December 9, 2004

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 – MECHANICAL SYSTEMS SECTIONS 3.1.2.4, B.2.1.13, AND 4.7.8 – 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 November 4, 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 Sections 3.1.2.4, B.2.1.13, and 4.7.8 of the License Renewal Application. Drafted forms of these RAIs were discussed with TVA Staff during a telephone conference call on October 6, 2004. This request was not received electronically at BFN

until November 9, 2004 and through discussion with the NRR Project Manager it was agreed upon that a 30 day response from November 9, 2004, was acceptable.

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 9th day of December, 2004.

Sincerely,

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cc: See page 3

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cc: continued page 4

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s://Licensing/Lic/BFN LR Mech Sections 3.1.2.4, B.2.1.13, and 4.7.8.

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 MECHANICAL SYSTEMS SECTION 3.1.2.4, B.2.1.13, AND 4.7.8.

(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 MECHANICAL SYSTEMS SECTION 3.1.2.4, B.2.1.13, AND 4.7.8.

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 November 4, 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 Sections 3.1.2.4, B.2.1.13, and 4.7.8 of the License Renewal Application. Drafted forms of these RAIs were discussed with TVA Staff during a telephone conference call on October 6, 2004. This request was not received electronically at BFN until November 9, 2004 and through discussion with the NRR Project Manager it was agreed upon that a 30 day response from November 9, 2004 was acceptable.

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

Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

RAI B.2.1.13-1

The staff's position regarding screening criteria to determine susceptibility of components to thermal aging embrittlement of CASS is detailed in the letter dated May 19, 2000, from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI). The licensee states that the only in-scope CASS components that were determined to be susceptible to thermal aging embrittlement are the main steam line flow restricting venturis. The following information is requested.

(a) Please provide the material specification including material grade, chemical content, casting method (i.e.

either static or centrifugal), percent ferrite (using Hull’s equation factors or a method producing an equivalent level of accuracy) and operating temperature for the flow restricting venturis.

(b) Have the flow restricting venturis been evaluated in accordance with the guidance detailed in the May 19, 2000, NRC letter?

(c) Are the components considered potentially susceptible to thermal aging embrittlement when screened using the criteria outlined in the May 19, 2000, NRC letter?

TVA Response to RAI B.2.1.13-1

(a) The cast austenitic stainless steel flow restricting venturis are ASTM A 351 GRADE CF8. The chemical content used for the determination of molybdenum and delta-ferrite content were worst case values for ASTM A 351 GRADE CF8 from the 1975 Annual Book of ASTM Standards.

Element	%
Carbon	0.08
Manganese	1.50
Silicon	2.00
Sulfur	0.04
Phosphorus	0.04
Chromium	21.0
Nickel	8.0
Molybdenum	----

Based on this chemical composition and using the Hull’s equation from NUREG/CR-4513, Section 3.2 the delta ferrite content was determined to be 18.3%. The operating temperature of the main steam line is 550°F. Based on the molybdenum content and calculated delta-ferrite content, the casting method for the flow restricting venturis is not a determining factor in evaluating the flow venturis’ thermal aging susceptibly. The casting method for the main steam line flow restricting venturis was not determined and was not required to perform this evaluation.

(b) Yes, the cast austenitic stainless steel flow restricting venturis have been evaluated for thermal aging in accordance with the guidance detailed in the May 19, 2000, NRC letter.

(c) Based on Table 2 of the NRC Letter of May 19, 2000, the flow restricting venturis are not susceptible to thermal aging based on low molybdenum content (<.5%) in conjunction with a calculated delta-ferrite content of less than 20%. This conclusion remains valid whether the static or centrifugal casting method was utilized during manufacturing of the main steam line flow restricting venturis.

Emergency Equipment Cooling Water (EECW) Weld Flaw Evaluation

RAI 4.7.8-1

The applicant states that analysis was performed on 17 EECW system piping welds that have flaws larger than what is normally considered acceptable. Please discuss the circumstances under which these flaws were discovered and provide a list and description of each of the 17 welds in question including the code class, flaw inspection history, flaw sizes and a detailed description of any analysis including the method that was used to determine flaw growth.

TVA Response to RAI 4.7.8-1

The flawed EECW welds are on BFN Seismic Class I piping that was designed to the B31.1-1967 Power Piping Code. For the BFN ASME Section XI program the welds are classified as ASME Class 3. Design conditions for the EECW system are 200 psig and 200° F. All of the related piping is qualified by analysis. This analysis satisfies BFN Design Criteria No. BFN -50-C-7103 which supplements B31.1 analysis requirements by invoking plant condition dependent stress equations from ASME Section III, 1971 Edition, Summer 1973 Addenda. The stress analyses of the piping systems are also considered a Time Limited Aging Analysis (TLAA) which is addressed in the Application TLAA Section 4.3.3.

History of Discovery – A weld inspection program was initiated at BFN to determine the effects of MIC on the stainless steel piping girth butt welds in the EECW system, as a result of MIC discoveries at other plants. The inspection program was implemented by performing Radiography on a sample of EECW piping welds. Radiography had not been performed on these welds during installation, as it was not required by the applicable code and specifications. The inspection identified defects in 33 welds. The 33 welds which had identified defects were reviewed by the ISI level III interpreter and 27 of the welds were rejected because they did not meet ISI flaw acceptance standards. The

ISI level III interpreter determined that the other welds did meet flaw acceptance standards.

Analysis Performed – Two analyses were performed in association with the qualification of the remaining 27 EECW welds with welding defects. These are:

I. Bounding Fracture Mechanics Analysis – Initially, a conservative bounding fracture mechanics analysis was performed for the scope of stainless steel EECW pipe sizes encompassing the 27 welds that were rejected based on ASME Section XI acceptance standards. Of the 27 welds 10 were found to be acceptable using the bounding fracture mechanics analysis. The qualification of the remaining 17 welds, which are the subject of this RAI, are addressed later.

   This calculation first computes the Code allowable flaw sizes in each applicable pipe size in accordance with ASME Section XI paragraph IWB-3640 supplemented by ASME Code Case N-436 and EPRI Report NP-4690-SR. The bounding nature of this analysis is based on the conservative assumption that at each weld location, applied primary stresses are set equal to the ASME Section III Subsection NC-3652 Code allowable stress for Piping Equation 9 and, the applied secondary stress (i.e., thermal expansion induced) is set equal to the ASME Section III Subsection NC-3652 Code allowable stress for Piping Equation 10. Piping Equation 9 accounts for primary stress during upset, emergency, or faulted plant conditions and includes seismic loading in addition to pressure and deadweight loads. Using the Code allowable stress for primary stress load combinations as input, the pc-CRACK fracture mechanics software developed by Structural Integrity Associates, Inc. was employed to compute allowable flaw sizes. The allowable flaw sizes include appropriate safety factors as required by ASME Section XI and are based on a limit load failure mode because the welds were fabricated using a gas tungsten arc weld (GTAW) procedure.

   To account for sub-critical flaw growth over the life of the plant, the Code allowable flaw sizes were then reduced by the fatigue crack growth applicable to each pipe size. The fatigue crack growth was computed based on 125 pressurization plus thermal expansion stress cycles which were postulated to occur over the remaining 25 years (i.e., at the time of the calculation) of the 40 year plant life.

The per cycle crack growth increment is based on a cyclic stress equal to the sum of longitudinal membrane stress due to design pressure and the bounding thermal expansion stress discussed above. The total fatigue crack growth was computed as a summation of the per cycle growth increment using differential crack tip stress intensities (i.e., ΔK's) based on the previously computed Code allowable flaw sizes.

The allowable flaws, following adjustment for fatigue crack growth, were output as a set of allowable flaw depths versus corresponding allowable flaw lengths for each pipe size within the scope of the 27 welds with defects. Mathematically, the allowable depth ranged from a through-wall flaw (i.e., allowable depth = pipe thickness) at some finite flaw length (i.e., length greater than zero) down to a finite depth allowable flaw (i.e., allowable depth greater than 0) for a flaw length equal to the pipe circumference (i.e., 360° around).

The defects in the 27 welds were then evaluated against these adjusted allowable flaw sizes. This evaluation was conservatively made by comparing the aggregate defect length in each weld (i.e., the sum of individual defects in each weld) with the allowable length for a through-wall flaw. 10 of the 27 welds were found to be acceptable on this basis.

Because the 125 stress cycles used in this analysis coincides with the LRA Section 4.7.8, the results and disposition presented in LRA Section 4.7.8 also applies to these 10 welds. Following is a list and description of the 10 welds which are acceptable based on the bounding fracture mechanics analysis:

1. T-EECW-1-A40; This weld attaches a 4" stainless steel weld neck flange to 4" stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 1A Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 0.44" and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 1.28" for 4" pipe.

2. T-EECW-1-C41; This weld attaches a 4" stainless steel elbow to 4" stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine

1C Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 0.38″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 1.28″ for 4″ pipe.

3. T-EECW-1-C47; This weld attaches a 4″ stainless steel elbow to 4″ stainless steel piping. The related piping provides the cross tie between the two Diesel Generator Engine 1C Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 0.34″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 1.28″ for 4″ pipe.

4. T-EECW-1-D29; This weld attaches the 4″ end of a 6″ x 4″ carbon steel reducer to a 4″ stainless steel elbow. The related piping provides the coolant supply for Diesel Generator Engine 1D Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is listed as "NIL" in TVA records documenting disposition of the weld defect issue. Review of the radiographic sheets by qualified examiners confirmed that this weld has no discernable defect length and is therefore acceptable relative to the bounding allowable through-wall flaw length of 1.28″ for 4″ pipe.

5. T-EECW-2-CAC-24B; This weld attaches a 2½″ stainless steel 45° elbow to 2½″ stainless steel piping. The related piping provides the coolant return from Core Spray Pump Room Cooler 2A, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 0.13″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 0.80″ for 2½″ pipe.

6. T-EECW-2-CBD-26B; This weld attaches a 2½″ stainless steel elbow to 2½″ stainless steel piping. The related piping provides the coolant supply for the Core Spray Pump Room Cooler 2B and is located in the Unit 2 Reactor Building. The aggregate flaw length is 0.25″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 0.80″ for 2½″ pipe.

7. T-EECW-2-CBD-33B; this weld attaches a 2½″ stainless steel elbow to 2½″ stainless steel piping. The related

piping provides the coolant return from Core Spray Pump Room Cooler 2B, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 0.47″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 0.80″ for 2½″ pipe.

8. T-EECW-3-B49; This weld attaches a 4″ stainless steel elbow to 4″ stainless steel piping. The related piping provides the cross tie between the two Diesel Generator Engine 3B Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 0.85″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 1.28″ for 4″ pipe.

9. T-EECW-3-C17; This weld attaches a 4″ stainless steel tee to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 3C Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 1.22″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 1.28″ for 4″ pipe.

10. T-EECW-3-C18; This weld attaches a 4″ stainless steel tee to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 3C Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 0.47″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 1.28″ for 4″ pipe.

II. Weld Specific Fracture Mechanics Analysis – For the 17 welds with defects that were not acceptable per the bounding fracture mechanics analysis discussed above, a location specific fracture mechanics analysis was performed. These are the 17 welds identified in LRA Section 4.7.8. The weld specific analysis applied essentially the same approach and considerations as the bounding analysis except that location specific stresses determined for ASME Section III, Subsection NC-3652 Equations 9 and 10 in the piping analyses of record were used to calculate both the ASME Section XI allowable flaw size and the fatigue crack growth due to cyclic load for the 25 years remaining in the plant life. It may be noted

that for the controlling location (i.e., maximum thermal stress) in each pipe size applicable to the 17 welds, fatigue crack growth for the 25 year period was found to be insignificant. Thus, this analysis provided a specific set of allowable flaw depths versus allowable flaw lengths for each of the 17 welds. As in the bounding fracture mechanics analysis, allowable depth for location specific stresses ranged from a through-wall flaw at some finite flaw length down to a finite depth allowable flaw for a flaw length equal to the pipe circumference.

One other difference relative to the bounding fracture mechanics analysis is that one of the 17 welds, EECW-1-D24, is a dissimilar metal joint where Type 316 stainless steel pipe is welded to a carbon steel reducer fitting (i.e., equivalent to A106 Grade B pipe) using ER309 filler metal. Although this weld is primarily austenitic, for the determination of allowable flaw size in the weld specific fracture mechanics analysis, it was treated as ferritic in accordance with the flaw evaluation procedures introduced by Article IWB-3650 of ASME Section XI and implemented by the pc-CRACK fracture mechanics software with appropriate adjustments. These adjustments included consideration of thermal expansion stress as a primary stress and use of a "Z" factor multiplier to account for the ductile tearing failure mode in the elastic-plastic fracture mechanics analysis method. In addition, the thermal expansion stress was adjusted to include the induced stress due to differential thermal expansion of carbon and stainless materials following heat-up of the system to operating temperature.

The 17 welds with defects that did not meet the bounding fracture mechanics analysis were conservatively evaluated by comparing the aggregate flaw length in the weld to maximum allowable length for a through-wall flaw. On this basis, 14 of the 17 welds were found to be acceptable. For the 3 remaining welds, the aggregate flaw length was greater than the allowable length for a through-wall flaw. To address this situation, the non-destructive examination results for the 3 welds in question were reviewed by a Level III examiner who determined that the actual defects were confined to the weld root with a maximum depth no greater than 0.09 inches in any case. Because the aggregate flaw length in each weld was less than the allowable flaw length for a flaw depth equal to 0.09

inches, the remaining 3 welds in question were found to be acceptable.

The weld specific fracture mechanics analysis was originally performed in 1988 but was revised in 1993 to address reanalysis of the EECW piping. Although pipe stresses increased for some welds due to the piping reanalysis, the 1993 revision of the weld specific fracture mechanics analysis demonstrated that all 17 welds remain acceptable. Because it includes consideration of defect growth due to plant operational stress cycles, the weld specific fracture mechanics analysis, as revised in 1993, contains the TLAA for the subject 17 welds.

The following is a list of the 17 subject welds, by weld number, and a brief description.

1. T-EECW-1-A42; This weld attaches a 4″ stainless steel elbow to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 1A Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 3.38″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 9.87″ for the 4″ piping at this location.

2. T-EECW-1-B11; This weld attaches a 4″ stainless steel tee to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 1B Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 2.38″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 4.23″ for the 4″ piping at this location.

3. T-EECW-1-B14; This weld attaches a 4″ stainless steel elbow to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 1B Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 1.40″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 7.05″ for the 4″ piping at this location.

4. T-EECW-1-D24; This weld attaches the 4″ end of a 6″ x 4″ carbon steel reducer to a 4″ stainless steel elbow. The related piping provides the coolant supply for Diesel Generator Engine 1D Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 2.22″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 7.05″ for the 4″ piping at this location.

5. T-EECW-2-AC-09B; This weld attaches a 3″ stainless steel elbow to 3″ stainless steel piping. The related piping provides the coolant supply for RHR Pump Room Coolers 2A and 2C and is located in the Unit 2 Reactor Building. The aggregate flaw length is 10.75″ and the depth is no greater than 0.09″. The 0.09″ (42%) flaw depth is conservatively assumed for the full length in the analysis. This was compared to an allowable of 80% through-wall flaw for the full circumference of the 3″ piping at this location.

6. T-EECW-2-BD-25B; This weld attaches a 3″ stainless steel elbow to 3″ stainless steel piping. The related piping provides the coolant supply for RHR Pump Room Cooler s 2B and 2D and is located in the Unit 2 Reactor Building. The aggregate flaw length is 1.63″ and it was conservatively assumed to be through-wall for the analysis. This was originally compared to an allowable through-wall flaw length of 6.6″ for the 3″ piping at this location. The later revision for increased pipe stress at this location resulted in a safety factor of 25.2 for a 1.63″ long through-wall flaw at this location compared with an ASME Code allowable minimum safety factor of 2.77 for the upset plant condition.

7. T-EECW-2-BD-37B; This weld attaches a 3″ stainless steel weld neck flange to 3″ stainless steel piping. The related piping provides the coolant supply for RHR Pump Coolers 2A and 2C and is located in the Unit 2 Reactor Building. The aggregate flaw length is 1.28″ and it was conservatively assumed to be through-wall for the analysis. This was originally compared to an allowable through-wall flaw length of 5.5″ for the 3″ piping at this location. The later revision for increased pipe stress at this location resulted in a safety factor of 15.9 for a 1.28″ long through-wall flaw at this location

compared with an ASME Code allowable minimum safety factor of 2.77 for the upset plant condition.

8. T-EECW-2-CAC-06B; This weld attaches a 2½″ stainless steel valve to 2½″ stainless steel piping. The related piping provides the coolant supply for Core Spray Pump Room Cooler 2A, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 7.03″ and the depth is no greater than 0.09″ (44%). This was compared to an allowable of 94% through-wall flaw for 80% of the full circumference (i.e., 7.2″) of the 2½″ piping at this location.

9. T-EECW-2-CAC-09B; This weld attaches a 2½″ stainless steel elbow to 2½″ stainless steel piping. The related piping provides the coolant supply for Core Spray Pump Room Cooler 2A, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 1.00″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 4.5″ for the 2½″ piping at this location.

10. T-EECW-2-CBD-02B; This weld attaches the 2½″ end of a 2½“ x 2″ stainless steel reducer to 2½″ stainless steel piping. The related piping provides the coolant supply for Core Spray Pump Room Cooler 2B, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 2.03″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 3.6″ for the 2½″ piping at this location.

11. T-EECW-2-AC-25B; This weld attaches the 2″ end of a 3“ x 2″ stainless steel reducer to 2″ stainless steel piping. The related piping provides the coolant supply for RHR Pump Room Cooler 2A, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 4.23″ and the depth is no greater than 0.09″ (58%). This was originally compared to an allowable of 94% through-wall flaw for 60% of the full circumference (i.e., 4.5″) of the 2″ piping at this location. The later revision for increased pipe stress at this location resulted in a safety factor of 8.54 for a 4.23″ long, 58% through-wall flaw at this location compared with an ASME Code allowable minimum safety factor of 2.77 for the upset plant condition.

12. T-EECW-2-BD-07B; This weld attaches a 3″ stainless steel 45° elbow to 3″ stainless steel piping. The related piping provides the coolant return from the Core Spray Pump Room Cooler 2B, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 1.38″ and it was conservatively assumed to be through-wall for the analysis. This was originally compared to an allowable through-wall flaw length of 5.5″ for the 3″ piping at this location. The later revision for increased pipe stress at this location resulted in a safety factor of 15.2 for a 1.38″ long through-wall flaw at this location compared with an ASME Code allowable minimum safety factor of 2.77 for the upset plant condition.

13. T-EECW-2-CBD-32B; This weld attaches a 2½″ stainless steel 45° elbow to 2½″ stainless steel piping. The related piping provides the coolant return from RHR Pump Room Coolers 2B and 2D, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 0.94″ and it was conservatively assumed to be through-wall for the analysis. This was originally compared to an allowable through-wall flaw length of 5.4″ for the 2½″ piping at this location. The later revision for increased pipe stress at this location resulted in a safety factor of 39.5 for a 0.94″ long through-wall flaw at this location compared with an ASME Code allowable minimum safety factor of 2.77 for the upset plant condition.

14. T-EECW-2-BD-10B; This weld attaches a 3″ stainless steel elbow to 3″ stainless steel piping. The related piping provides the coolant return from RHR Pump Room Coolers 2B & 2D, and is located in the Unit 2 Reactor Building. The aggregate flaw length is 2.13″ and it was conservatively assumed to be through-wall for the analysis. This was originally compared to an allowable through-wall flaw length of 6.6″ for the 3″ piping at this location. The later revision for increased pipe stress at this location resulted in a safety factor of 19.3 for a 2.13″ long through-wall flaw at this location compared with an ASME Code allowable minimum safety factor of 2.77 for the upset plant condition.

15. T-EECW-3-B50; This weld attaches a 4″ stainless steel elbow to another 4″ stainless steel elbow. The related

piping provides the cross tie between the two Diesel Generator Engine 3B Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 3.57″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 8.46″ for the 4″ piping at this location.

16. T-EECW-3-C01; This weld attaches a 4″ stainless steel weld neck flange to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 3C Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 1.69″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 4.23″ for the 4″ piping at this location.

17. T-EECW-3-D08; This weld attaches a 4″ stainless steel elbow to 4″ stainless steel piping. The related piping provides the coolant supply for Diesel Generator Engine 3D Coolers and is located in the Standby Diesel Generator Building. The aggregate flaw length is 2.53″ and it was conservatively assumed to be through-wall for the analysis. This was compared to an allowable through-wall flaw length of 8.46″ for the 4″ piping at this location.

Reactor Recirculation System Aging Management Evaluation

RAI 3.1.2.4-3

In Table 3.1.2.4 Page 3.1-54 (on CD ROM version of LRA) lists a copper alloy material used in heat exchangers operating in a raw water (internal) environment. The aging effects are listed as loss of material due to biofouling, microbiologically induced corrosion (MIC), crevice and pitting corrosion. The applicant lists the AMP as the One-Time Inspection Program (B.2.1.29). The table item shows that neither the component nor the material and environment combination is evaluated in NUREG-1801. The table notes indicate that the aging effects identified for this material/environment combination are consistent with industry guidance.

(a) Identify the heat exchangers involved and the type of copper alloy (including material specification). Discuss the age, operating history, and inspection history of the

heat exchangers. Discuss the consequence of leakage in the heat exchangers. Based on the operating and inspection history, explain why periodic inspections are not necessary during the extended license period.

(b) Discuss in detail the extent of the planned One-Time inspection and explain why a One-Time inspection will be adequate to detect early stages of degradation that may effect the reliable operation of the heat exchanger through the extended licensing period.

TVA Response to RAI 3.1.2.4-3

The heat exchangers identified in Table 3.1.2.4, Reactor Recirculation System, are the reactor recirculation pump motor generator raw water/lubrication oil heat exchangers for Unit 1 and the reactor recirculation pump variable frequency drive raw water/treated water heat exchangers for Units 2 and 3. The Unit 1 reactor recirculation pump motor generators will be replaced by variable frequency drives prior to Unit 1 restart. The variable frequency drives are recent additions, with Unit 2 completed in March 2003 and Unit 3 completed in March 2004. The heat exchangers are in scope for license renewal because they form a portion of the Raw Cooling Water System pressure boundary.

The raw water environment for these heat exchangers is supplied from the Raw Cooling Water System and the appropriate aging management program is the Open-Cycle Cooling Water Program. This correction was identified in response to Question 397 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).

RAI 3.1.2.4-4

In Table 3.1.2.4, Page 3.1-55 lists copper alloy piping operating in a treated water (internal) environment. NUREG-1801 Vol. 2 Item VII.C2.1-a. The aging effects are listed as loss of material due to crevice and pitting corrosion. The applicant lists the AMP as the One-Time Inspection Program (B.2.1.29). Table 3.1.2.4 notes show that the material is not listed in NUREG-1801 for this component. The table notes indicate that

the aging effects identified for this material/environment combination are consistent with industry guidance.

Discuss the age, operating history and inspection history of the aforementioned copper alloy piping. Discuss the consequence of leakage of this piping. Discuss in detail the extent of the planned One-Time inspection and explain why a One-Time inspection will be adequate to detect early stages of degradation that may effect the reliable operation of these components through the extended licensing period. In addition, please provide the material specification of the piping.

TVA Response to RAI 3.1.2.4-4

The copper alloy piping operating in a treated water (internal) environment in Table 3.1.2.4, Reactor Recirculation System, refers to a self-contained cooling water subsystem supplied as an integral portion of the reactor recirculation pump variable frequency drives. The reactor recirculation pump variable frequency drives are recent additions, with Unit 2 installed in March 2003 and Unit 3 installed in March 2004. The piping is in scope for license renewal in accordance with 10 CFR 54.21(a)(2). In addition, reactor recirculation pump reactor recirculation pump variable frequency drives will be installed on Unit 1 prior to restart.

This copper piping was provided as part of the recirculation pump variable frequency drives. The recirculation pump variable frequency drives vendor manual identifies the material as red brass. The appropriate aging management programs for this piping are the Chemistry Control Program and the One-Time Inspection Program. This correction was identified in response to Question 397 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).

RAI 3.1.2.4-5

In Table 3.1.2.4 Page 3.1-62 lists stainless steel valves operating in a treated water (internal) environment. The table notes show that the aging effect in NUREG-1801 (V.C.1-b) for this component, material and environment combination is not applicable. The table notes also indicate that based on the

system design and operating history, MIC and biofouling are not applicable to the treated water portions of this system.

Discuss the age, operating history, and inspection history of the valves. Provide a detailed explanation of the attributes of the system design that make degradation due to MIC and biofouling not applicable.

TVA Response to RAI 3.1.2.4-5

The stainless steel valves operating in a treated water (internal) environment in Table 3.1.2.4, Reactor Recirculation System, refers to a self-contained cooling water subsystem supplied as an integral portion of the reactor recirculation pump variable frequency drives. The reactor recirculation pump variable frequency drives are recent additions, with Unit 2 installed in March 2003 and Unit 3 installed in March 2004. The valves are in scope for license renewal in accordance with 10 CFR 54.21(a) (2). In addition, reactor recirculation pump reactor recirculation pump variable frequency drives will be installed on Unit 1 prior to restart.

Consistent with all treated water environments identified in the BFN license renewal application, microbiologically influenced corrosion and biofouling are not aging mechanisms requiring management in a treated water environment. Microbiologically influenced corrosion is not identified for treated water systems where there is no mechanism for the introduction of the bacteria that causes microbiologically influenced corrosion. The water in this cooling water subsystem is demineralized water that has no history of microbiologically influenced corrosion activity. Biofouling refers to macrobiological growth that is not applicable to a treated water environment.

RAI 3.1.2.4-6

In Table 3.1.2.4 identifies a One-Time inspection specified in B.2.1.29 as a part of the Aging Management Program (AMP). According to B.2.1.29 a One-Time inspection is applicable for piping and fittings with diameter less than 4 inch nominal pipe size (NPS). Identify whether the reactor recirculation system has previously experienced cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC) or cyclic loading, and the extent of cracking. If cracking has occurred, identify whether the cracking was in piping less than 4 inch NPS and/or over 4 inch NPS. Identify the method of the One-Time inspection. The LRA does not address the type and

frequency of the inspection requirements as a part of the AMP, for piping greater than 4 inch NPS. Identify the AMP used for piping greater than 4 inch NPS.

TVA Response to RAI 3.1.2.4-6

These are the Recirculation System welds with known IGSSC indications.

UNIT 1

WELD ID	SIZE	IGSCC Cat.
GR-1-01	28	F
GR-1-02	28	F
GR-1-03(OL)	28	E
GR-1-27(OL)	28	E
GR-1-41(OL)	12	E
GR-1-45	12	F
GR-1-46(OL)	12	E
GR-1-54(OL)	28	E
GR-1-56(OL)	28	E
GR-1-57(OL)	28	E
GR-1-58(OL)	28	E
GR-1-60(OL)	28	E
GR-1-61(OL)	28	E
GR-1-64(OL)	28	E
KR-1-02	28	F
KR-1-03(OL)	28	E
KR-1-12	22	F
KR-1-14(OL)	12	E
KR-1-15(OL)	22	E
KR-1-16(OL)	12	E
KR-1-18(OL)	12	E
KR-1-20(OL)	12	E
KR-1-21(OL)	12	E
KR-1-22(OL)	12	E
KR-1-24	28	F
KR-1-25	28	F
KR-1-34	22	F
KR-1-36(OL)	12	E
KR-1-37(OL)	22	E
KR-1-42(OL)	12	E
KR-1-45(OL)	28	E
KR-1-47	28	F
KR-1-48(OL)	28	E
KR-1-52(OL)	28	E

UNIT 2

WELD ID	**SIZE**	**IGSCC Cat.**
GR-2-15(OL)	12	E
GR-2-45(OL)	12	E
GR-2-53	28	E
GR-2-59(OL)	28	E
GR-2-61(OL)	28	E
GR-2-64(OL)	28	E
KR-2-14	12	E
KR-2-36	12	E
KR-2-37	22	E
KR-2-41	12	E

UNIT 3

WELD ID	**SIZE**	**IGSCC Cat.**
GR-3-03(OL)	28	E
GR-3-27(OL)	28	E
GR-3-53(OL)	28	E
GR-3-54(OL)	28	E
GR-3-57(OL)	28	E
GR-3-59(OL)	28	E
GR-3-60(OL)	28	E
GR-3-63	28	E
GR-3-64(OL)	28	E

There were no Recirculation System welds less than 4-inches NPS identified as having cracking or crack indications in the inservice inspection records. Also note that during the BFN Unit 1 recovery efforts the Recirculation System piping greater than or equal to 4-inches NPS is being replaced with IGSCC resistant piping (316NG or 316L), which includes removal of the BFN Unit 1 welds listed above. The Recirculation system piping less than 4-inch NPS is also being replaced with 316 grade material, which is socket welded.

Cracking of RCPB piping and fittings less than 4 - inches NPS are addressed by Table 3.1.2.4, line items 41, 48, 50, 88, and 90. The ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection and One-Time Inspection Programs identified in Table 3.1.2.4, provides the method and frequency of inspection in Appendix B.2.1.4, and B.2.1.29. The One-Time Inspection Program will include UT inspections on a sample of less than 4-inch NPS full penetration butt welds.

Cracking of RCPB piping 4-inches NPS or greater are addressed by Table 3.1.2.4, line item 85. The BWR Stress Corrosion Cracking Program identified in Table 3.1.2.4 provides the method and frequency of inspection in Appendix B.2.1.10.