December 1, 2004

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3, LICENSE RENEWAL APPLICATION (TAC NOS. MC1704, MC1705, AND MC1706)

Dear Mr. Singer:

By letter dated December 31, 2003, Tennessee Valley Authority (TVA) submitted an application pursuant to 10 CFR Part 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, for review by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified areas where additional information is needed to complete the review. Specifically, the enclosed requests for additional information (RAIs) are related to the aging management of reactor vessels and internals, Sections 3.1, 4.2, and B.2.1 of the LRA. Drafted forms of these RAIs were discussed with TVA staff on a telephone conference call held on September 16, 2004.

Based on discussions with Ken Brune of your staff, a mutually agreeable date for your response to these RAIs is within 30 days of the date of this letter. If you have any questions regarding this letter or if circumstances result in your need to revise the response date, please contact me at (301) 415-1478 or by e-mail at rxs2@nrc.gov.

Sincerely,
/RA/
Ram Subbaratnam, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-259, 50-260 and 50-296

Enclosure: As stated

cc w/encl: See next page

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BROWNS FERRY NUCLEAR PLANT (BFNP), UNITS 1, 2 AND 3
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION (RAI)
AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS,
AND REACTOR COOLANT SYSTEM
SECTIONS 3.1, 4.2, AND B.2.1

Steam Dryer

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 he cover plates of 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 CFR54.4(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.

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.

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.

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.

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, that 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.

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.

Vessel Attachment Welds

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 raiser 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.

Reactor Vessel Closure Studs and Nuts

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.

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.

Reactor Vessel Nozzles and Safe Ends

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.

Reactor Vessel Penetrations

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."

Reactor Vessel Internals Core Shroud and Core Plate

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.

Core Spray Spargers and Piping

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.

Reactor Vessel Internals Dry Tube and Guide Tube

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.

Jet Pump Assembly

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

Top Guide

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.

Bolting of Reactor Vessel Vents and Drains

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.

Bolting of Reactor Recirculation Systems

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.

Time Limited Aging Analysis (TLAA)

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.

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 x 1018 n/cm2 (E > 1.0 MeV) while Units 2 and 3 achieve 2.3x1018 n/cm2 (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.

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 1021 n/cm2 (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 1021 n/cm2. Since this fluence is greater than the irradiation assisted stress corrosion cracking (IASCC) threshold fluence [5 x 1020 n/cm2 (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 x 1021 n/cm2 (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.

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.

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.

Aging Management Programs (AMPs)

Reactor Head Closure Stud Program

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.

Boiling Water Reactor Feedwater Nozzle Program

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.

Boiling Water Reactor Control Rod Drive Return Line Nozzle Program

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 Piligrim (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.

Boiling Water Stress Corrosion Cracking Program

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.

<u>Boiling Water Reactor Internal Program</u>

<u>RAI B.2.1.12-1</u>

(A) Top Guide: BWRVIP-26, "Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines" BWRVIP-26 lists 5 x 1020 n/cm2 (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.

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.

Reactor Vessel Surveillance Program

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.

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