibration Failures after a Recent Power Uprate" dated January 9, 2004, discusses flow-induced vibration damage to steam dryer cover plates, main steam electromatic relief valve as well as main steam line support clamps and tieback supports at Quad Cities Units 1 and 2. This damage is due to the extended power uprate which can significantly increase the steam velocity through the dryers. Failures of the cover plates of a steam dryer component would be a source of loose parts.
Title 10 of the Code of Federal Regulations 10 CFR 54.4(a)(2), states in part, that all non-safety related systems which would include the steam dryer, are within the scope of license renewal regulations if the failure of the component could prevent satisfactory accomplishment of an intended function defined in 10 CFR 54.4(a)(1).

Based on the aforementioned regulation, and that flow-induced vibration damage at Quad Cities during power uprate condition has resulted in loose parts, the staff believes that the loose parts have the potential to prevent a safe shutdown of the reactor, and maintain it in a safe shutdown condition. Therefore, the staff believes that the steam dryer component should be included within the scope of license renewal regulations in accordance with the requirements of 10 CFR 54.4(a)(2) at the BFN Units. The staff requests that the applicant provide an AMR and the appropriate AMP that will be implemented during the license renewal period for the steam dryer components.

TVA RESPONSE TO NRC RAI 3.1-1

The steam dryers have been added to the scope of license renewal. The aging management review will be completed and an aging management program will be established. The aging management program will be the NRC approved BWRVIP Steam Dryer Inspection and Evaluation Guidelines. If the BWRVIP Steam Dryer Inspection and Evaluation Guidelines are not approved by the NRC, then a plant specific aging management program will be submitted to the NRC for review and approval two years before the first BFN unit enters the period of extended operation.

NRC RAI 3.1.1-1

Table 3.1.1 Item 3.1.1.8, and Section 3.1.2.2.4 Paragraph 2 of the License Renewal Application (LRA) states that the plant-specific aging management program (AMP) for the vessel flange leak detection line will be implemented. The applicant has proposed to use a One-Time Inspection Program which is specified in Section B.2.1.29, "One-Time Inspection Program," of the LRA for the vessel flange leak detection line. Identify whether the vessel flange leak detection line has previously experienced cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC) or cyclic loading, and the extent of cracking. Identify the method of inspection in the "One-Time Inspection Program." Provide justification for why a One-Time inspection is adequate.

TVA RESPONSE TO NRC RAI-3.1.1-1

The vessel flange leak detection line has not previously experienced cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), or cyclic loading.

TVA agrees that the One-Time Inspection Program is not sufficient to address aging of the reactor vessel flange leak-off line. This correction was identified in response to Question 394 of the NRC's Consistent with GALL Audit (Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Additional Information (RAI) Developed During the License Renewal Audit Inspections for Comparison to Generic Aging Lessons Learned (GALL) During The Weeks of June 21, 2004 and July 26, 2004, dated October 8, 2004). As discussed in that response, the BFN reactor vessel flange leak detection line should have included the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program in addition to the One-Time Inspection Program as an aging management program.

NUREG-1801, Volume 1, Table 1, Item 3.1.1.8 states that the corresponding NUREG-1801 Volume 2 line item for the reactor vessel flange leak-off line is IV.A1.1-d. The Browns Ferry top head enclosure - vessel flange leak detection line is not consistent with NUREG-1801 Volume 2, Line IV.A1.1-d due to material differences. The BFN components corresponding to this line item are carbon and low alloy steel, whereas NUREG-1801, Volume 2, Line IV.A1.1-d refers to stainless steel. The components included in this line item are the penetration through the carbon and low alloy steel vessel flange and a short segment of carbon and low alloy steel piping and fittings external to the reactor vessel.

Carbon steel and low alloy steel components for the vessel flange leak detection line are not considered susceptible to SCC or IGSCC. The reactor vessel flange leak-off line is still considered susceptible to cracking growth from cyclic loading. Therefore, the corresponding aging management results line item should read:

Reactor Vessel Heads, Flanges, Shell	PB, SS	Carbon and Low Alloy Steel	Air/gas (internal)	Crack initiation/ growth due to cyclic loading. Loss of material due to crevice, general, and pitting corrosion.	ASME Section XI Subsections IWB, IWC and IWD Inservice Inspection Program (B.2.1.4) One-Time Inspection (B.2.1.29)	IV.A1.1-d	None	F, 2

The remaining RCPB portion of the vessel flange leak detection line is stainless steel. This stainless steel piping is included in the Feedwater System (003) at Browns Ferry. Aging of this piping is addressed in Table 3.4.2.3 as piping and fittings - small bore piping less than NPS 4.

The first paragraph of LRA Section 3.1.2.2.4.2 is revised to include the ASME Section XI Subsections IWB, IWC and IWD Inservice Inspection Program. Paragraph 3.1.2.2.4.2 thus reads as follows:

> "The Aging Management Review (AMR) results for the reactor vessel flange leak detection line are listed in Table 3.1.2.1, Reactor Vessel - Summary of Aging Management Evaluation and Table 3.4.2.3, Feedwater System (003) – Summary of Aging Management Evaluation. The AMPs for managing cracking of the vessel flange leak detection line are the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program (B.2.1.4) and the One-Time Inspection Program (Section B.2.1.29)."

NRC RAI 3.1.1-2

Table 3.1.1 Item 3.1.1.8, and Section 3.1.2.2.4 Paragraph 2 of LRA states that the plant-specific aging management program (AMP) for the jet pump sensing line will be implemented. The applicant has proposed to use One-Time Inspection Program which is specified in Section B.2.1.29 "One-Time Inspection Program," of the LRA for the jet pump sensing line. Identify whether the jet pump sensing line has previously experienced cracking due to SCC, IGSCC or cyclic loading, and the extent of cracking. Identify the method and frequency of inspection. Provide justification for why one-time inspection is adequate.

TVA RESPONSE TO NRC RAI-3.1.1-2

The jet pump assemblies - jet pump sensing line has not previously experienced cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), or cyclic loading.

NUREG-1801, Volume 1, Table 1, Item 3.1.1.8 states that the corresponding NUREG-1801 Volume 2 line item for the jet pump assemblies - jet pump sensing line is IV.B1.4-d. Section IV.B1 addresses BWR reactor vessel internals and thus NUREG-1801 Volume 2, Line IV.B1.4-d is referring to the portion of the jet pump sensing lines internal to the reactor vessel. BFN has

determined that the jet pump sensing lines internal to the reactor vessel are not within the scope of license renewal. Therefore, no aging management program is identified for the jet pump sensing lines internal to the reactor vessel. This question was responded to previously in response to Question 394 of the NRC's Consistent with GALL Audit (Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Additional Information (RAI) Developed During the License Renewal Audit Inspections for Comparison to Generic Aging Lessons Learned (GALL) During Weeks of June 21, 2004 and July 26, 2004, dated October 8, 2004).

The additional portions of the jet pump sensing line not covered by Table 3.1.1 Item 3.1.1.8 are the reactor vessel penetration and the external sensing lines. The jet pump instrumentation penetration is stainless steel clad carbon steel and is included in LRA Table 3.1.2.1, as a penetration. External to the reactor vessel, the stainless steel jet pump sensing lines are included in LRA Table 3.1.2.4 as piping and fittings - small bore piping less than NPS 4.

The second paragraph of LRA Section 3.1.2.2.4.2 is revised to read as follows:

> "BFN jet pump sensing lines internal to the reactor vessel are not subject to an AMR. The AMR results for the jet pump sensing lines penetrations and external lines are listed in Table 3.1.2.1, Reactor Vessel - Summary of Aging Management Evaluation and Table 3.1.2.4, Reactor Recirculation System (068) – Summary of Aging Management Evaluation. The AMPs for managing cracking of the jet pump sensing lines penetrations are the Boiling Water Reactor Penetrations Program (Section B.2.1.11) and the Chemistry Control Program (Section B.2.1.5). The AMPs for managing SCC of the jet pump sensing lines external to the reactor vessel are the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program (B.2.1.4), the Chemistry Control Program (Section B.2.1.5), and the One-Time Inspection Program (Section B.2.1.29)."

NRC RAI 3.1.1-3

Table 3.1.1, Item 3.1.1.33 indicates that the AMP specified in B.2.1.14, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel Program," is consistent with NUREG-1801 with exceptions. However, no exceptions are taken in the AMP B.2.1.14. Provide an explanation for this discrepancy.

TVA RESPONSE TO NRC RAI-3.1.1-3

Table 3.1.1, item 3.1.1.33 states that the, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel Program," is "Consistent with NUREG-1801." No discrepancy exists. See RAI-3.1.2.2-9 for additional discussion of the "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel Program."

NRC RAI 3.1.2-1

The RAI described below applies to the following systems that are listed in Tables 3.1.2.1 and 3.1.2.3:

Table 3.1.2.1 - Vessel Attachment Welds
Table 3.1.2.1 - Vessel Heads, Flanges and Shells
Table 3.1.2.1 - Reactor Vessel Nozzles and Safe Ends
Table 3.1.2.1 - Reactor Vessel Penetrations
Table 3.1.2.3 - Reactor Vessel Vents and Drains

The LRA identifies no aging effect for the external surface of carbon and low alloy steel reactor vessel attachment weld components and vessel heads, flanges and shells, which are exposed to containment environment for considerable length of time during the dry lay up for BFN Unit 1. The BWR containment environment typically has high humidity. The carbon and low alloy steel components that are exposed to this environment may experience loss of material due to corrosion. Explain why loss of material is not considered as an aging effect for these components, or provide a program for managing such an effect.

TVA RESPONSE TO NRC RAI-3.1.2-1

The BFN LRA identifies no aging effects for the external surface of carbon and low alloy steel reactor vessel and associated components because the LRA addressed the period of extended operation. During the period of extended operation, the external surfaces will be greater than 212°F and will not be subject to external corrosion.

Verification that the reactor vessel and associated components are acceptable for operation during the current and extended operating period is addressed by the Unit 1 restart program. As part of the Unit 1 restart program, selected portions of the reactor vessel and associated components not being replaced are inspected for degradation. Degraded components identified by this inspection are refurbished or replaced as appropriate.

NRC RAI 3.1.2-2

The RAI described below applies to the following systems that are listed in Table 3.1.2.1:

Table 3.1.2.1 - Reactor Vessel Nozzles and Safe End

Table 3.1.2.1 - Reactor Vessel Penetrations

The aging management program specified in B.2.1.12, "Boiling Water Reactor Vessel Internals Program" is applicable to the aforementioned components. Therefore, the AMP specified in B.2.1.12 should be referenced in the Tables 3.1.2.1 for these components.

TVA RESPONSE TO NRC RAI-3.1.2-2

Based on NUREG 1801 Volume 2, line items IV.A1-3 (Nozzles) and IV.A1-4 (Nozzle safe ends), the BWR Vessel Internals Program is not applicable to the Reactor Vessel Nozzles and Safe Ends. A review of the identified aging management programs confirms the identified aging management programs adequately cover the Reactor Vessel Nozzles and Safe Ends.

Based on NUREG 1801 Volume 2, line item IV.A1-5 (Penetrations), the BWR Vessel Internals Program is not applicable to the Reactor Vessel Penetrations. A review of the identified aging management program identified that penetrations listed in NUREG 1801 Volume 2, line item IV.A1-5 are not sufficiently covered by the Boiling Water Reactor Penetrations Program. NUREG-1801, Volume 2, Chapter XI.M8 BWR Penetrations states in the program description, "The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-49 and BWRVIP-27 documents ... BWRVIP-49 provides guidelines for instrument penetrations, and BWRVIP-27 addresses the standby liquid control (SLC) system nozzle or housing." In addition, to instrument penetrations and SLC nozzle currently included in NUREG-1801, Volume 2, Chapter XI.M8 BWR Penetrations, NUREG 1801 Volume 2, line item IV.A1-5 also includes control rod drive stub tubes, flux monitor, and drain line.

The Boiling Water Reactor Penetrations Program will be revised to state that the control rod drive stub tubes and flux monitor tubes are inspected in accordance with BWRVIP-47 and that the reactor vessel drain line is inspected in accordance with the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program.

Vessel Attachment Welds

NRC RAI 3.1.2.1-1

In Table 3.1.2.1 of the LRA indicates that the AMP for the vessel inside diameter (ID) attachment welds comply with the requirements specified in Section B.2.1.7, "Boiling Water Reactor Vessel Inside Diameter Attachment Welds Program" of the LRA. Section B 2.1.7 states that the frequency and the method of inspection specified in BWRVIP-48 will be implemented for the attachment welds. These requirements apply to jet pump riser brace attachment, core spray piping bracket attachment, steam dryer support and hold down brackets, feedwater spargers, guide rod and surveillance sample holder. According to the Section 2.2.3 of BWRVIP-48, furnace-sensitized stainless steel vessel ID attachment welds are highly susceptible to IGSCC. The applicant should identify whether there are any furnace sensitized stainless steel attachment welds at Browns Ferry Nuclear (BFN) units, and explain what type of AMP is implemented for any existing furnace-sensitized stainless steel attachment welds. The applicant should also provide details on any additional augmented inspection program that is implemented for any existing furnace-sensitized stainless steel attachment welds at BFN units.

TVA RESPONSE TO RAI-3.1.2.1-1

BWRVIP-48, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines", requires augmented inspections for two attachments (steam dryer support and feedwater bracket attachment) if they include furnace-sensitized stainless steel material. At BFN, all reactor vessel inside diameter attachment welds are assumed to be furnace-sensitized and are inspected per the BWRVIP requirements.

Reactor Vessel Closure Studs and Nuts

NRC RAI 3.1.2.1-2

The LRA identifies aging effects as distortion/plastic deformation due to stress relaxation, and loss of material due to mechanical wear, which are specified in the AMP B.2.1.6, "Reactor Head Closure Stud Program" for the reactor vessel closure studs and nuts. Identify whether the reactor closure studs and nuts have experienced the aforementioned aging effects at the BFN units. Provide information as to how the plant-

specific experience related to this aging effect impacts the attributes specified in AMP-B.2.1.6.

TVA RESPONSE TO NRC RAI-3.1.2.1-2

BFN has not identified any reactor vessel closure stud or nut degradation resulting in distortion/plastic deformation due to stress relaxation or loss of material due to mechanical wear. No reactor vessel closure studs or nuts have been replaced for these reasons. Two studs were replaced on Unit 2 during the Unit 2 Cycle 4 refueling outage. These were replaced because of physical thread damage. From discussions with plant personnel present during that time period, this damage was the result of impacts during handling and refueling operations and not the result of inservice stress or wear. Based on this plant-specific experience, there was no impact on the attributes specified in AMP-B.2.1.6.

Vessel Heads, Flanges and Shells

RAI 3.1.2.1-3

The LRA identifies aging effects as loss of material due to crevice, general and pitting corrosion, and cracking due to cyclic loading, which are monitored by the AMP B.2.1.29, "One-Time Inspection Program." Identify whether the vessel heads, flanges and shells have previously experienced cracking due to cyclic loading or loss of material due to crevice, general and pitting corrosion. Identify the method of inspection for the "One-Time Inspection Program." Provide justification for why a One-Time inspection is adequate.

TVA RESPONSE TO NRC RAI-3.1.2.1-3

The line item in Table 3.1.2.1 referenced addresses the reactor vessel flange leak-off line. Correction to this LRA line item is addressed in the RAI-3.1.1-1 response.

Reactor Vessel Nozzles and Safe Ends

NRC RAI 3.1.2.1-4

(A) Table 3.1.2.1 of the LRA identifies no aging effect for the external surface of carbon, and low alloy steel reactor vessel nozzles and safe end components that are exposed to containment environment for a considerable length of time during the dry lay up for BFN Unit 1. The BWR containment

environment typically has high humidity. Components that are exposed to this environment may experience loss of material due to corrosion. Explain why loss of material is not considered as an aging effect for these components, or provide a program for managing such effect.

(B) Table 3.1.2.1 of the LRA indicates that the AMP for the feedwater vessel nozzle complies with the requirements specified in the Section B.2.1.8, "Boiling Water Reactor Feedwater Nozzle Program" of the LRA. AMP B 2.1.8 states that the recommendations of NUREG-0619 "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" will be implemented for the AMP of the feedwater nozzle. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of the aging effect of the feedwater nozzles at BFN units. Provide information as to how the plant-specific experience related to this aging effect impacts the attributes specified in AMP B.2.1.8.

(C) Identify whether the dissimilar metal welds of reactor vessel nozzles and safe end components have previously experienced cracking due to SCC, IGSCC or cyclic loading, and the extent of cracking. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of the aging effects of the dissimilar metal welds of reactor vessel nozzles and safe end components at BFN units. Provide information as to how the plant-specific experience related to these aging effects impacts the attributes specified in AMP-B.2.1.8.

TVA RESPONSE TO NRC RAI-3.1.2.1-4(A)

The BFN LRA identifies no aging effects for the external surface of carbon and low alloy steel reactor vessel nozzles and safe end components because the LRA addressed the period of extended operation. During the period of extended operation, the external surfaces will be greater than 212°F and will not be subject to external corrosion.

Verification that the reactor vessel nozzles and safe end components are acceptable for operation during the current and

extended operating period is addressed by the Unit 1 restart program. As part of the Unit 1 restart program, selected portions of the reactor vessel nozzles and safe end components not being replaced are inspected for degradation. Degraded components identified by this inspection are refurbished or replaced as appropriate.

TVA RESPONSE TO NRC RAI-3.1.2.1-4(B)

The BFN program requires inspections of the feedwater nozzles in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section XI Subsection IWB and the recommendations of General Electric NE-523-A71-0594-A, Revision 1.

The following tables provide information on the scope, techniques of past inspections, the results obtained, and the frequency of the inspections.

Unit One, First Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-A-IR	1	UT	Acceptable
N4-A-IR	4	UT	Acceptable
N4-A-NB	1	UT	Acceptable
N4-A-NB	4	UT	Acceptable
N4-A-NV	1	UT	Acceptable
N4-A-NV	4	UT	Acceptable
N4-B-IR	1	UT	Acceptable
N4-B-IR	4	UT	Acceptable
N4-B-NB	1	UT	Acceptable
N4-B-NB	4	UT	Acceptable
N4-B-NV	1	UT	Acceptable
N4-B-NV	4	UT	Acceptable
N4-C-IR	1	UT	Acceptable
N4-C-IR	4	UT	Acceptable
N4-C-NB	1	UT	Acceptable
N4-C-NB	4	UT	Acceptable
N4-C-NV	1	UT	Acceptable
N4-C-NV	4	UT	Acceptable
N4-D-IR	1	UT	Acceptable
N4-D-IR	4	UT	Acceptable
N4-D-IR	5	UT	Acceptable
N4-D-NB	1	UT	Acceptable
N4-D-NB	4	UT	Acceptable
N4-D-NV	1	UT	Acceptable

Unit One, First Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-D-NV	5	UT	Acceptable
N4-E-IR	1	UT	Acceptable
N4-E-IR	4	UT	Acceptable
N4-E-NB	1	UT	Acceptable
N4-E-NB	4	UT	Acceptable
N4-E-NV	1	UT	Acceptable
N4-F-IR	1	UT	Acceptable
N4-F-IR	4	UT	Acceptable
N4-F-NB	1	UT	Acceptable
N4-F-NB	4	UT	Acceptable
N4-F-NV	1	UT	Acceptable

IR - Inner Radius; NV - Nozzle to Vessel Weld; NB - Nozzle Bore

N4 Feed Water Nozzles, A through F, inner radius and nozzle bore ultrasonic examinations are scheduled to be examined per NUREG-0619 during Unit 1, Cycle 6, prior to restart. N4D-NV and N4E-NV are scheduled to be ultrasonically examined for ASME Section XI Code credit in Cycle 6.

Unit Two, First Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-A-IR	1	UT	Acceptable
N4-A-IR	3	UT	Acceptable
N4-A-IR	5	UT	Acceptable
N4-A-NB	1	UT	Acceptable
N4-A-NB	3	UT	Acceptable
N4-A-NB	5	UT	Acceptable
N4-A-NV	3	UT	Acceptable
N4-B-IR	1	UT	Acceptable
N4-B-IR	3	UT	Acceptable
N4-B-IR	4	UT	Acceptable
N4-B-IR	5	UT	Acceptable
N4-B-NB	1	UT	Acceptable
N4-B-NB	3	UT	Acceptable
N4-B-NB	5	UT	Acceptable
N4-B-NV	4	UT	Acceptable
N4-C-IR	1	UT	Acceptable
N4-C-IR	3	UT	Acceptable
N4-C-IR	4	UT	Acceptable

Unit Two, First Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-C-IR	5	UT	Acceptable
N4-C-NB	1	UT	Acceptable
N4-C-NB	3	UT	Acceptable
N4-C-NB	5	UT	Acceptable
N4-C-NV	4	UT	Acceptable
N4-D-IR	1	UT	Acceptable
N4-D-IR	3	UT	Acceptable
N4-D-IR	5	UT	Acceptable
N4-D-IR	5B	UT	Acceptable
N4-D-NB	1	UT	Acceptable
N4-D-NB	3	UT	Acceptable
N4-D-NB	5	UT	Acceptable
N4-D-NV	5B	UT	Acceptable
N4-E-IR	1	UT	Acceptable
N4-E-IR	3	UT	Acceptable
N4-E-IR	5	UT	Acceptable
N4-E-IR	5B	UT	Acceptable
N4-E-NB	1	UT	Acceptable
N4-E-NB	3	UT	Acceptable
N4-E-NB	5	UT	Acceptable
N4-E-NV	5B	UT	Acceptable
N4-F-IR	1	UT	Acceptable
N4-F-IR	3	UT	Acceptable
N4-F-IR	5	UT	Acceptable
N4-F-NB	1	UT	Acceptable
N4-F-NB	3	UT	Acceptable
N4-F-NB	5	UT	Acceptable
N4-F-NV	3	UT	Acceptable

IR - Inner Radius; NV - Nozzle to Vessel Weld; NB - Nozzle Bore

Unit Two, Second Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-A-IR	7	UT	Acceptable
N4-A-IR	9	UT	Acceptable
N4-A-NB	7	UT	Acceptable
N4-A-NB	9	UT	Acceptable
N4-A-NV	7	UT	Acceptable
N4-B-IR	7	UT	Acceptable
N4-B-IR	9	UT	Acceptable

Unit Two, Second Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-B-NB	7	UT	Acceptable
N4-B-NB	9	UT	Acceptable
N4-B-NV	9	UT	Acceptable
N4-C-IR	7	UT	Acceptable
N4-C-IR	9	UT	Acceptable
N4-C-NB	7	UT	Acceptable
N4-C-NB	9	UT	Acceptable
N4-C-NV	9	UT	Acceptable
N4-D-IR	7	UT	Acceptable
N4-D-IR	9	UT	Acceptable
N4-D-IR	11	UT	Acceptable
N4-D-NB	7	UT	Acceptable
N4-D-NB	9	UT	Acceptable
N4-D-NV	11	UT	Acceptable
N4-E-IR	7	UT	Acceptable
N4-E-IR	9	UT	Acceptable
N4-E-IR	11	UT	Acceptable
N4-E-NB	7	UT	Acceptable
N4-E-NB	9	UT	Acceptable
N4-E-NV	11	UT	Acceptable
N4-F-IR	7	UT	Acceptable
N4-F-IR	9	UT	Acceptable
N4-F-NB	7	UT	Acceptable
N4-F-NB	9	UT	Acceptable
N4-F-NV	7	UT	Acceptable

IR - Inner Radius; NV - Nozzle to Vessel Weld; NB - Nozzle Bore

Unit Two, Third Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-A-IR	12	UT	Acceptable
N4-A-NB	12	UT	Acceptable
N4-A-NV	12	UT	Acceptable
N4-B-IR	12	UT	Acceptable
N4-B-NB	12	UT	Acceptable
N4-B-NV	12	UT	Acceptable
N4-C-IR	12	UT	Acceptable
N4-C-NB	12	UT	Acceptable
N4-C-NV	12	UT	Acceptable

Unit Two, Third Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-D-IR	12	UT	Acceptable
N4-D-NB	12	UT	Acceptable
N4-D-NV	12	UT	Acceptable
N4-E-IR	12	UT	Acceptable
N4-E-NB	12	UT	Acceptable
N4-E-NV	12	UT	Acceptable
N4-F-IR	12	UT	Acceptable
N4-F-NB	12	UT	Acceptable
N4-F-NV	12	UT	Acceptable

IR - Inner Radius; NV - Nozzle to Vessel Weld; NB - Nozzle Bore

Unit Three, First Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-A-IR	1	UT	Acceptable
N4-A-IR	3	UT	Acceptable
N4-A-IR	5	UT	Acceptable
N4-A-NB	1	UT	Acceptable
N4-A-NB	3	UT	Acceptable
N4-A-NB	5	UT	Acceptable
N4-A-NV	3	UT	Acceptable
N4-B-IR	1	UT	Acceptable
N4-B-IR	3	UT	Acceptable
N4-B-IR	4	UT	Acceptable
N4-B-IR	5	UT	Acceptable
N4-B-NB	1	UT	Acceptable
N4-B-NB	3	UT	Acceptable
N4-B-NB	5	UT	Acceptable
N4-B-NV	4	UT	Acceptable
N4-C-IR	1	UT	Acceptable
N4-C-IR	3	UT	Acceptable
N4-C-IR	4	UT	Acceptable
N4-C-IR	5	UT	Acceptable
N4-C-NB	1	UT	Acceptable
N4-C-NB	3	UT	Acceptable
N4-C-NB	5	UT	Acceptable
N4-C-NV	4	UT	Acceptable
N4-D-IR	1	UT	Acceptable
N4-D-IR	3	UT	Acceptable
N4-D-IR	5	UT	Acceptable

Unit Three, First Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-D-IR	5B	UT	Acceptable
N4-D-NB	1	UT	Acceptable
N4-D-NB	3	UT	Acceptable
N4-D-NB	5	UT	Acceptable
N4-D-NV	5B	UT	Acceptable
N4-E-IR	1	UT	Acceptable
N4-E-IR	3	UT	Acceptable
N4-E-IR	5	UT	Acceptable
N4-E-IR	5B	UT	Acceptable
N4-E-NB	1	UT	Acceptable
N4-E-NB	3	UT	Acceptable
N4-E-NB	5	UT	Acceptable
N4-E-NV	5B	UT	Acceptable
N4-F-IR	1	UT	Acceptable
N4-F-IR	3	UT	Acceptable
N4-F-IR	5	UT	Acceptable
N4-F-NB	1	UT	Acceptable
N4-F-NB	3	UT	Acceptable
N4-F-NB	5	UT	Acceptable
N4-F-NV	3	UT	Acceptable

IR - Inner Radius; NV - Nozzle to Vessel Weld; NB - Nozzle Bore

Unit Three, Second Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-A-IR	8	UT	Acceptable
N4-A-IR	10	UT	Acceptable
N4-A-IR	11	UT	Acceptable
N4-A-NB	8	UT	Acceptable
N4-A-NB	11	UT	Acceptable
N4-A-NV	10	UT	Acceptable
N4-B-IR	8	UT	Acceptable
N4-B-IR	11	UT	Acceptable
N4-B-NB	8	UT	Acceptable
N4-B-NB	11	UT	Acceptable
N4-B-NV	8	UT	Acceptable
N4-C-IR	8	UT	Acceptable
N4-C-IR	11	UT	Acceptable
N4-C-IR	5	UT	Acceptable
N4-C-NB	8	UT	Acceptable

Unit Three, Second Interval, Feed Water Nozzle Examinations

Component ID	Cycle Examined	Examination Method	Results
N4-C-NB	11	UT	Acceptable
N4-C-NV	8	UT	Acceptable
N4-D-IR	8	UT	Acceptable
N4-D-IR	11	UT	Acceptable
N4-D-NB	8	UT	Acceptable
N4-D-NB	11	UT	Acceptable
N4-D-NV	11	UT	Acceptable
N4-E-IR	8	UT	Acceptable
N4-E-IR	11	UT	Acceptable
N4-E-NB	8	UT	Acceptable
N4-E-NB	11	UT	Acceptable
N4-E-NV	11	UT	Acceptable
N4-F-IR	8	UT	Acceptable
N4-F-IR	10	UT	Acceptable
N4-F-IR	11	UT	Acceptable
N4-F-NB	8	UT	Acceptable
N4-F-NB	11	UT	Acceptable
N4-F-NV	10	UT	Acceptable

IR - Inner Radius; NV - Nozzle to Vessel Weld; NB - Nozzle Bore

No cracking has been observed in the reactor feedwater nozzles as shown above. Based on the BFN operating experience, no repairs to the reactor feedwater nozzles have been performed. Improvements in the Chemistry Control Program are the primary mitigative measure to preclude IGSCC of reactor vessel nozzles and safe-end components. In addition, the cladding has been removed from the reactor vessel feedwater nozzles. The plant-specific experience related to feedwater nozzles has no impact on the attributes specified in AMP-B.2.1.8, "Boiling Water Reactor Feedwater Nozzle Program."

TVA RESPONSE TO NRC RAI-3.1.2.1-4(C)

The ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program inspections and frequencies are in accordance with ASME Section XI, Table IWB-2500-1 examination category B-F. Alternative examination requirements for examination category B-F are provided by the Risk-Informed Inservice Inspection Program. The BWR Stress Corrosion Cracking Program inspections and frequencies are in accordance with the normal water chemistry guidelines contained in BWRVIP-75. Alternative examination requirements for IGSCC Category A welds are provided

by the Risk-Informed Inservice Inspection Program. BFN Units 2 and 3 have implemented a Risk-Informed Inservice Inspection Program.

The following tables provide information on the scope, techniques of past inspections, the results obtained, and the frequency of the inspections. As identified in LRA Appendix B.2.1.10, "Boiling Water Reactor Stress Corrosion Cracking Program," the BWR Stress Corrosion Cracking Program will be implemented on Unit 1 prior to the period of extended operation. With the exception of RCRD-1-33, "CRD nozzle to cap weld," the nozzle to safe end welds will be cut out and replaced during the Unit 1 recovery outage. The IGSCC Category and examination frequency for the welds will reflect material improvements and mitigation efforts. Historical results for RCRD-1-33 are presented below.

Unit 1 Dissimilar Metal Welds

Component	Component ID	Cycle Examined	Examination	Results
CRD nozzle to cap	RCRD-1-33	1	UT	Acceptable
	RCRD-1-33	4	PT/UT	Acceptable

Unit 2 Dissimilar Metal Welds

Component	Component ID	Cycle Examined	Examination	Results
Recirculation inlet nozzle to safe end	2RA1	5B	Baseline UT/PT	Acceptable
	2RA1	9	PT/UT	Acceptable
	2RB1	5B	Baseline UT/PT	Acceptable
	2RB1	9	PT/UT	Acceptable
	2RC1	5B	Baseline UT/PT	Acceptable
	2RC1	7	PT/UT	Acceptable
	2RD1	5B	Baseline UT/PT	Acceptable
	2RD1	9	PT/UT	Acceptable
	2RE1	5B	Baseline UT/PT	Acceptable
	2RF1	5B	Baseline UT/PT	Acceptable
	2RG1	5B	Baseline UT/PT	Acceptable
	2RH1	5B	Baseline UT/PT	Acceptable

Unit 2 Dissimilar Metal Welds

Component	Component ID	Cycle Examined	Examination	Results
	2RJ1	5B	Baseline UT/PT	Acceptable
	2RJ1	6	PT/UT	Acceptable
	2RK1	5B	Baseline UT/PT	Acceptable
Recirculation outlet nozzle to safe end	N1A-SE	3	PT/UT	Acceptable
	N1A-SE	4	UT	Acceptable
	N1A-SE	5	PT/UT	Acceptable
	N1A-SE	5B	UT	Acceptable
	N1A-SE	7	PT/UT	Acceptable
	N1B-SE	4	PT/UT	Acceptable
	N1B-SE	5	PT/UT	Acceptable
	N1B-SE	5B	UT	Acceptable
	N1B-SE	7	UT	Acceptable
	N1B-SE	9	PT/UT	Acceptable
Core spray nozzle to safe end	TCS-2-401	1	PT/UT	Acceptable
	TCS-2-401	4	UT	Acceptable
	TCS-2-401	5	PT/UT	Acceptable
	TCS-2-401	5B	PT/UT	Acceptable
	TCS-2-401	7	UT	Acceptable
	TCS-2-401	8	PT/UT	Acceptable
	TCS-2-417	1	PT/UT	Acceptable
	TCS-2-417	5	PT/UT	Acceptable
	TCS-2-417	5B	PT/UT	Acceptable
	TCS-2-417	7	PT/UT	Acceptable

Unit 2 Dissimilar Metal Welds

Component	Component ID	Cycle Examined	Examination	Results
Core spray safe end to pipe	TSCS-2-418	1	PT/UT	Acceptable
	TSCS-2-418	5B	PT/UT	Acceptable
	TSCS-2-418	7	UT	Acceptable
	TCS-2-403	1	PT/UT	Acceptable
	TCS-2-403	5	PT/UT	Acceptable
	TCS-2-403	5B	PT/UT	Acceptable
	TCS-2-403	7	UT	Acceptable
CRD nozzle to cap	RCRD-2-33	1	UT	Acceptable
	RCRD-2-33	6	UT	Acceptable
	RCRD-2-33	8	PT/UT	Acceptable
	RCRD-2-33	10	UT	Acceptable

Unit 3 Dissimilar Metal Welds

Component	Component ID	Cycle Examined	Examination	Results
Recirculation inlet nozzle to safe end	RWR-3-001-G001	5B	Baseline UT/PT	Acceptable
	RWR-3-001-G004	5B	Baseline UT/PT	Acceptable
	RWR-3-001-G007	5B	Baseline UT/PT	Acceptable
	RWR-3-001-G007	11	UT	Acceptable
	RWR-3-001-G010	5B	Baseline UT/PT	Acceptable
	RWR-3-001-G013	5B	Baseline UT/PT	Acceptable
	RWR-3-002-G001	5B	Baseline UT/PT	Acceptable
	RWR-3-002-G004	5B	Baseline UT/PT	Acceptable
	RWR-3-002-G007	5B	Baseline UT/PT	Acceptable
	RWR-3-002-G007	11	UT	Acceptable
	RWR-3-002-G010	5B	Baseline UT/PT	Acceptable
	RWR-3-002-G013	5B	Baseline UT/PT	Acceptable

Unit 3 Dissimilar Metal Welds

Component	Component ID	Cycle Examined	Examination	Results
Recirculation outlet nozzle to safe end	N1A-SE	2	PT/UT	Acceptable
	N1A-SE	5	UT	Acceptable
	N1A-SE	5B	UT	Acceptable
	N1A-SE	8	PT/UT	Acceptable
	N1B-SE	4	PT/UT	Acceptable
	N1B-SE	5	UT	Acceptable
	N1B-SE	5B	UT	Acceptable
	N1B-SE	8	UT	Acceptable
Core spray nozzle to safe end	TCS-3-401	2	PT/UT	Acceptable
	TCS-3-401	4	PT/UT	Acceptable
	TCS-3-401	5B	UT	Acceptable
	TCS-3-417	2	PT/UT	Acceptable
	TCS-3-417	5B	UT	Acceptable
	TCS-3-417	8	PT/UT	Acceptable
Core spray safe end to pipe	TSCS-3-402	2	PT/UT	Acceptable
	TSCS-3-402	5	PT/UT	Acceptable
	TSCS-3-402	5B	PT/UT	Acceptable
	TSCS-3-418	2	PT/UT	Acceptable
	TSCS-3-418	5	PT/UT	Acceptable
	TSCS-3-418	5B	UT	Acceptable
	TSCS-3-418	8	PT/UT	Acceptable
CRD nozzle to cap	RCRD-3-33	1	UT	Acceptable
	RCRD-3-33	5B	UT	Acceptable
	RCRD-3-33	8	UT	Acceptable

No cracking has been observed in any dissimilar metal welds of reactor vessel nozzles and safe-end components shown above. Following the Units 2 and 3 extended shutdowns, some RWCU, CS and Recirculation piping was replaced. Improvements in the Chemistry Control Program and use of IGSCC resistant materials are the primary mitigative measures implemented to preclude IGSCC of reactor vessel nozzles and safe-end components.

The plant-specific experience related to dissimilar metal welds of reactor vessel nozzles and safe-end components has no impact on the attributes specified in AMP-B.2.1.4 ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program and AMP-B.2.1.10 Boiling Water Reactor Stress Corrosion Cracking Program. Note that RAI-3.1.2.1-4(C) incorrectly refers to AMP-B.2.1.8. AMP-B.2.1.8 is only applicable to the Reactor Feedwater Nozzles.

Reactor Vessel Penetrations

NRC RAI 3.1.2.1-5

(A) Table 3.1.2.1 of the LRA identifies no aging effect for the external surface of carbon and low alloy steel reactor vessel penetrations that are exposed to containment environment for a considerable length of time during the dry lay up for BFN Unit 1. The BWR containment environment typically has high humidity. Components that are exposed to this environment may experience loss of material due to corrosion. Explain why loss of material is not considered as an aging effect for these components or provide a program for managing such an effect.

(B) Identify whether the dissimilar metal welds of reactor vessel penetrations have previously experienced cracking due to SCC, IGSCC or cyclic loading, and the extent of cracking. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of these aging effects of the reactor vessel penetrations at BFN units. Provide information as to how the plant-specific experience related to these aging effects impacts the attributes specified in AMP B.2.1.11, "Boiling Water Reactor Penetration Program."

TVA RESPONSE TO NRC RAI-3.1.2.1-5(A)

The BFN LRA identifies no aging effects for the external surface of carbon and low alloy steel reactor vessel penetrations because the LRA addressed the period of extended operation. During the period of extended operation, the external surfaces will be greater than 212°F and will not be subject to external corrosion.

Verification that the reactor vessel penetrations are acceptable for operation during the current and extended operating period is addressed by the Unit 1 restart program. As part of the Unit 1 restart program, selected portions of the reactor vessel penetrations not being replaced are inspected for degradation. Degraded components identified by this inspection are refurbished or replaced as appropriate.

TVA RESPONSE TO RAI-3.1.2.1-5(B)

The CRD stub tubes, instrumentation nozzles/nozzle safe-ends, standby liquid control nozzle, jet pump instrumentation nozzle, drain line nozzle, and incore monitor housing penetrations are currently inspected during the ASME Section XI, IWB-2500, Code Category B-P system leakage test each refueling outage. No cracking has been observed in dissimilar metal welds of reactor vessel penetrations. Based on the BFN operating experience, no repairs to penetrations have been performed. Improvements in the Chemistry Control Program are the primary mitigative measure to preclude IGSCC of reactor vessel penetrations.

The plant-specific experience related to penetrations has no impact on the attributes specified in AMP B.2.1.11, "Boiling Water Reactor Penetration Program."

BFN procedures are currently being revised to perform an enhanced leakage inspection, in accordance with BWRVIP-27, of the Standby Liquid Control (SLC) safe-end-to-nozzle weld during the ASME Section XI, IWB-2500, Code Category B-P system leakage test will be performed. The ASME Section XI, IWB-2500, Code Category B-F surface examination of the Standby Liquid Control (SLC) and Instrument Penetrations, are included in the Risk Informed Inservice Inspection Program. Based on the risk evaluation, surface inspections of these penetrations are not currently performed. BFN Units 2 and 3 have implemented a Risk-Informed Inservice Inspection Program.

Reactor Vessel Internals Core Shroud and Core Plate

NRC RAI 3.1.2.1-6

(A) According to Section IV-B.1.1-d/B1.1.4 of NUREG-1801, augmented inspection of access hole covers is required for Alloy 600 materials and Alloy 182 welds. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections of the access hole covers.

(B) According to Section IV-B.1.1-d of NUREG-1801, irradiation assisted stress corrosion cracking is an aging effect for core shroud components. Please provide an explanation for excluding this aging mechanism in the Table 3.1.2.1. Provide details of the AMP that will be implemented on this system.

(C) Describe plant-specific experience related to IGSCC cracking of the stainless steel and Inconel components in the core shroud, and shroud support access hole covers. Provide details on any occurrence of IGSCC cracking, and the effective AMP that will be implemented on these systems at BFN Units.

(D) The applicant should address the plant-specific experience on sudden increase in RCS water conductivity due to a leak in condensate and or reactor water clean up systems. Provide information on the impact of sudden increase in RCS water conductivity on IGSCC of core shroud welds.

(E) Provide information on verification methods to monitor the effectiveness of the hydrogen water chemistry program. Explain the methodology of ensuring hydrogen availability in the core shroud region. If ECP probes are not used to monitor availability of hydrogen, explain the validity of using secondary parameters (i.e., main steam/feedwater oxygen levels) to assess the hydrogen availability at core shroud welds.

TVA RESPONSE TO NRC RAI-3.1.2.1-6(A)

See the response to RAI-B.2.1.12-1(C) response for details on the scope, techniques, and frequency of the inspections of the core shroud access hole covers.

The Unit 1 access hole covers currently have indications of cracking and will be replaced prior to Unit 1 restart.

Unit 2 access hole cover inspections results are as follows:

1996 - UT examination with no reportable indications.
1999 - UT examination with no reportable indications.

Unit 3 access hole cover inspections results are as follows:

1994 - UT examination with no reportable indications.
1998 - UT examination with no reportable indications.
2004 - EVT-1 examination with no reportable indications.

Based on the plant operating experience, the Unit 1 Access Hole Covers will be replaced prior to Unit 1 restart. No repairs have been performed on the Unit 2 and 3 Access Hole Covers. Improvements in the Chemistry Control Program are the primary

mitigative measure to preclude IGSCC. In addition, the Unit 1 access hole cover design will be changed to a bolted design versus the current welded design.

TVA RESPONSE TO NRC RAI-3.1.2.1-6(B)

The stress corrosion cracking identified as an aging effect for core shroud components includes irradiation assisted stress corrosion cracking. The following note from the aging management review was inadvertently omitted from the LRA, "The SCC aging mechanism includes intergranular SCC and irradiation assisted SCC." The applicable aging management programs are the Boiling Water Reactor Vessel Internals Program and the Chemistry Control Program.

TVA RESPONSE TO NRC RAI-3.1.2.1-6(C)

Indications have been reported in Unit 1 core shroud welds (H-1, H-2, H-3, H-4, and H-5). Core shroud welds H-6 and H-7 have not been examined due to interference from vibration sensing lines. These welds will be ultrasonically (UT) examined prior to Unit 1 Restart.

Indications have been reported in Unit 2 core shroud welds (H-1, H-2, H-3, H-5, H-6, and H-7).

Indications have been reported in the Unit 3 core shroud welds (H-1, H-2, H-3, H-4, H-5, and H-7).

BFN Unit 1 Core Shroud Access Hole Covers have crack indications and are scheduled for replacement prior to plant restart.

The core shroud and core shroud access hole covers are managed by the BWR Vessel Internals Program (B2.1.12) and the Chemistry Control Program (B.2.1.5). Refer to RAI-3.1.2.1-6(b), and RAI- 4.2.4-1 related to the core shroud. Refer to RAI-B.2.1.12-1(C) related to shroud support access hole covers.

TVA RESPONSE TO NRC RAI-3.1.2.1-6(D)

A review of BFN operating experience identified no instances of a sudden increase in RCS water conductivity due to a leak in condensate and/or reactor water clean up systems in the previous five years.

The guidelines for handling a sudden conductivity excursion are included in Appendix A of BFN procedure CI-13.1, Chemistry Program. These action levels are:

Action Level I: Action Level I is defined as the level at which data or engineering judgment indicates long term system reliability may be threatened, thereby warranting an improvement in operating practices.

Action Level II: Action Level II is defined as the level at which data or engineering judgment indicates that significant degradation of the system in the short term, thereby warranting prompt corrective action.

Action Level III: Action Level III is defined as the level at which data or engineering judgment indicates that it is inadvisable to continue unit operation.

TVA RESPONSE TO NRC RAI-3.1.2.1-6(E)

A conservative H_2/O_2 molar ratio is maintained to ensure hydrogen availability in the core shroud region. BFN does not utilize ECP probes and, therefore, alternate means are used to monitor HWC control. The acceptable alternate means are described in Section 5.4 of BWRVIP-79, "EPRI-103515-R2, BWR Water Chemistry Guidelines – 2000 Revision,"

BFN procedure CI-13.1, Chemistry Program, specifies a reactor water H_2/O_2 molar ratio of ≥4 for power operation. The effectiveness of maintaining an adequate H_2/O_2 molar ratio, and thus ECP, is described in BWRVIP-79.

Core Spray Spargers and Piping

NRC RAI 3.1.2.2-7

(A) According to the Section IV-B.1.3-a of NUREG-1801, irradiation assisted stress corrosion cracking is an aging effect for the core spray lines and spargers components. Please provide an explanation for excluding this aging effect in the Table 3.1.2.1. Provide details of the AMP that will be implemented on this system.

(B) Core Spray piping and spargers contain crevice conditions in some weld areas. Explain the methodology of ensuring hydrogen availability in these systems. Since the presence of crevice conditions enhances the occurrence of IGSCC, the

applicant should provide details on the type and extent of inspections to identify IGSCC, and the mitigation techniques at BFN Units 2 and 3.

TVA RESPONSE TO NRC RAI-3.1.2.2-7(A)

The stress corrosion cracking identified as an aging effect for core spray piping and spargers includes irradiation assisted stress corrosion cracking. The following note from the aging management review was inadvertently omitted from the LRA, "The SCC aging mechanism includes intergranular SCC and irradiation assisted SCC." The applicable aging management programs are the Boiling Water Reactor Vessel Internals Program and the Chemistry Control Program.

TVA RESPONSE TO NRC RAI-3.1.2.2-7(B)

A conservative H_2/O_2 molar ratio is maintained to ensure hydrogen availability at the core spray piping and spargers. BFN procedure CI-13.1, Chemistry Program, specifies a reactor water H_2/O_2 molar ratio of ≥ 4 for power operation. The effectiveness of maintaining an adequate H_2/O_2 molar ratio, and thus ECP, is described in BWRVIP-79.

The type and extent of inspections to identify core spray piping and sparger IGSCC is included in the Boiling Water Reactor Vessel Internals Program. See the response to RAI-B.2.1.12-1(B) for additional information.

Reactor Vessel Internals Dry Tube and Guide Tube

NRC RAI 3.1.2.2-8

(A) According to the Section IV-B.1.6-a of NUREG-1801, irradiation assisted stress corrosion cracking is an aging effect for dry tube and guide tube components. Please provide an explanation for excluding this aging effect in the Table 3.1.2.1. Provide information on the AMP that will be implemented on this system.

(B) The AMP for the dry tube and guide tube components addressed in the application references BWRVIP-47, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines." Table 3.1-2 of BWRVIP-47 indicates that some of the incore housing guide tubes and dry tubes for BFN Units 2 and 3 experienced cracking and were subsequently replaced with materials resistant to cracking. Provide information on the type and

grade of the replaced material, and its performance at BFN Units 2 and 3. The staff requests additional information on the type and extent of subsequent inspections of the dry tubes and guide tubes for BFN Units 2 and 3. The applicant should also address any existence of cracks in BFN Unit 1 dry tubes and guide tubes. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections, and any other relevant information related to the identification of cracks in the BFN Unit 1 dry tubes and guide tubes. Provide information as to how the plant-specific experience related to this aging effect impacts the attributes specified in AMP B.2.1.12, "Boiling Water Reactor Vessel Internals Program," and BWRVIP-47.

(C) According to the Section 2.2.1.2 of BWRVIP-47, furnace-sensitized stainless steel stub tubes are more susceptible to IGSCC. The applicant should provide information on any existing furnace-sensitized stub tubes at BFN Units. Provide details on the AMP that will be implemented for the furnace-sensitized stainless steel stub tubes at the BFN Units. Identify whether any furnace-sensitized stainless steel stub tubes have previously experienced cracking due to SCC, IGSCC, or cyclic loading, and the extent of the cracking. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections, and any other relevant information related to the identification of these aging effects at the BFN units. Provide information as to how the plant-specific experience related to these aging effects impacts the attributes specified in AMP-B.2.1.12, and BWRVIP-47.

(D) According to the Section 2.2.1.2 of BWRVIP-47, weld metal 182 is more susceptible to IGSCC. Provide details on the AMP for components that have 182 weld metal in these systems at BFN Units. Identify whether any 182 weld metals have previously experienced cracking due to SCC, IGSCC, or cyclic loading, and the extent of cracking. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of these aging effects at BFN units. Provide information as to how the plant-specific experience related to these aging effects

impacts the attributes specified in AMP-B.2.1.12, and BWRVIP-7.

TVA RESPONSE TO NRC RAI-3.1.2.2-8(A)

The stress corrosion cracking identified as an aging effect for dry tube and guide tube components includes irradiation assisted stress corrosion cracking. The following note from the aging management review was inadvertently omitted from the LRA, "The SCC aging mechanism includes intergranular SCC and irradiation assisted SCC." The applicable aging management programs are the Boiling Water Reactor Vessel Internals Program and the Chemistry Control Program.

TVA RESPONSE TO RAI-3.1.2.2-8(B)

Due to widespread cracking found during inspection of the Unit 2 and 3 dry tubes, all 12 Unit 2 SRM/IRM dry tubes were replaced (Unit 2 in 1994 during the Unit 2 Cycle 7 (U2C7) Refueling Outage), and in Unit 3 prior to its Restart in 1995) with dry tubes of improved materials and design that have been manufactured after 1986. The new dry tube design eliminated a crevice in the upper portion (plunger area) of the existing design. This crevice had been determined to be a source of crack propagation in the dry tubes due to exposure to reactor coolant. Additionally, the material in the plunger area was changed from the existing 304 stainless steel material to 304L stainless steel, making the new dry tubes less susceptible to IGSCC and IASCC.

Inspection of the Unit 1 dry tubes has revealed widespread cracking. All 12 SRM/IRM dry tubes on Unit 1 will be replaced with the same design currently installed in Units 2 and 3 prior to Unit 1 Restart.

The BFN aging management review does not identify an inspection of the dry tubes. Operating experience for the original dry tube design, which contains a crevice is discussed above. No degradation has been identified in the new crevice-free design. Mitigative measures include improvements in the Chemistry Control Program and installation of a crevice-free design dry tube on Units 2 and 3. Unit 1 dry tubes will be replaced prior to plant restart.

The plant-specific experience related to the dry tubes has no impact on the attributes specified in AMP-B.2.1.12 and BWRVIP-47.

TVA RESPONSE TO NRC RAI-3.1.2.2-8(C)

BFN does not have furnace-sensitized stainless steel stub tubes on Units 1, 2, and 3. The stub tubes are manufactured from a nickel-alloy (Reference UFSAR Table 4.2-1.).

At BFN Unit 2, a general area inspection was performed in 1991 while a jet pump was removed. The periphery stub tubes were visible during this general area inspection. No stub tube indications or abnormalities were noted during this inspection. There have been no repairs associated with the CRD stub tubes.

Improvements in the BWR Chemistry Control Program help mitigate aging and degradation of the lower plenum components.

The plant-specific experience related to the stub tubes has no impact on the attributes specified in AMP-B.2.1.12 and BWRVIP-47 as no degradation has been identified. Note that for BFN, aging management of the stub tubes is included in AMP-B.2.1.11, "Boling Water Reactor Penetrations Program." The plant-specific experience related to the stub tubes has no impact on the attributes specified in AMP-B.2.1.11.

TVA RESPONSE TO RAI-3.1.2.2-8(D)

The following locations associated with the Lower Plenum have been identified at BFN as containing 182 weld metal:

- CRD Housing-to-Stub Tube Weld
- CRD Stub Tube-to-RPV Weld
- In-Core Housing to RPV Lower Head Penetration Weld
- In-Core Guide Tube to In-Core Housing Weld

The BFN aging management review does not identify an inspection of the listed welds. No cracking has been identified at BFN for the listed weld metal 182 welds. Improvements in the BWR Chemistry Control Program help mitigate aging and degradation of the lower plenum components.

The plant-specific experience related to the lower plenum weld metal 182 welds has no impact on the attributes specified in AMP-B.2.1.12 and BWRVIP-47 as no degradation has been identified.

Jet Pump Assembly

NRC RAI 3.1.2.2-9

Provide information on any existing Cast Austenitic Stainless Steel (CASS) jet pump components. The applicant should provide the information on the jet pump components:

(a) Information on type of casting (i.e.; centrifugal or static);

(b) The composition of CASS (i.e.; molybdenum content and delta ferrite values);

(c) Previous plant-specific experience regarding the cracked components and type and extent of subsequent inspection of CASS jet pump components due to neutron and thermal embrittlement. The fluence values should be based on the end of the extended period of operation and power uprate;

(d) The LRA should address any technical specification changes related to jet pump components.

TVA RESPONSE TO NRC RAI 3.1.2.2-9

The CASS jet pump components were manufactured to ASTM A351, grade CF8. These castings are low molybdenum and the maximum calculated delta ferrite percentage is below 20%. According to Table 2, CASS Thermal Aging Susceptibility Screening Criteria, contained in the May 19, 2000 NRC letter from Christopher I. Grimes to Douglas J. Walters, low molybdenum content and < 20% delta ferrite material are not susceptible to thermal aging for statically or centrifugally cast components. The NRC letter from Christopher I. Grimes to Carl Terry, dated June 5, 2001, states:

> "It is important to note that thermal and/or neutron embrittlement of CASS components becomes a concern only if cracks are present in the components, and that cracking has not been observed in CASS jet pump assembly components."

Section 2.4 of the same letter states:

> "Further, the BWRVIP and the NRC's Office of Nuclear Regulatory Research (RES) is engaged in a joint confirmatory research program to determine the effects of high levels of neutron fluence on BWR internals."

For open issues between the BWRVIP and NRC, TVA will work as part of the BWRVIP to resolve these issues generically. When resolved, TVA will follow the BWRVIP recommendations resulting from that resolution. The BWR Reactor Vessel Internals Program requires inspections of several jet pump assembly welds, which are more susceptible to cracking than the CASS components and will serve as an indication of the potential need for more extensive inspections later in life. Based on the above discussion no program is needed to manage the effects of thermal/neutron embrittlement of the CASS jet pump components.

Similar to the CASS jet pump components, the orificed fuel supports (OFS) are also manufactured to ASTM A351, grade CF8. These castings are low molybdenum and the maximum calculated delta ferrite percentage is below 20%.

For similar reasons as discussed for the jet pumps, CASS components, no program is needed to manage the effects of thermal/neutron embrittlement of the CASS orficed fuel supports.

Therefore, based on the above, Table 3.1.2.2, line items 21 and 30 should read as shown below and Appendix A.1.13 and Appendix B.2.1.14, which describe the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Program, are deleted.

Reactor Vessel Internals Fuel Support	SS	Stainless Steel - CASS	Treated Water (internal)	Change in material properties/ reduction in fracture toughness due to thermal aging and neutron irradiation embrittlement.	None	IV.B1.5-a	None	I, 3

Reactor Vessel Internals Jet Pump Assemblies	PB, SS	Stainless Steel - CASS	Treated Water (internal)	Change in material properties/ reduction in fracture toughness due to thermal aging and neutron irradiation embrittlement.	None	IV.B1.4-c	None	I, 3

Industry Standard Notes:

Note I Aging effect in NUREG-1801 item for this component, material and environment combination is not applicable.

Plant Specific Notes:

Note 3 Based on the CASS material properties, thermal and neutron irradiation embrittlement are not aging mechanisms requiring management for the period of extended operation.

Table 3.1.1, line items 33 should read as shown below:

3.1.1.33	Jet pump assembly castings and orificed fuel support	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	Not applicable to BFN.

Top Guide

NRC RAI 3.1.2.2-10

According to Section IV-B.1.2-a of NUREG-1801, irradiation assisted stress corrosion cracking is an aging effect for top guide components. Please provide explanation for excluding this issue in the Table 3.1.2.1. Provide details on the AMP that will be implemented on this system.

TVA RESPONSE TO NRC RAI-3.1.2.2-10

The stress corrosion cracking identified as an aging effect for top guide components includes irradiation assisted stress corrosion cracking. The following note from the aging management review was inadvertently included in the LRA, "The SCC aging mechanism includes intergranular SCC and irradiation assisted SCC." The applicable aging management programs are the Boiling Water Reactor Vessel Internals Program and the Chemistry Control Program. Refer to RAI-B.2.1.12-1(A) for additional information related to the top guide.

Bolting of Reactor Vessel Vents and Drains

NRC RAI 3.1.2.3-1

(A) According to the Section VIII.H.2-b of NUREG-1801, crack growth due to cyclic loading and stress corrosion cracking are aging effects for bolting applications. However, Table 3.1.2.3 does not address these aging effects. Provide an explanation for not including these aging effects in the LRA or provide a program for managing such an effect.

(B) According to the AMP specified in B.2.1.16, "Bolting Program," the BFN units previously experienced bolting degradation. Identify the location of these bolts, and provide information whether this degradation was related to the cracking due to cyclic loading or stress corrosion cracking. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of bolting degradation of the reactor vessel vents and drains at BFN units. Provide information as to how the plant-specific experience related to these aging effects impacts the attributes specified in AMP B 2.1.16.

(C) Table 3.1.2.3 indicates that the bolting function can be lost due to wear. This aging effect is not addressed in the Section VIIIH.2-b of NUREG-1801. Provide additional information on the previous plant-specific experience of loss of bolting function due to wear at BFN units. The applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of this aging effect of the reactor vessel vents and drains at BFN units. Provide information as to how the plant-specific experience related to this aging effect impacts the attributes specified in AMP B.2.1.16.

TVA RESPONSE TO NRC RAI-3.1.2.3-1(A)

Section VIII.H.2-b of NUREG-1801 addresses Steam and Power Conversion Systems, which are included in Section 3.4 of the BFN LRA and is not applicable to Section 3.1 of the BFN LRA. Section IV of NUREG 1801, Reactor Vessel, Internals, and Reactor

Coolant System does not contain a section similar to Section VIII.H of NUREG-1801. However, the BFN aging management review did evaluate Reactor Vessel Vents and Drains bolt/stud cracking. The BFN aging management review determined that these aging effects were not applicable based on industry and plant operating experience. The aging effects evaluations for these aging effects are summarized below.

- Stress Corrosion Cracking:

  Stress corrosion cracking of bolted closures and fasteners is a condition of high yield strength bolting material (> 150 ksi) where a fastener that is statically loaded well below its yield strength can experience sudden failure. Stress corrosion cracking occurs through the combination of high stress (both applied and residual tensile stresses), a corrosive environment, and a susceptible material. Stress corrosion cracking of high yield strength bolted closures in BWRs requires a corrosive environment typically attributed to leakage of pressure boundary joints or exposure to wetted ambient environments and the use of thread lubricant containing MoS_2 (molybdenum disulfide).

  Potentially susceptible mechanical bolting materials include alloy steels (ASTM A354 Grade BD, A540 and A574) and high yield strength heat-treated alloy steels (heat-treated 4130, 4140 and 4340 material). High yield strength heat-treated alloy steel bolting materials are not specified for flanged connections at BFN. High strength bolting of vendor supplied equipment has not been identified for mechanical components (such as pump casing studs or valve body/bonnet studs) where the material specifications are available. A review of the BFN operating experience did not identify any instances where mechanical component failure was attributable to stress corrosion cracking of high strength bolting. Therefore, loss of bolting function due to stress corrosion cracking of bolted joints of mechanical equipment is not expected and no aging management is required for the period of extended operation.

- Cyclic Loading:

  Crack initiation and growth due to cyclic loading is typically characterized by; 1) the failure is sudden with little or no necking-down of the part, 2) the component has been subjected to cyclic tensile loads, and 3) usually the cyclic loads are well below the material tensile strength. The susceptibility depends upon many factors including the

properties of the bolting material, bolting processing, defects in the material, stress levels, and the shape of the fastener. However, cracking due to high cycle fatigue is not considered a license renewal concern since it would be discovered during the current license period and corrected. In addition, cyclic primary loads are evaluated against conservative stress limits and should not be a contributor to fatigue due to the few number of stress cycles postulated (e.g., earthquake and fluid transient loads). Therefore cracking due to cyclic loading of mechanical bolted joints does not require aging management for the period of extended operation.

TVA RESPONSE TO NRC RAI-3.1.2.3-1(B)

The previous bolting degradation identified during the aging management review was restricted to general surface corrosion of carbon and low alloy steel bolting. The operating experience review identified no instances where this general corrosion resulted in component or system failure. The Bolting Integrity Program provides aging management for all mechanical bolting in the scope of license renewal. Class 1 bolting and Class 2 equivalent bolting greater than 2 inches are inspected in accordance with the "ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program" inspection requirements. The remaining mechanical bolting is managed by the System Monitoring Program. Reactor Vessel Vents and Drains bolting falls within the later category and is inspected by the System Monitoring Program. The System Monitoring Program performs an entire system inspection once per fuel cycle and includes visual inspections for evidence of material condition and bolting torque relaxation. The System Monitoring Program documents failures in either the maintenance work order or plant corrective action program, as appropriate. No instances of reactor vessel vents and drains bolting failure due to stress corrosion cracking or cracking due to cyclic loading was identified in this operating experience review. The plant-specific experience related to Reactor Vessel Vents and Drains bolting has no impact on the attributes specified in AMP-B.2.1.16.

TVA RESPONSE TO NRC RAI-3.1.2.3-1(C)

Section VIII.H.2-b of NUREG-1801 addresses Steam and Power Conversion Systems, which are included in Section 3.4 of the BFN LRA and is not applicable to Section 3.1 of the BFN LRA. Section IV of NUREG 1801, Reactor Vessel, Internals, and Reactor Coolant System does not contain a section similar to Section VIII.H of NUREG-1801. The BFN aging management review did include an evaluation of Reactor Vessel Vents and Drains bolt/stud wear. Wear was conservatively identified to be an aging effect that requires management for the period of extended operation for reactor coolant pressure boundary bolting. The aging management program for reactor vessel vents and drains bolting is described in RAI-3.1.2.3-1(B). No instances of reactor vessel vents and drains bolting failure due to wear were identified in this review. The aging effect evaluation for this aging effect is summarized below.

Wear:

Bolting degradation due to wear could potentially occur at locations of repeated relative motion of mechanical component bolted joints. Wear of bolted joint components is generally not a concern; however, for license renewal purposes, wear is being assumed as a potential mechanism for 'critical bolting applications.' 'Critical bolting applications' constitute reactor coolant pressure boundary components where closure bolting failure could result in loss of reactor coolant and jeopardize safe operation of the plant. These locations include bolted joints on the recirculation pumps and reactor coolant pressure boundary valves. Therefore, wear of reactor coolant pressure boundary bolted joints requires aging management for the period of extended operation.

The plant-specific experience related to Reactor Vessel Vents and Drains bolting has no impact on the attributes specified in AMP-B.2.1.16.

Bolting of Reactor Recirculation Systems

NRC RAI-3.1.2.4-1

(A) According to the Section IV.C1.2-e of NUREG-1801, loss of bolting function due to stress relaxation is identified as an aging effect. Provide additional information on the previous plant-specific experience of loss of bolting function due to this aging effect at BFN units. The

applicant should provide information on the scope and the techniques of the past inspections, the results obtained, applied mitigative methods, repairs, frequency of the inspections and any other relevant information related to the identification of this aging effect of the reactor recirculation systems at BFN units. Provide information as to how the plant-specific experience related to this aging effect impacts the attributes specified in AMP-B.2.1.16, "Bolting Integrity Program."

(B) According to the Section IV.C1.2-f of NUREG-1801, loss of bolting function due to fatigue is identified as the aging effect. Table 3.1.2.4 should address the relevant AMP for monitoring this aging effect for stainless steel, carbon and low alloy steel bolts.

TVA RESPONSE TO NRC RAI-3.1.2.4-1(A)

Stress relaxation was identified to be an aging effect that requires management for the period of extended operation for the reactor water recirculation pump closure bolting in Table 3.1.2.4, line item 2. The reactor water recirculation pump closure bolting is inspected in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Category B-G-1. Results of these inspections are provided below.

Unit 1 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-NUTS-0001A	5	VT	ACCEPTABLE
RECIRPMP-NUTS-0001B	5	VT	ACCEPTABLE
RECIRPMP-STUD-0001A	4	UT	ACCEPTABLE
RECIRPMP-STUD-0001A	5	UT	ACCEPTABLE
RECIRPMP-STUD-0001B	4	UT	ACCEPTABLE
RECIRPMP-STUD-0001B	5	UT	ACCEPTABLE

Unit 2 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-A-FLG	6	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-01	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-01	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-02	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-02	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-03	6A	VT	ACCEPTABLE

Unit 2 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-A-NUT-2-03	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-04	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-04	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-05	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-05	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-06	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-06	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-07	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-07	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-08	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-08	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-09	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-09	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-10	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-10	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-11	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-11	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-12	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-12	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-13	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-13	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-14	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-14	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-15	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-15	10	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-16	6A	VT	ACCEPTABLE
RECIRPMP-A-NUT-2-16	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-01	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-01	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-01	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-02	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-02	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-02	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-03	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-03	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-03	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-04	6A	UT	ACCEPTABLE

Unit 2 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-A-STUD-2-04	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-04	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-05	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-05	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-05	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-06	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-06	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-06	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-07	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-07	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-07	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-08	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-08	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-08	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-09	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-09	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-09	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-10	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-10	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-10	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-11	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-11	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-11	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-12	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-12	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-12	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-13	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-13	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-13	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-14	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-14	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-14	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-15	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-15	10	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-15	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-16	6A	UT	ACCEPTABLE
RECIRPMP-A-STUD-2-16	10	UT	ACCEPTABLE

Unit 2 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-A-STUD-2-16	10	VT	ACCEPTABLE
RECIRPMP-A-STUD-2-16	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-01	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-01	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-02	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-02	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-03	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-03	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-04	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-04	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-05	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-05	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-06	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-06	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-07	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-07	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-08	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-08	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-09	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-09	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-10	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-10	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-11	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-11	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-12	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-12	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-13	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-13	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-14	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-14	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-15	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-15	10	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-16	6A	VT	ACCEPTABLE
RECIRPMP-A-WASH-2-16	10	VT	ACCEPTABLE
RECIRPMP-B-FLG	6	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-01	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-02	6A	VT	ACCEPTABLE

Unit 2 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-B-NUT-2-03	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-04	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-05	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-06	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-07	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-08	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-09	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-10	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-11	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-12	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-13	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-14	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-15	6A	VT	ACCEPTABLE
RECIRPMP-B-NUT-2-16	6A	VT	ACCEPTABLE
RECIRPMP-B-STUD-2-01	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-02	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-03	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-04	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-05	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-06	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-07	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-08	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-09	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-10	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-11	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-12	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-13	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-14	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-15	6A	UT	ACCEPTABLE
RECIRPMP-B-STUD-2-16	6A	UT	ACCEPTABLE
RECIRPMP-B-WASH-2-01	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-02	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-03	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-04	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-05	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-06	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-07	6A	VT	ACCEPTABLE

Unit 2 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-B-WASH-2-08	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-09	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-10	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-11	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-12	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-13	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-14	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-15	6A	VT	ACCEPTABLE
RECIRPMP-B-WASH-2-16	6A	VT	ACCEPTABLE

Unit 3 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-A-NUT-3-01	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-02	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-03	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-04	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-05	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-06	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-07	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-08	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-09	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-10	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-11	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-12	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-13	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-14	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-15	11	VT	ACCEPTABLE
RECIRPMP-A-NUT-3-16	11	VT	ACCEPTABLE
RECIRPMP-A-STUD-3-01	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-02	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-03	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-04	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-05	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-06	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-07	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-08	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-09	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-10	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-11	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-12	11	UT	ACCEPTABLE

Unit 3 Category B-G-1 Examinations

Component ID	Cycle	Examination Method	Results
RECIRPMP-A-STUD-3-13	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-14	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-15	11	UT	ACCEPTABLE
RECIRPMP-A-STUD-3-16	11	UT	ACCEPTABLE
RECIRPMP-A-WASH-3-01	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-02	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-03	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-04	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-05	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-06	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-07	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-08	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-09	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-10	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-11	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-12	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-13	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-14	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-15	11	VT	ACCEPTABLE
RECIRPMP-A-WASH-3-16	11	VT	ACCEPTABLE

Based on this review, no repairs have been performed on the reactor recirculation pump closure bolting. As discussed in Appendix B.2.1.16, EPRI NP-5769 and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting have been implemented at BFN.

The plant-specific experience related to reactor recirculation pump closure bolting has no impact on the attributes specified in AMP-B.2.1.16.

TVA RESPONSE TO NRC RAI-3.1.2.4-1(B)

Fatigue was identified to be an aging effect that requires management for the period of extended operation for the reactor water recirculation pump seal flange closure bolting in Table 3.1.2.4, line item 3. A review of plant specific operating experience identified no instances where bolting failure due to fatigue occurred within the scope of license renewal. Fatigue is addressed as a TLAA at BFN and is addressed in Section 4.3, Metal Fatigue. The confirmation Program for the Metal Fatigue TLAA is the B.3.2 Fatigue Monitoring Program described in

Appendix B.3.2. Table 3.1.2.4, line item 3 is revised to include "3.1.1.1". in the 8th column, "Table 1 Item," and an "A" in the 9th column, "Notes".

Time Limited Aging Analysis (TLAA)

NRC RAI 4.2.1-1

Provide the values of upper shelf energy (USE) for the end of the extended period of operation and power uprate condition, percent reduction in USE, percentage of copper, and 1/4 T fluence at the end of the extended period of operation and power uprate condition for all the plates and weld metals in the beltline region of BFN Units 1, 2, and 3.

TVA RESPONSE TO NRC RAI 4.2.1-1

Upper shelf energy values are not available for the Browns Ferry (BFN) units; however, BFN has used the Equivalent Margin Analysis method [References 1 and 2] to demonstrate that the BFN vessels will maintain adequate fracture toughness throughout the extended period of operation. The EPU bounding value for Effective Full Power Years (EFPY) for Unit 1 is 54 EFPY and for Units 2 and 3 is 52 EFPY. The attached Tables 4.2.1-1 through 4.2.1-3 include all beltline materials for BFN Units 1, 2, and 3; all materials meet the EMA acceptance criteria of 23.5% and 39% for plates and welds, respectively [Reference 3].

References:

1. J.T. Wiggins (NRC) to L.A. England (Gulf States Utilities Co.), "Acceptance for Referencing of Topical Report NEDO-32205, Revision 1, '10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels'", December 8, 1993.

2. L.A. England (BWR Owners' Group) to Daniel G. McDonald (USNRC), "BWR Owners' Group Topical Report on Upper Shelf Energy Equivalent Margin Analysis – Approved Version", BWROG-94037, March 21, 1994.

3. C.I. Grimes (NRC) to Carl Terry (Niagara Mohawk Power Company), "Acceptance for Referencing Of EPRI Proprietary Report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)" And Appendix A, "Demonstration of

Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)", October 18, 2001.

Table 4.2.1-1
BFN Unit 1 Beltline EMA

Component	Heat or Heat/Lot	%Cu	1/4 T Fluence n/cm^2	USE % decrease (RG 1.99)	Equivalent Margin Analysis (EMA)%	Acceptable Margin? (Yes/No)
PLATES:						
Lower Shell						
6-127-1	A0999-1	0.14	1.35E+18	14.5	23.5	Y
6-127-2	B5864-1	0.15	1.35E+18	15.5	23.5	Y
6-127-4	A1009-1	0.14	1.35E+18	14.5	23.5	Y
Lower-Intermediate Shell						
6-139-19	C2884-2	0.12	1.66E+18	14	23.5	Y
6-139-20	C2868-2	0.09	1.66E+18	12	23.5	Y
6-139-21	C2753-1	0.08	1.66E+18	11	23.5	Y
WELDS:						
Axial	ESW	0.24	1.66E+18	25.5	39	Y
Girth	406L44	0.27	1.35E+18	26.5	39	Y

Table 4.2.1-2
BFN Unit 2 Beltline EMA

Component	Heat or Heat / Lot	%Cu	1/4 T Fluence n/cm^2	USE % decrease (RG 1.99)	Equivalent Margin Analysis (EMA)%	Acceptable Margin? (Yes/No)
PLATES:						
Lower Shell						
6-127-14	C2467-2	0.16	1.29E+18	16	23.5	Y
6-127-15	C2463-1	0.17	1.29E+18	16.5	23.5	Y
6-127-17	C2460-2	0.13	1.29E+18	14	23.5	Y
Lower-Intermediate Shell						
6-127-6	A0981-1	0.14	1.59E+18	15.5	23.5	Y
6-127-16	C2467-1	0.16	1.59E+18	16	23.5	Y
6-127-20	C2849-1	0.11	1.59E+18	13	23.5	Y
WELDS:						
Axial	ESW	0.24	1.59E+18	25.5	39	Y
Girth	D55733	0.09	1.29E+18	14.5	39	Y

Table 4.2.1-3
BFN Unit 3 Beltline EMA

Component	Heat or Heat/Lot	%Cu	1/4 T Fluence n/cm^2	USE % decrease (RG 1.99)	Equivalent Margin Analysis (EMA)%	Acceptable Margin? (Yes/No)
Plates:						
Lower Shell						
6-145-4	C3222-2	0.15	1.29E+18	15	23.5	Y
6-145-7	C3213-1	0.13	1.29E+18	14	23.5	Y
6-145-12	C3217-2	0.14	1.29E+18	14.5	23.5	Y
Lower-Intermediate Shell						
6-145-1	C3201-2	0.13	1.59E+18	14.5	23.5	Y
6-145-2	C3188-2	0.10	1.59E+18	13	23.5	Y
6-145-6	B7267-1	0.13	1.59E+18	14.5	23.5	Y
Welds:						
Axial	ESW	0.24	1.59E+18	25.5	39	Y
Girth	D55733	0.09	1.29E+18	14.5	39	Y

NRC RAI 4.2.2-1

Table 4.2.2.1: It is stated in the submittal that the NRC - approved fluence method was used to calculate bounding fluence values for BFN Units 1, 2, and 3 for 54, 52 and 52 effective full power years (EFPYs) of the operation, respectively.

(A) However, the values in Table 4.2.2.1 seem to be inconsistent. How can BFN Unit 1 achieve 54 EFPYs in a 60 year span given its operating history? Why does BFN Unit 1 have a peak surface fluence value of 1.95×10^{18} n/cm^2 (E > 1.0 MeV) while Units 2 and 3 achieve 2.3×10^{18} n/cm^2 (E > 1.0 MeV)?

(B) Provide the initial RTNDT, and ART values at 1/4 T and vessel ID surface, at the end of the extended period of the operation for BFN Units 1, 2, and 3 for all the materials in the beltline region of the BFN reactor vessels.

TVA RESPONSE TO NRC RAI 4.2.2-1(A)

For BFN Unit 1, 54 EFPY was selected as a bounding value as part of the Extended Power Uprate (EPU) evaluation. For consistency with the EPU evaluation, the 54 EFPY value was incorporated into the License Renewal Application. The reason the report peak fluence for BFN Unit 1 is lower than Units 2 and 3 is because the maximum ΔRT_{NDT} and ART occurs in the girth weld material, which is located away from the peak vessel fluence location, whereas both Units 2 and 3 maximum ΔRT_{NDT} and ART occurs in the axial weld materials. Therefore, the reported peak fluence for Unit 1 has an axial correction factor of 0.81 applied and Units 2 and 3 do not have the axial correction factor of 0.81 applied. See attached Tables 4.2.2-1 through 4.2.2-6.

TVA RESPONSE TO NEC RAI 4.2.2-1(B)

See attached Tables 4.2.2-1 through 4.2.2-6. The Adjusted Reference Temperature values were calculated for 54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3).

Table 4.2.2-1
BFN Unit 1 Adjusted Reference Temperatures (ID)
Lower-Intermediate Plates and Axial Welds

Thickness = 6.13 inches Ratio Peak/Location = 1.00
54 EFPY Peak I.D. fluence = 2.40E+18 n/cm^2

Lower Plates and Axial Welds & Lower to Lower-Intermediate Girth Weld

Thickness = 6.13 inches Ratio Peak/Location = 0.81
54 EFPY Peak I.D. fluence = 1.95E+18 n/cm^2

Component	Heat or Heat/Lot	%Cu	%Ni	CF	Initial RT_{NDT} °F	ID Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower Shell												
6-127-1												
6-127-2	A0999-1	0.14	0.60	100	-20	1.95E+18	56	0	17	34	90	70
6-127-4	B5864-1	0.15	0.44	101	-20	1.95E+18	57	0	17	34	91	71
	A1009-1	0.14	0.50	96	-10	1.95E+18	54	0	17	34	88	78
Lower-Intermediate Shell												
6-139-19	C2884-2	0.12	0.53	82	14	2.40E+18	50	0	17	34	84	98
6-139-20	C2868-2	0.09	0.48	58	30	2.40E+18	36	0	17	34	70	100
6-139-21	C2753-1	0.08	0.50	51	2	2.40E+18	31	0	16	31	63	65
WELDS:												
Axial	ESW	0.24	0.37	141	23.1	2.40E+18	87	13	28	62	148	171
Girth	406L44	0.27	0.60	184	20	1.95E+18	104	10	28	59	163	183

Table 4.2.2-2
BFN Unit 1 Adjusted Reference Temperatures (1/4T)
Lower-Intermediate Plates and Axial Welds

Thickness = 6.13 inches Ratio Peak/Location = 1.00
54 EFPY Peak 1/4 T fluence = 1.66E+18 n/cm^2

Lower Plates and Axial Welds & Lower to Lower-Intermediate Girth Weld

Thickness = 6.13 inches Ratio Peak/Location = 0.81
54 EFPY Peak 1/4 T fluence = 1.35E+18 n/cm^2

Component	Heat or Heat/Lot	%Cu	%Ni	CF	Initial RT_{NDT} °F	1 / 4 T Fluence n/cm^2	54 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower Shell												
6-127-1	A0999-1	0.14	0.60	100	-20	1.35E+18	48	0	17	34	82	62
6-127-2	B5864-1	0.15	0.44	101	-20	1.35E+18	48	0	17	34	82	62
6-127-4	A1009-1	0.14	0.50	96	-10	1.35E+18	46	0	17	34	80	70
Lower-Intermediate Shell												
6-139-19	C2884-2	0.12	0.53	82	14	1.66E+18	43	0	17	34	77	91
6-139-20	C2868-2	0.09	0.48	58	30	1.66E+18	31	0	15	31	61	91
6-139-21	C2753-1	0.08	0.50	51	2	1.66E+18	27	0	13	27	54	56
WELDS:												
Axial	ESW	0.24	0.37	141	23.1	1.66E+18	74	13	28	62	136	159
Girth	406L44	0.27	0.60	184	20	1.35E+18	88	10	28	59	148	168

Table 4.2.2-3
BFN Unit 2 Adjusted Reference Temperatures (ID)
Lower-Intermediate Plates and Axial Welds

Thickness = 6.13 inches Ratio Peak/Location = 1.00
52 EFPY Peak I.D. fluence = 2.30E+18 n/cm^2

Lower Plates and Axial Welds & Lower to Lower-Intermediate Plates Girth Weld

Thickness = 6.13 inches Ratio Peak/Location = 0.81
52 EFPY Peak I.D. fluence = 1.86E+18 n/cm^2

Component	Heat or Heat/Lot	%Cu	%Ni	CF	Initial RT_{NDT} °F	ID Fluence n/cm^2	52 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	52 EFPY Shift °F	52 EFPY ART °F
PLATES:												
Lower Shell												
6-127-14	C2467-2	0.16	0.52	112	-20	1.86E+18	62	0	17	34	96	76
6-127-15	C2463-1	0.17	0.48	117	-20	1.86E+18	65	0	17	34	99	79
6-127-17	C2460-2	0.13	0.51	88	0	1.86E+18	49	0	17	34	83	83
Lower-Intermediate Shell												
6-127-6	A0981-1	0.14	0.55	98	-10	2.30E+18	59	0	17	34	93	83
6-127-16	C2467-1	0.16	0.52	112	-10	2.30E+18	68	0	17	34	102	92
6-127-20	C2849-1	0.11	0.50	73	-10	2.30E+18	44	0	17	34	78	68
WELDS:												
Axial	ESW	0.24	0.37	141	23.1	2.30E+18	85	13	28	62	147	170
Girth	D55733	0.09	0.65	117	-40	1.86E+18	65	0	28	56	121	81

Table 4.2.2-4
BFN Unit 2 Adjusted Reference Temperatures (1/4T)
Lower-Intermediate Plates and Axial Welds

Thickness = 6.13 inches Ratio Peak/Location = 1.00
52 EFPY Peak 1/4 T fluence = 1.59E+18 n/cm^2

Lower Plates and Axial Welds & Lower to Lower-Intermediate Girth Weld

Thickness = 6.13 inches Ratio Peak/Location = 0.81
52 EFPY Peak 1/4 T fluence = 1.29E+18 n/cm^2

Component	Heat or Heat/Lot	%Cu	%Ni	CF	Initial RT_{NDT} °F	1 / 4 T Fluence n/cm^2	52 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	52 EFPY Shift °F	52 EFPY ART °F
PLATES:												
Lower Shell												
6-127-14	C2467-2	0.16	0.52	112	-20	1.29E+18	53	0	17	34	87	67
6-127-15	C2463-1	0.17	0.48	117	-20	1.29E+18	55	0	17	34	89	69
6-127-17	C2460-2	0.13	0.51	88	0	1.29E+18	41	0	17	34	75	75
Lower-Intermediate Shell												
6-127-6	A0981-1	0.14	0.55	98	-10	1.59E+18	51	0	17	34	85	75
6-127-16	C2467-1	0.16	0.52	112	-10	1.59E+18	58	0	17	34	92	82
6-127-20	C2849-1	0.11	0.50	73	-10	1.59E+18	38	0	17	34	72	62
WELDS:												
Axial	ESW	0.24	0.37	141	23.1	1.59E+18	73	13	28	62	134	157
Girth	D55733	0.09	0.65	117	-40	1.29E+18	55	0	27	55	110	70

Table 4.2.2-5
BFN Unit 3 Adjusted Reference Temperatures (ID)
Lower-Intermediate Plates and Axial Welds

Thickness = 6.13 inches Ratio Peak/Location = 1.00
52 EFPY Peak I.D. fluence = 2.30E+18 n/cm^2

Lower Plates and Axial Welds & Lower to Lower-Intermediate Girth Weld

Thickness = 6.13 inches Ratio Peak/Location = 0.81
52 EFPY Peak I.D. fluence = 1.86E+18 n/cm^2

Component	Heat or Heat/Lot	%Cu	%Ni	CF	Initial RT_{NDT} °F	ID Fluence n/cm^2	52 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	52 EFPY Shift °F	52 EFPY ART °F
PLATES:												
Lower Shell												
6-145-4	C3222-2	0.15	0.52	106	10	1.86E+18	59	0	17	34	93	103
6-145-7	C3213-1	0.13	0.58	90	-20	1.86E+18	50	0	17	34	84	64
6-145-12	C3217-2	0.14	0.66	101.5	-4	1.86E+18	56	0	17	34	90	86
Lower-Intermediate Shell												
6-145-1	C3201-2	0.13	0.60	91	-20	2.30E+18	55	0	17	34	89	69
6-145-2	C3188-2	0.10	0.48	65	-20	2.30E+18	39	0	17	34	73	53
6-145-6	B7267-1	0.13	0.51	88	-20	2.30E+18	53	0	17	34	87	67
WELDS:												
Axial	ESW	0.24	0.37	141	23.1	2.30E+18	85	13	28	62	147	170
Girth	D55733	0.09	0.66	117	-40	1.86E+18	65	0	28	56	121	81

Table 4.2.2-6
BFN Unit 3 Adjusted Reference Temperatures (1/4T)
Lower-Intermediate Plates and Axial Welds

Thickness = 6.13 inches Ratio Peak/Location = 1.00
52 EFPY Peak 1/4 T fluence = 1.59E+18 n/cm^2

Lower Plates and Axial Welds & Lower to Lower-Intermediate Plates Weld

Thickness = 6.13 inches Ratio Peak/Location = 0.81
52 EFPY Peak 1/4 T fluence = 1.29E+18 n/cm^2

Component	Heat or Heat/Lot	%Cu	%Ni	CF	Initial RT_{NDT} °F	1 / 4 T Fluence n/cm^2	52 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	52 EFPY Shift °F	52 EFPY ART °F
PLATES:												
Lower Shell												
6-145-4	C3222-2	0.15	0.52	106	10	1.29E+18	50	0	17	34	84	94
6-145-7	C3213-1	0.13	0.58	90	-20	1.29E+18	42	0	17	34	76	56
6-145-12	C3217-2	0.14	0.66	101.5	-4	1.29E+18	48	0	17	34	82	78
Lower-Intermediate Shell												
6-145-1	C3201-2	0.13	0.60	91	-20	1.59E+18	47	0	17	34	81	61
6-145-2	C3188-2	0.10	0.48	65	-20	1.59E+18	34	0	17	34	67	47
6-145-6	B7267-1	0.13	0.51	88	-20	1.59E+18	45	0	17	34	79	59
WELDS:												
Axial	ESW	0.24	0.37	141	23.1	1.59E+18	73	13	28	62	134	157
Girth	D55733	0.09	0.66	117	-40	1.29E+18	55	0	27	55	110	70

NRC RAI 4.2.4-1

(A) In LRA Section 4.2.4, "Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware," the applicant states that the total integrated neutron flux at the end of license at the shroud inside surface is expected to be 5.34 x 10^{21} n/cm^2 (E > 1 MeV). The staff requests that the applicant provide an explanation whether this value is bounding at the inside shroud surface for Units 1, 2 and, 3. If so, submit information whether the neutron fluence values are estimated based on the implementation of the power uprate.

(B) In LRA Section 4.2.4, the applicant states that the maximum 54 EFPY fluence at the inside surface of the core shroud is 5.34 x 10^{21} n/cm^2. Since this fluence is greater than the irradiation assisted stress corrosion cracking (IASCC) threshold fluence [5 x 10^{20} n/cm^2 (E > 1.0 MeV)], identify the AMP to monitor IASCC for the core shroud. In addition, these welds are also prone to IGSCC. Please provide plant-specific aging management program for core shroud welds at the BFN Units based on the following attributes:

(a) Type of material (i.e., 304 or 304L)
(b) Hot operating time
(c) Conductivity
(d) Fabrication features

Provide information on the type and the extent of inspection on core shroud welds.

(C) The applicant calculated thermal strain resulting from the low-pressure coolant injection reflood thermal shock transient in the core shroud region. The applicant compared the calculated thermal strain with the measured values of per cent elongation of annealed type 304 stainless steel irradiated to 8×10^{21} n/cm^2 (E > 1.0 MeV). In a previous analysis performed by Dresden/Quad Cities, the applicant used the percent reduction in area as a criterion to evaluate the thermal strain. The staff requests that the applicant for BFN units, provide information on the measured percent reduction in area values for the irradiated type 304 stainless steel. The applicant should compare the results of the analysis obtained from using the reduction in area, with the ones using the percent elongation, and justify which of these properties is more appropriate to use in evaluating the local thermal shock strain associated with the reflood thermal shock event at the most irradiated core shroud region.

TVA RESPONSE TO NRC RAI 4.2.4-1(A)

The calculation of shroud fluence, 5.34×10^{21} n/cm^2 (E > 1 MeV), is based upon a shroud inner diameter peak flux of 3.14×10^{12} n/cm^2-sec for 54 EFPY, the lifetime used for Unit 1 evaluations. Since Units 2 and 3 have a projected lifetime of 52 EFPY, the 5.34×10^{21} n/cm^2 (E > 1 MeV) fluence from Unit 1 is bounding for all three units. The fluence value for the shroud inner diameter was estimated based upon the implementation of extended power uprate.

The following is appended to the disposition statement of LRA Section 4.2.2.5:

"The Unit 1 fluence of 5.34×10^{21} n/cm^2 is bounding for all three BFN Units."

TVA RESPONSE TO NRC RAI 4.2.4-1(B)

IASCC of the core shroud was evaluated in LRA Section 4.7.6. The stress corrosion cracking identified as an aging effect for reactor vessel core shroud and core plate component types in LRA Table 3.1.2.2 includes IASCC and IGSCC. The following note from the aging management review was inadvertently omitted from the LRA: "The SCC aging mechanism includes intergranular SCC and irradiation assisted SCC." The Browns Ferry Nuclear Plant (BFN) Aging Management Programs that manages IASCC and IGSCC for the core shroud are BWR Vessel Internals and Water Chemistry. The BFN core shrouds are classified as "Category C" based on the core shroud classification criteria contained in Appendix B of BWRVIP-76. The BFN BWR Vessel Internals Aging Management Program requires inspection of core shroud welds in accordance with "Category C" core shroud inspection requirements contained in BWRVIP-76.

TVA RESPONSE TO NRC RAI 4.2.4-1(C)

Reduction in area and elongation values for irradiated stainless steel are as follows:

Reduction in Area

Fluence (n/cm^2, E>1MeV)	Test Temperature (°F)	Reduction in Area (%)	Reference
1×10^{21}	550	40	1
6.9×10^{21}	750	52.5	2

Elongation

Material	Fluence (n/cm^2, E>1MeV)	Test Temperature (°F)	Elongation (%)	Reference
Base	8×10^{21}	554	20	3
Weld	8×10^{21}	567	4	3

Since the bounding shroud fluence (BFN Unit 1) is 5.34×10^{21} n/cm^2 (E>1 MeV), the listed ductility values bound all three BFN shrouds. Reduction in area is significantly less affected by irradiation than elongation, as shown by the above tables. As described in LRA Section 4.2.4, the maximum thermal shock stress results in a calculated thermal shock strain amplitude of 0.57%. Both reduction in area and elongation values are significantly

in excess of the calculated thermal shock strain at the most irradiated location.

While the analysis indicates that either measure of ductility is acceptable for the period of extended operation, reduction in area is a more appropriate measure of ductility for the reflood thermal shock event. The strain associated with the reflood thermal shock event is very localized and is constrained by the surrounding bulk material. As such, it is similar to the triaxial stress condition present in the neck region (where the area reduction is taking place) during a tensile test. The percent reduction in area is a measure of this triaxial stress state and, as such, is the most appropriate property for evaluating the effect of thermal shock on the shroud.

References:

1. "The Effects of Radiation on Structural Materials," ASTM Special Technical Publication No. 426, ASTM, Philadelphia, Pa., 1966, pages 278-327.

2. L.A. Waldman and M. Doumas, "Fatigue and Burst Tests on Irradiated In-Pile Stainless Steel Pressure Tubes," Nuclear Applications, Vol. 1, October 1965.

3. "Fracture Toughness and Tensile Properties of Irradiated Austenitic Stainless Steel Components Removed from Service," EPRI TR-108279 (BWRVIP-35), EPRI, Palo Alto CA, June 1997 (EPRI Proprietary Information).

NRC RAI 4.2.6-1

The reactor vessel circumferential weld examination relief analyses satisfy the requirements of 10 CFR 54.3(a), and the analyses are considered a TLAA. In Section 4.2.6 of the LRA, an evaluation for the reactor vessel circumferential weld examination relief for Unit 1 was not provided. By letter dated May 12, 2004, the applicant submitted a relief request, whereby the applicant requested relief from the reactor vessel circumferential weld examination for the current license period for BFN Unit 1. The staff is currently reviewing this request for the current license period. However, the staff requests that the applicant provide the reactor vessel circumferential weld examination relief analyses for BFN Unit 1 for the extended licensed operating period.

TVA RESPONSE TO NRC RAI 4.2.6-1

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis are: 1) the neutron fluence was the estimated end-of-life mean fluence, 2) the chemistry values are mean values based on vessel types, and 3) the potential for beyond-design-basis events is considered. The following table provides a comparison of the BFN Unit 1 reactor vessel limiting circumferential weld parameters to those used in the NRC evaluation of BWRVIP-05 for the first two key assumptions. Data provided in this table was supplied from Tables 2.6.4 and 2.6.5 of the Final Safety Evaluation of the BWRVIP-05 Report (NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, "Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report," (TAC No. M93925), July 28, 1998)

Effects of Irradiation on RPV Circumferential Weld Properties BFN Unit 1

Group	B&W 64 EFPY	BFN Unit 1 54 EFPY
Cu%	0.31	0.27
Ni%	0.59	0.60
CF	196.7	184
Fluence at clad/weld interface (10^{19} n/cm^2)	0.19	0.2
ΔRT_{NDT} w/o margin (°F)	109.4	104
$RT_{NDT(U)}$ (°F)	20	20
Mean RT_{NDT} (°F)	129.4	124
P(F/E) NRC	4.83×10^{-4}	---
P(F/E) BWRVIP	---	---

The fluence assumed for Unit 1 is very conservative based on an extended shutdown period from 1985 to a scheduled restart in 2007, which will result in less than 32 EFPY of vessel exposure through the end of the extended period of operation. However, TVA conservatively chose to use the higher exposure of 54 EFPY to simplify the basis for the Unit 1 vessel evaluations. As shown in the table, the Unit 1 unirradiated weld RT_{NDT} is identical to the reference B&W plant unirradiated weld RT_{NDT} used in the NRC analysis, and the Unit 1 fluence value is approximately equivalent to that used in the NRC analysis. However, because the Unit 1 chemistry factor is less than the

reference B&W plant, the mean RT_{NDT} values for Unit 1 at 54 EFPY are bounded by the 64 EFPY Mean RT_{NDT} assumed by the NRC in its analysis. Accordingly, Unit 1 is bounded by the conditional failure probability calculated by the Staff for the limiting B&W vessel. An extension of this relief for the 60-year period will be submitted to the NRC for approval prior to entering the period of extended operation.

NRC RAI 4.2.7-1

The reactor vessel axial weld failure probability analyses satisfy the requirements of 10 CFR 54.3(a), and the analyses are considered a TLAA. In Section 4.2.7 of the LRA, an evaluation for reactor vessel axial weld failure probability analyses for Unit 1 was not provided. The staff requests that the applicant provide the reactor vessel axial weld failure probability analyses for BFN Unit 1 for the extended licensed operating period.

TVA RESPONSE TO NRC RAI 4.2.7-1

The table provided below compares the limiting axial weld 54 EFPY properties for Unit 1 against the values taken from Table 2.6.5 found in the NRC SER for BWRVIP-05 and associated supplement to the SER (NRC letter from Jack R. Strosnider, to Carl Terry, BWRVIP Chairman, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," (TAC No. MA3395), March 7, 2000). The SER supplement required the limiting axial weld to be compared with data found in Table 3 of the document. For Unit 1 the comparison was made to the 'Mod 2' plant information. The supplemental SER stated that the 'Mod 2' calculations most closely match the 5×10^{-6} RPV failure frequency.

Effects of Irradiation on RPV Axial Weld Properties BFN Unit 1

Value	NRC BWRVIP-05 SER Mod 2	BFN Unit 1 54 EFPY
Cu%	0.219	0.24
Ni%	0.996	0.37
CF	--	141
Fluence $\times 10^{19}$ n/cm^2	0.148 (Peak Axial Fluence)	0.24
ΔRTN_{DT} (°F)	116	86
$RTN_{DT(U)}$ (°F)	-2	23
Mean RT_{NDT} (°F)	114	109

Value	NRC BWRVIP-05 SER Mod 2	BFN Unit 1 54 EFPY
P(F/E)	5.02×10^{-6}	Not Calculated

The limiting axial weld is an electroslag weld with similar chemistry. The Unit 1 limiting weld chemistry, chemistry factor, and 54 EFPY mean RT_{NDT} values are within the limits of the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 SER supplement and the 64 EFPY limits and values obtained from Table 2.6.5 of the SER. Therefore, the probability of failure for the axial welds is bounded by the NRC evaluation.

Aging Management Programs (AMPs)

Reactor Head Closure Stud Program

NRC RAI B.2.1.6-1

(A) The applicant states in LRA AMP-B 2.1.6, "Reactor Head Closure Stud Program," that the reactor head studs at BFN Units 1, 2, and 3 are not metal plated. Provide information on the type of corrosion protection coating that was applied to these studs. Submit plant-specific experience regarding any type of degradation of these studs and any AMP that is required to maintain their integrity.

(B) The staff reviewed the UFSAR supplement (A.1.6) to determine whether it provides an adequate description of the program. The UFSAR supplement should be revised to indicate the type of inspections that are to be used for detecting loss of material and cracking in the reactor head closure studs.

TVA RESPONSE TO NRC RAI-B.2.1.6-1(A)

The operating experience review performed during the aging management review process identified no instances of reactor head closure stud degradation. The only instance of any damage to reactor closure studs occurred during the Unit 2 Cycle 4 refueling outage. Two studs (22 and 24) were identified as having damage that resulted from personnel handling and damaging the threads on the studs; the damage was not service induced. These two studs were subsequently replaced prior to exiting the refueling outage.

According to GE specification 21A1111, the reactor head closure studs were shipped with a corrosion inhibiting material applied to the stud surfaces. However, no corrosion protection coating is applied to the studs during operation of the reactor vessels. During outage conditions (disassembly and reassembly), an approved lubricant such as Neolube No. 2 is utilized. The Neolube No. 2 lubricant is a carbon/graphite material, which assists in protecting the closure studs. The Neolube No. 2 lubricant is used in nuclear applications and has very low contaminants.

The aging management program for potential reactor vessel closure studs aging effects is the Reactor Head Closure Studs Program that is described in Appendix B.2.1.6.

TVA RESPONSE TO NRC RAI-B.2.1.6-1(B)

The UFSAR supplement is intended to be a program summary description. The UFSAR supplement references American Society of Mechanical Engineers B&PV Code Section XI Subsection IWB Table IWB 2500-1. The types of inspections are contained in Table IWB 2500-1.

Boiling Water Reactor Feedwater Nozzle Program

NRC RAI B.2.1.8-1

The BWR Feed Water Nozzle AMP references GE report GE-NE-523-A71-0594, which is not the NRC-approved version of the report. Confirm that the applicant will implement the recommendations of Revision 1, Version A of the report (GE-NE-523-A71-0594-A, Revision 1) which is approved by the staff.

TVA RESPOSE TO NRC RAI-B.2.1.8-1

The BWR Feedwater Nozzle program will implement feedwater nozzle inspection recommendations based on GE Report GE-NE-523-A71-0594-A Revision 1. Replace references to GE-NE-523-A71-0594 in LRA Appendix A.1.8 and Appendix B.2.1.8 with GE-NE-523-A71-0594-A Revision 1.

Boiling Water Reactor Control Rod Drive Return Line Nozzle Program

NRC RAI B.2.1.9-1

The control rod drive (CRD) return line nozzle has been capped, and therefore augmented inspection for the nozzle is not required per NUREG-0619. The requirements in NUREG-0619 provide actions to be taken to address cracking in these nozzles. However, the aging effects for the cap and applicable weld are not covered in NUREG-0619. Therefore, the staff requests the following concerning the cap and weld which provides a pressure boundary function:

(1) Describe the configuration, location and material of construction of the capped nozzle. This should include the existing base material for the nozzle, piping (if piping remnants exist) and cap material, and any welds.

(2) Describe how this weld and cap is managed in accordance with the guidelines of BWRVIP-75.

(3) Discuss whether the event at Pilgrim (leaking weld at capped nozzle, September 30, 2003) is applicable to BFN units. The staff issued Information Notice 2004-08, dated April 22, 2004, which states that the cracking occurred in 82/182 weld that was previously repaired extensively. Discuss any plant experience with previous leakage at the capped nozzle. Include in your discussion the past inspection techniques applied, the results obtained, and mitigative strategies imposed. Provide information as to how the plant-specific experience related to this aging effect impacts the attributes specified in AMP-B.2.1.9.

TVA RESPONSE TO NRC RAI-B.2.1.9-1(1)

At BFN Units 1, 2, and 3 the configuration consists of a stainless steel cap welded to the original carbon steel nozzle. The weld material is stainless steel. The safe end and corresponding piping were removed from the nozzle.

TVA RESPONSE TO NRC RAI-B.2.1.9-1(2)

The requirements of BWRVIP-75 are implemented by the BWR Stress Corrosion Cracking Program. The control rod drive return line nozzles welds are currently categorized as Category D for Unit 2 and Category C for Unit 3. The control rod drive return line

nozzles welds are examined by the UT technique at the frequency specified by BWRVIP-75, Table 3-1 for normal water chemistry conditions.

As stated in Appendix B.2.1.10 of the License Renewal Application, the BWR Stress Corrosion Program will be implemented on Unit 1 prior to the period of extended operation.

TVA RESPONSE TO NRC RAI-B.2.1.9-1(3)

The event at Pilgrim was determined to be not applicable to the BFN units. According to NRC IN 2004-08, the Pilgrim reactor pressure vessel nozzle is made of SA-508, Class 2 low-alloy steel, while the CRD return line cap is made of Alloy 600. The subject weld is fabricated with Alloy 82/182 material, and the nozzle side of the weld is buttered with Alloy 182 material. The materials of construction of the nozzle to cap weld at BFN are different and are described in the response to RAI B.2.1.9-1(1). In addition, Pilgrim had initial weld deficiencies (lack of fusion) that required weld repair. The BFN welds were completed without recordable indications.

Plant experience for Unit 2 and Unit 3 indicate that there has been no leakage at the capped CRD return line nozzle. Ultrasonic exams have been performed with no reportable indications. The Unit 3 capped CRD return line nozzle weld had MSIP performed to mitigate IGSCC, which changed the frequency of inspection. Refer to examination information in RAI-B.2.1.9-1(2).

The plant-specific experience related to control rod drive return line nozzle has no impact on the attributes specified in AMP-B.2.1.9, Boiling Water Reactor Control Rod Drive Return Line Nozzle Program.

Boiling Water Stress Corrosion Cracking Program

NRC RAI B.2.1.10-1

Intergranular Stress Corrosion Cracking (IGSCC) of Stainless Steel and Inconel materials of the Reactor Pressure Vessel (RPV) internals:

> The applicant credits BWR water chemistry program (AMP B.2.1.5), and Inservice Inspection Program (AMP B.2.1.4) for managing crack initiation and growth due IGSCC in stainless steel and Inconel components for the following RPV systems:

(1) Reactor Vessel Attachments (treated water internal);

(2) Reactor Vessel Heads, Flanges and Shells (treated water internal);

(3) Reactor Vessel Nozzles (treated water internal);

(4) Reactor Vessel Internals Core Shroud and Core Plate – Inconel only (treated water internal);

(5) Control Rod Guide Tube, Control Rod Housing, Stub Tube, Incore Housing, Guide Tube and Dry Tube assemblies;

(6) Reactor Vents and Drains – Piping and Fittings;

(7) High Pressure Coolant Injection (HPCI), core spray, Reactor Coolant Incore Circulation (RCIC), Residual Heat Removal (RHR), Low Pressure Coolant Injection LPCI), Stand By Liquid Control (SBLC), Reactor Water Clean Up (RWCU), Main Steam (MS), and Feed Water (FW) systems.

    (A) Describe plant-specific experience related to IGSCC cracking of the stainless steel and Inconel components in the aforementioned systems.

    (B) Submit information on the mitigation actions taken at BFN with respect to selection of materials that are resistant to sensitization, use of special processes that reduce residual tensile stress and monitoring of water chemistry, such as discussed in NUREG-1801, Chapter XI.M7.

    (C) Provide information if any noble metal chemical application (NMCA) is applied at BFN. Confirm the method of controlling hydrogen water chemistry and any noble metal chemical application (NMCA) in the reactor vessel. Provide details on the methods for determining the effectiveness of hydrogen water chemistry and/or NMCA by using the following parameters:

        (1) Electro Chemical Potential (ECP)
        (2) Feedwater hydrogen flow
        (3) Main steam oxygen content
        (4) Hydrogen/oxygen molar ratio.

TVA RESPONSE TO NRC RAI-B.2.1.10-1(A)

As identified in LRA Section B.2.1.10-1, the Boiling Water Stress Corrosion Cracking Program will be implemented on Unit 1 prior to the period of extended operation. The following describes the current known extent of cracking at BFN.

PLANT-SPECIFIC EXPERIENCE RELATED TO IGSCC CRACKING

Component	Unit 2	Unit 3
Reactor Vessel Attachments	None identified	None identified
Reactor Vessel Heads, Flanges, and Shells	None identified	One flange surface indication due to mechanical damage.
Reactor Vessel Nozzles	None identified	None identified
Reactor Vessel Internals Core Shroud and Core Plate	CORE SHROUD H-1 H-2 H-3 H-5 H-6 H-7 CORE PLATE None identified	CORE SHROUD H-1 H-2 H-3 H-4 H-5 H-7 CORE PLATE None identified
Control Rod Guide Tube Control Rod Housing Control Rod Stub Tube Incore Housing Guide Tube and Dry Tube assemblies;	CR GUIDE TUBE None identified CR HOUSING None identified CR STUB TUBE None identified INCORE HOUSING None identified GUIDE TUBE/DRY TUBE None identified (original dry tubes replaced in 1991 due to cracking)	CR GUIDE TUBE None identified CR HOUSING None identified CR STUB TUBE None identified INCORE HOUSING None identified GUIDE TUBE/DRY TUBE None identified (original dry tubes replaced in 1994 due to cracking)
Reactor Vents and Drains	None identified	None identified
High Pressure Coolant Injection	HPCI None identified	HPCI None identified

PLANT-SPECIFIC EXPERIENCE RELATED TO IGSCC CRACKING

Component	Unit 2	Unit 3
Core Spray	CS	CS
Reactor Coolant Incore	TCS-2-421-(OL)	None identified
Circulation	RCIC	RCIC
Residual Heat Removal	None identified	None identified
Standby Liquid Control	RHR(LPSI)	RHR(LPSI)
Reactor Water Cleanup	DRHR-2-09	DSRHR-3-11(OL)
Main Steam	DRHR-2-22	
Feedwater	SLC	SLC
	None identified	None identified
	RWCU	RWCU
	DSRWC-2-03(OL)	None identified
	DSRWC-2-04(OL)	
	DSRWC-2-05(OL)	
	MS	MS
	None identified	None identified
	FW	FW
	None identified	None identified

TVA RESPONSE TO NRC RAI-B.2.1.10-1(B)

Mitigation efforts include selection of IGSCC resistant materials and monitoring/control of water chemistry parameters. The criteria for the design, installation, and testing associated with the replacement or removal of selected piping to limit the susceptibility to IGSCC is provided in General Design Criteria BFN-50-779, "Replacement of Selected Piping to Limit Susceptibility to IGSCC."

Monitoring and control of chemistry parameters is controlled by the Chemistry Control Program, which implements the guidance of BWRVIP-79, "EPRI-103515-R2, BWR Water Chemistry Guidelines – 2000 Revision."

TVA RESPONSE TO NRC RAI-B.2.1.10-1(C)

BFN currently utilizes NMCA (zinc addition) as part of the reactor water chemistry control program. BFN does not utilize ECP probes and, therefore, alternate means are used to monitor NMCA/HWC control. The acceptable alternate means are described in Section 5.4 of BWRVIP-79, "EPRI-103515-R2, BWR Water Chemistry Guidelines – 2000 Revision,"

BFN procedure CI-13.1, Chemistry Program, specifies a reactor water H_2/O_2 molar ratio of ≥4 for power operation. The

effectiveness of maintaining an adequate H_2/O_2 molar ratio, and thus ECP, is described in BWRVIP-79.

Boiling Water Reactor Internal Program

NRC RAI B.2.1.12-1

(A) Top Guide: BWRVIP-26, "Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines" BWRVIP-26 lists 5 x 10^{20} n/cm^2 (E > 1.0 MeV) as the threshold fluence beyond which components may be susceptible to IASCC. The location on the top guide that will see this high fluence is the grid beams. This is Location 1, as identified in BWRVIP-26, Table 3-2, "Matrix of Inspection Options." In its evaluation of the top guide assembly, including the grid beams, General Electric (GE) assumed a lower allowable stress value, acknowledging the high fluence value at this location. The conclusion of this analysis, and the fact that a single failure at this location has no safety consequence, was that no inspection was considered necessary to manage this potential aging effect. The staff is concerned that multiple failures of the top guide beams are possible when the threshold fluence for IASCC is exceeded. According to BWRVIP-26, multiple cracks have been observed in top guide beams at Oyster Creek. In addition, baffle-former bolts on PWRs that exceeded the threshold fluence have had multiple failures. In order to exclude the top guide beam from inspection when its fluence exceeds the threshold value, it must be demonstrated that failure of all beams that exceed the threshold fluence will not impact the safe shutdown of the reactor during normal, upset, emergency, and faulted conditions. If this can not be demonstrated, then an inspection program to manage this aging effect to preclude loss of component intended function is required.

(B) Identify whether the following systems have previously experienced cracks due to SCC, IGSCC, or cyclic loads. Discuss if an augmented inspection program was implemented for these systems as a part of AMP.

(1) CRD Nozzle welds
(2) In Core Monitor (ICM) nozzle welds
(3) Standby Liquid Control (SLC) nozzle welds
(4) Core Shroud
(5) Top Guide
(6) Core Spray Piping and Spargers

(7) Jet Pump Assembly

(C) The applicant credits its ASME Code Section XI inservice inspection program for managing cracking in the welded access hole covers due to SCC. This program requires visual inspection for detecting cracking. However, a crevice may be present near the weld and visual inspection may not be adequate for detecting cracks initiated in the crevice region. According to Section IV-B1.1.4 of NUREG-1801, an augmented inspection technique that includes ultrasonic testing (UT) or other demonstrated acceptable inspection method for the welded access hole cover should be used. Identify if this examination is implemented as a part of AMP for the welded access hole covers at the BFN Units.

TVA RESPONSE TO NRC RAI-B.2.1.12-1(A)

LRA section 4.7.6 considered fluence at the top guide as a TLAA. BFN manages this TLAA with the Chemistry Control Program and the BWR Vessel Internals Program. The BWR Vessel Internals Program implements the requirements of NRC accepted BWRVIP-26. NRC letter to Carl Terry, BWRVIP Chairman, dated June 10, 2003 states the following: "The staff believes that a comprehensive evaluation of the impact of IASCC and multiple failures of the top guide beams is necessary, and that an inspection program for top guide beams for all BWRs should be developed by the BWRVIP to ensure that all BWRs can meet the requirements of 10 CFR Part 54 throughout the period of extended operation." TVA will work as part of the BWRVIP to resolve these issues generically. When resolved, TVA will follow the BWRVIP recommendations resulting from that resolution. Prior to the period of extended operation, BFN will develop a site specific inspection program if these issues are not generically resolved.

TVA RESPONSE TO NRC RAI-B.2.1.12-1(B)

No cracking has been observed to date at BFN for the CRD Nozzle Welds, In-Core Monitor (ICM) Nozzle Welds, Standby Liquid Control (SLC) Nozzle Welds, and the Top Guide. Cracking in the other components listed under RAI-B.2.1.12 (B) is summarized as follows:

- Core Shroud –

  Refer to RAI-3.1.2.1.6 (c) response for cracking related to core shroud components.

- Core Spray Piping and Spargers –

  During the Unit 3 Cycle 7 (U3C7) Refueling Outage in 1997, UT examination indicated cracking in the Elbow to Shroud Pipe and Collar to Shroud welds in Downcomer "C". A calculation to allow operation for one fuel cycle was performed by GE-NE, and the lower section of Downcomer "C" was replaced with a bolted piping assembly during the U3C8 Refueling Outage in 1998.

  In 1991, cracking was discovered in the in the Unit 3 Core Spray Sparger adjacent to the T-Box located at 240°. Welded brackets were installed at both T-Boxes prior to Unit 3 Restart in 1995.

- Jet Pump Assembly –

  In 1991, crack indications were identified at the two attachment welds of the riser brace to the riser pipe adjacent to Unit 3 Jet Pump #5 at reactor vessel 90° azimuth. A repair clamp was installed prior to Unit 3 Restart in 1995, prior to issuance of any BWRVIP Jet Pump Repair Guidelines.

No augmented inspection program has been implemented for any of the components listed under RAI-B.2.1.12 (B) as part of this AMP. The CRD Nozzle Welds, ICM Nozzle Welds, and SLC Nozzle Welds are inspected in accordance with ASME Code Section XI as specified in BWRVIP-27 (SLC) and BWRVIP-47 (CRD, ICM). The Core Shroud, Top Guide, Core Spray Piping and Spargers, and Jet Pump Assembly are inspected in accordance with their respective BWRVIP Inspection and Evaluation Guideline as part of BFN's BWR Vessel Internals Program with three exceptions.

- In accordance with BWRVIP-27, an enhanced leakage inspection of the Standby Liquid Control (SLC) safe-end-to-nozzle weld during the ASME Section XI, IWB-2500, Code Category B-P system leakage test will be performed.

- The affect that the Unit 3 Jet Pump #5 repair design has on the implementation of the weld inspection requirements specified in BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," is currently being evaluated in the BFN Corrective Action Program.

- The affect that the Unit 3 Core Spray repair designs have on the implementation of the inspection requirements specified in BWRVIP-18, "BWR Core Spray Internals

Inspection and Flaw Evaluation Guidelines" is currently being evaluated in the BFN Corrective Action Program.

TVA RESPONSE TO NRC RAI-B.2.1.12-1(C)

Per Section 7.7 of Technical Instruction 0-TI-365, the core shroud access hole covers (AHCs) are examined in accordance with GE SIL No. 462, Revision 1. The GE SIL allows for inspection of the AHCs either by ultrasonic testing (UT) or top-surface visual (VT-1) inspection. BFN Site Engineering has always preferred the UT technique as this methodology is superior to any visual examination, due to not only the superior flaw detection but the provision for a longer reinspection interval. Due to tooling constraints, a top-surface EVT-1 (which is superior to the visual examination guidelines of GE SIL No. 462) was performed during the U3C11 Refueling Outage in March 2004. Only when tooling constraints warrant would a visual examination of the AHCs' welds be utilized at BFN.

Prior to the period of extended operation, BFN will enhance the BWR Vessel Internals Program to require visual inspection of the AHCs and inspection of the AHCs welds by UT unless tooling constraints prohibit performance of a UT. In the event tooling constraints prohibit inspection by UT, then the inspection will be performed by EVT-1. BFN inspects the AHCs utilizing the BWR Vessel Internals Program rather than the ASME Section XI Inservice Inspection Program currently specified in NUREG-1801.

Revise item 3.1.1.31 contained in LRA Table 3.1.1 as follows:

Table 3.1.1: Summary of Aging Management Evaluations for Reactor Coolant System Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management	Further Evaluation Recommended	Discussion
3.1.1.31	Core shroud and core plate access hole cover (welded and mechanical covers)	Crack initiation and growth due to SCC, IGSCC, and/or IASCC	ASME Section XI inservice inspection; water chemistry	No	Utilizes BWR Vessel Internals Program rather than ASME Section XI Inservice Inspection Program. Water Chemistry is consistent with NUREG-1801 with exceptions. See description of AMP in Section B.2.1.12 and B.2.1.5.

On page 3.1-37 of the LRA, revise the first line contained in Table 3.1.2.2 as follows:

Table 3.1.2.2: Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG - 1801 Vol. 2 Item	Table 1 Item	Notes
Reactor Vessel Internals Core Shroud and Core Plate	PB, SS	Nickel alloy	Treated Water (internal)	Crack initiation/ growth due to stress corrosion cracking (SCC).	BWR Vessel Internals (**B.2.1.12**) Chemistry Control Program (**B.2.1.5**)	IV.B1.1-d	None	B, E

Add the following note to LRA Table 3.1.2.2:

Industry Standard Notes:

Note E - Consistent with NUREG-1801 item for material, environment, and aging effect, a different aging management program is credited.

NRC RAI B.2.1.12-2

The NRC staff has approved the applicable BWRVIP reports and attached the following required license renewal applicant action items, in accordance with 10 CFR Part 54, when incorporating the reports in a license renewal application.

The license renewal applicant is to verify that its plant is bounded by the report. Further, the renewal applicant is to commit to programs described as necessary in the BWRVIP reports to manage the effects of aging during the period of extended operation. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within these BWRVIP reports described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for license renewal referencing the applicable BWRVIP report shall ensure that the programs and activities specified as necessary in the applicable BWRVIP reports are summarily described in the FSAR supplement.

The 10 CFR 54.22 requires that each application for license renewal include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. The applicable BWRVIP reports may state that there are no generic changes or additions to technical specifications associated with the report as a result of its aging management review and that the applicant will

provide the justification for plant-specific changes or additions. Those applicants for license renewal referencing the applicable BWRVIP reports shall ensure that the inspection strategy described in the reports does not conflict with or result in any changes to their technical specifications. If technical specifications changes do result, then the applicant must ensure that those changes are included in its application for license renewal. If required by the applicable BWRVIP report, the applicant referencing a particular report for license renewal should identify and evaluate any potential TLAA issues and/or commitments to perform future inspections when inspection tooling is made available.

Provide the necessary commitments, information and changes as described above for each of the following applicable BWRVIP reports:

- BWRVIP-74
- BWRVIP-05
- BWRVIP-38
- BWRVIP-76
- BWRVIP-75
- BWRVIP-25
- BWRVIP-27
- BWRVIP-48
- BWRVIP-18
- BWRVIP-26
- BWRVIP-41
- BWRVIP-47
- BWRVIP-49
- BWRVIP-78
- BWRVIP-86
- BWRVIP-42

Other reports applicable to license renewal for BFN Units 1, 2, and 3.

TVA RESPONSE TO NRC RAI-B.2.1.12-2

BFN does not have LPCI couplings, therefore BWRVIP-42 is not applicable to BFN. BWRVIP-74 is addressed in LRA section 3.1.2.2.16. For the remaining specific BWRVIP documents listed in RAI-B.2.1.12-2 and all other BWRVIP documents that are credited by BFN to manage the effects of aging during the period of extended operation, TVA will perform a review to confirm BFN Units 1, 2, and 3 are bounded by the conditions specified in the BWRVIP documents or identify and evaluate any exceptions.

Following this review, TVA will provide a list of commitments to the applicable BWRVIP documents or identify specific exceptions taken. Appendix A of the BFN LRA currently addresses several BWRVIP documents. Following this review, TVA will supplement Appendix A of the LRA as required to provide a summary program description to address each applicable BWRVIP document that is credited to manage the effects of aging during the period of extended operation.

Table 4-1 in BWRVIP-74-A summarizes reactor vessel component Inspections recommended by BWRVIP-27, 38, 41, 47, 48, and 49. Section 3.1.2.2.16.3 of the LRA states: "No Technical Specification changes are required for the inspection strategy described in the BWRVIP-74-A report." No Technical Specification changes are required for the reactor vessel internals inspection strategy described in BWRVIP-18, 25, and 26. BWRVIP-05 addresses reactor vessel shell weld inspection recommendations. BFN Units 2 and 3 have received relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 for the remainder of the 40 year licensed operating period. Section 4.2.6 and Appendix A.3.1.6 of the LRA state: "An extension of this relief for the 60-year period will be submitted to the NRC for approval prior to entering the period of extended operation." BWRVIP-78 and 86 address current term operation and do not apply to the period of extended operation. Currently, the only Technical Specification changes identified involve revision to the P-T limits. Section 4.2.5 of the LRA states: "Revised P-T limits will be prepared and submitted to the NRC for approval prior to the start of the extended period of operation." TVA will review additional BWRVIP documents credited for managing the aging effects during the period of extended operation and confirm no additional Technical Specification changes are required or identify the needed changes to be processed prior the period of extended operation.

TVA's commitments to the NRC are controlled by Standard Program and Process (SPP)-9.3, NRC Commitment Management. This SPP defines the requirements for initiating, identifying, documenting, revising, extending, tracking to completion, and maintaining Nuclear Regulatory Commission (NRC) commitments.

All applicable TLAAs are addressed in LRA sections 4.2, 4.3, and 4.7.

For open issues between the BWRVIP and NRC, such as future inspections when tooling is made available, TVA will work as

part of the BWRVIP to resolve these issues generically. When these issues are resolved, TVA will follow the BWRVIP recommendations resulting from that resolution.

Reactor Vessel Surveillance Program

NRC RAI B.2.1.28-1

(A) The applicant stated that it will implement the BWRVIP integrated surveillance program (ISP) BWRVIP-116, "BWR Vessel Internals Project Integrated Surveillance Program Implementation for License Renewal," which is currently being reviewed by the staff. If the BWRVIP-116 report is not approved by the staff, the applicant must submit a plant-specific surveillance program for each BFN unit, two years prior to the commencement of the extended period of operation. The applicant should make a commitment to indicate that it will implement either BWRVIP-116, as approved by the staff, or if the ISP is not approved two years prior to the commencement of the license renewal period, a plant-specific surveillance program for each BFN unit will be submitted. This commitment should also be stated in the updated final safety analysis report (UFSAR) Section A.1.25 "Reactor Vessel Surveillance Program" of the LRA.

(B) BWRVIP-116 provides guidelines for an Integrated Surveillance Program (ISP) to monitor neutron irradiation embrittlement of the reactor vessel beltline materials for all U.S. boiling water reactor (BWR) power plants for the license renewal period. However, BWRVIP-116 does not include BFN Unit 1 in the ISP. Provide a plant-specific surveillance program for BFN Unit 1 or discuss how BFN Unit 1 will be incorporated into BWRVIP-116 and provide an evaluation of the vessel-to-capsule material compatibility for the limiting plate and weld as was performed for the ISP program similar to the other plants specified in BWRVIP-86 and BWRVIP-116.

(C) The 10 CFR Part 50, Appendix H, requires that an Integrated Surveillance Program (ISP) used as a basis for a licensee-implemented reactor vessel surveillance program be reviewed and approved by the NRC staff. The ISP to be used by the applicant is a program that was developed by the BWRVIP. The applicant will apply the BWRVIP ISP as the method by which the BFN units comply with the requirements of 10 CFR Part 50, Appendix H. The BWRVIP ISP identifies capsules that must be tested to monitor neutron radiation embrittlement for all

licensees participating in the ISP, and identifies capsules that need not be tested (standby capsules). Tables 2-3 and 2-4 of BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal" indicate that capsules from Unit 2 will be tested and capsules from Unit 3 are not tested (standby capsules). These 10 untested capsules were originally part of the licensee's plant-specific surveillance program and have received significant amounts of neutron radiation. The staff requests that the applicant provide its plan to maintain the standby capsules in a condition which would permit their future use, including the period of extended operation, if necessary.

TVA RESPONSE TO NRC RAI B.2.1.28-1(A)

BFN will implement either BWRVIP-116, as approved by the staff, or if the ISP is not approved two years prior to the commencement of the license renewal period, a plant-specific surveillance program for each BFN unit will be submitted to the NRC. LRA Section A.1.25 is revised as shown below.

"The BFN Reactor Vessel Surveillance Program is mandated by 10 CFR 50 Appendix H. The BFN Reactor Vessel Surveillance Program is an integrated surveillance program in accordance with 10 CFR Part 50 Appendix H Paragraph III.C that is based on requirements established by the BWR Vessel and Internals Project. This program will be enhanced to implement either BWRVIP-116, as approved by the staff, or, if the ISP is not approved two years prior to the commencement of the license renewal period, a plant-specific surveillance program for each BFN unit will be submitted that ensures the BFN Unit 1, Unit 2, and Unit 3 reactor vessels meet the requirements of 10 CFR 50 Appendix H."

TVA RESPONSE TO NRC RAI B.2.1.28-1(B)

LRA Appendix B.2.1.28 discusses Unit 1 enhancements required to the Reactor Vessel Surveillance Program. As stated in Appendix B.2.1.28 of the LRA, "Unit 1 will be included within the BWRVIP Integrated Surveillance Program, or a plant specific surveillance program will be submitted for NRC approval that meets the requirements of 10 CFR 50 Appendix H for the period of extended operation."

The BWRVIP has evaluated the Browns Ferry Unit 1 vessel and surveillance program for participation in the ISP. The BWRVIP

proposed in their letter from William A. Eaton (Chairman, BWR Vessel and Internals Project) to Document Control Desk (NRC), Project No. 704 – BWRVIP Response to NRC Requests for Additional Information on BWRVIP-116," dated January 11, 2005, to include Browns Ferry Unit 1 in the ISP. BWRVIP-86-A and BWRVIP-116 will be updated to incorporate Browns Ferry Unit 1 accordingly and a license amendment will be submitted to the NRC to implement the ISP for site-specific use for Unit 1 prior to the beginning of the period of extended operation.

TVA RESPONSE TO NRC RAI B.2.1.28-1(C)

Presently, there are no plans to withdraw surveillance capsules from the Unit 3 reactor vessel since the BFN Unit 2 reactor vessel capsule provides the best representative material for both units. As stated in NRC Safety Evaluation of the BWRVIP Integrated Surveillance Program, dated February 1, 2002:

> "Although some surveillance capsules will be deferred and not tested as part of the ISP, all capsules that were previously credited as part of plant-specific surveillance programs will continue to be irradiated in their host reactors. Therefore, all irradiated material samples continue to remain available to the ISP, if needed, and no overall reduction in the number of materials being irradiated, number of specimen types, or number of specimens per reactor occurs as a result of the ISP."

BFN Unit 3 surveillance capsules will remain in place and will continue to be irradiated during plant operation, including the period of extended operation. Therefore, the Unit 3 irradiated material samples continue to remain available to the ISP, if needed.