UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 257 TO FACILITY OPERATING LICENSE NO. DPR-52,

AND AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT UNITS 2, AND 3

DOCKET NOS. 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated March 3, 1998, Tennessee Valley Authority, (TVA or the licensee) submitted a request to amend the pressure-temperature (P-T) limit curves in the Technical Specifications (TSs) for Browns Ferry Nuclear Plant (BFN) Units 2 and 3. The amendments are intended to extend the validity of the BFN Units 2 and 3 curves to 32 effective full power years (EFPY). The U.S. Nuclear Regulatory Commission (NRC) staff found that the licensee did not use appropriately calculated fluence values in its evaluation for the BFN Units 2 and 3 P-T curves. It was also noted that the licensee did not use the appropriate initial RT_{NDT} values and the appropriate sigma initial values for the electroslag welds, as previously found acceptable in a safety evaluation dated February 28, 1997 for Dresden nuclear facility, and as recommended in the BAW-2258 and BAW-2259 Reports (references 1 and 2).

By letter dated October 19, 1998, the NRC staff issued a request for additional information (RAI) regarding clarifications of the licensee's submittal. TVA addressed the RAI by letter dated November 13, 1998. In addition, several follow-up telephone conversations took place to discuss the fluence and initial RT_{NDT} values, as mentioned above. The licensee then submitted a supplement to the March 3, 1998 letter. In this submittal, dated December 15, 1998, the licensee indicated that they resolved the subject issues by re-evaluating the proposed P-T curves of the March 3, 1998, letter, utilizing appropriately calculated fluence values, and the recommended values for the initial RT_{NDT} and sigma initial for the electroslag welds for BFN Units 2 and 3. The re-evaluated Units 2 and 3 P-T curves are identical to the curves of the March 3, 1998, letter with the exception that the proposed P-T curves are valid to 16 EFPY and 20 EFPY, respectively.

On April 22, 1998, the NRC noticed its proposed no significant hazards consideration determination, of the TVA's March 3, 1998 application, in the Federal Register (FR) (63 FR 19979). The licensee's letters dated November 13, and December 15, 1998, which provided additional information did not expand the scope of the application as noticed in the above-cited FR notice, or affect the NRC staff's proposed no significant hazards consideration determination.

2.0 EVALUATION

The NRC staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees that the NRC staff would use RG 1.99, Revision 2 to review P-T Limit curves. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the NRC staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the NRC staff as the basis for the NRC staff's review of P-T limit curves, and as the basis for the NRC staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T Limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code.

SRP 5.3.2 provides an acceptable method of calculating the P-T Limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T Limit Curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term.

2.1 BFN Unit 2

The licensee determined that the most limiting material at the 1/4T and 3/4T locations was the electroslag longitudinal weld for BFN Unit 2. The licensee calculated an ART of 103°F at the 1/4T location and 80°F at the 3/4T location at 16 EFPY for BFN Unit 2. The neutron fluence

used in the ART calculation was 3.8 X 10^{17} n/cm^2 at the 1/4T location and 1.8 X 10^{17} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting electroslag weld was a generic value of 23.1°F with a standard deviation of 13°F, as previously found acceptable in a safety evaluation dated February 28, 1997 for Dresden nuclear facility, and as recommended in the BAW-2258 and BAW-2259 Reports (references 1 and 2). The values for sigma delta at the 1/4T and 3/4T locations were 17.9°F and 11.4°F, respectively. Using these values for sigma delta and 13°F for sigma initial resulted in a margin value of 44.2°F at the 1/4T location and 34.5°F at the 3/4T location.

2.2 BFN Unit 3

The licensee determined that the most limiting material at the 1/4T and 3/4T locations was the electroslag longitudinal weld for BFN Unit 3. The licensee calculated an ART of 113°F at the 1/4T location and 86°F at the 3/4T location at 20 EFPY for BFN Unit 3. The neutron fluence used in the ART calculation was 4.7 X 10^{17} n/cm^2 at the 1/4T location and 2.3 X 10^{17} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting electroslag weld was a generic value of 23.1°F with a standard deviation of 13°F, as previously found acceptable in a safety evaluation dated February 28, 1997 for Dresden nuclear facility, and as recommended in the BAW-2258 and BAW-2259 Reports (references 1 and 2). The values for sigma delta at the 1/4T and 3/4T locations were 20.5°F and 13°F, respectively. Using these values for sigma delta and 13°F for sigma initial resulted in a margin value of 48.5°F at the 1/4T location and 36.9°F at the 3/4T location.

The ART is determined using the chemistry values for each beltline material of BFN Units 2 and 3. The Reactor Vessel Integrity Database (RVID) contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee provided updated chemistry data for the beltline materials of BFN Units 2 and 3 by letter dated September 8, 1998. It should be noted that the licensee and NRC staff used the most recent updated chemistry data for the beltline materials or the most conservative weld chemistry data in the BFN Units 2 and 3 P-T limit evaluations.

The NRC staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. The NRC staff verified that the licensee used appropriately calculated fluence values for the BFN Units 2 and 3 P-T limit evaluation. In addition, the NRC staff verified that the licensee used the recommended initial RT_{NDT} and sigma initial values of 23.1°F and 13°F, respectively, for electroslag welds, as previously found acceptable in a safety evaluation dated February 28, 1997 for Dresden nuclear facility, and as recommended in the BAW-2258 and BAW-2259 Reports (references 1 and 2). Based on these calculations, the NRC staff verified that the licensee's limiting material for the BFN Units 2 and 3 reactor vessels was the electroslag longitudinal weld. The NRC staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value at 16 and 20 EFPY, for BFN Units 2 and 3, respectively. Substituting the ART values for the BFN limiting weld into the equations in SRP 5.3.2, the NRC staff verified that the proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions

highly stressed by the bolt reload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 22°F for BFN Unit 2 and 10°F for BFN Unit 3, provided by the licensee, the NRC staff has determined that the proposed P-T limits satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test.

Based on the above discussion, the NRC staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for BFN Units 2 and 3, for 16 and 20 EFPY, respectively. The proposed P-T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Hence, the proposed P-T limits may be incorporated into the BFN Units 2 and 3 TS.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 19979). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Evaluation of RT_{NDT}, USE, and Chemical Composition of Core Region Electroslag Welds for Dresden Units 1 and 2, Framatome Technologies, BAW-2258, January 1996.

2. Evaluation of RT_{NDT}, USE, and Chemical Composition of Core Region Electroslag Welds for Quad Cities Units 1 and 2, Framatome Technologies, BAW-2259, January 1996.

3. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

4. NUREG-0800, Standard Review Plan, Section 5.3.2: "Pressure-Temperature Limits."

5. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."

6. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," July 12, 1988.

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Non-ductile Failure."

8. Letter from Tennessee Valley Authority (TVA) to USNRC Document Control Desk, Subject: "Browns Ferry Nuclear Plant - Units 2 and 3 - TS Change No. 393 - P-T Curve Update," March 3, 1998.

9. Letter from TVA to USNRC Document Control Desk, Subject: "Browns Ferry Nuclear Plant - Units 2 and 3 - TS Change No. 393, Supplement 1- P-T Curve Update," December 15, 1998.

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