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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 239 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated March 31, 1995, the Tennessee Valley Authority (the licensee) submitted an application to amend the Technical Specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. The licensee proposed revision of the pressure/temperature (P/T) curves for the units, which lowers the temperature at which the reactor vessel head bolting studs may be fully tensioned (bolt-up temperature). The curves remain valid for 12 effective full power years (EFPY). Supplemental information provided by the licensee on July 14, 1995 does not affect the staff's proposed finding of no significant hazards considerations.

2.0 BACKGROUND

The current P/T curves for BFN Units 1, 2, and 3 require a 100°F bolt-up temperature. Lowering the bolt-up temperature reduces the complexity of the Integrated Leak Rate Test and further ensures the reactor vessel and drywell heads (heavy loads) would only have to be lifted once at the end of each outage. In addition, there may not be sufficient decay heat available to heat the primary system to the bolt-up temperature in a timely fashion during an extended outage. Similarly, the use of alternate heat sources (i.e., running the residual heat removal pumps) also results in a delay of the outage. Thus, decreasing the required bolt-up temperature will decrease the overall Unit 2 Cycle 8 and Unit 3 Cycle 6 outage time by several hours, with similar improvements in future outages.

The reactor vessel bolt-up temperatures for each unit were established by adding 60°F to the highest value of the reference temperature for nil-ductile transition (RT_{NDT}) of the closure flange region. The limiting values of RT_{NDT} are based on fracture toughness data from the Certified Material Test Reports

Enclosure 4

and the General Electric (GE) methodology, which yields lower bolt-up temperatures than currently in the TS.

The new P/T limits were calculated using chemistries that were reported in response to Generic Letter (GL) 92-01, and embrittlement estimations that are in accordance with Regulatory Guide 1.99, Revision 2. Fluence and unirradiated RT_{NDT} values were established using GE methodologies. Application of the revised P/T curves would decrease the length of the Unit 2 Cycle 8 and Unit 3 Cycle 6 outages.

The staff evaluates the P/T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that when the core is not critical P/T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the American Society of Mechanical Engineers (ASME) Code. GL 88-11 advises that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff will use in the review of the P/T limits submittal.

SRP 5.3.2 provides guidance on calculation of the P/T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 the beltline thickness. The critical locations in the vessel for this methodology is the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

3.0 EVALUATION

For the BFN Unit 1 reactor vessel, the licensee determined that lower to low-intermediate girth weld (I.D. SAW WF154), is the limiting material for both the 1/4T and 3/4T locations. The licensee calculated an ART of 87.2°F at the 1/4T location and 61.7°F at the 3/4T location.

For the BFN Unit 2 reactor vessel, the licensee determined that lower to low-intermediate girth weld (I.D. ESW), is the limiting material for both the 1/4T and 3/4T locations. The licensee calculated an ART of 74.8°F at the 1/4T location and an ART of 51.0°F at the 3/4T location.

For the BFN Unit 3 reactor vessel, the licensee determined that lower to low-intermediate girth weld (I.D. ESW), is the limiting material for both the

1/4T and 3/4T locations. The licensee calculated an ART of 74.8°F at the 1/4T location and an ART of 51.0°F at the 3/4T location.

The staff verified that copper and nickel contents and initial RT_{NDT} of the reactor vessel materials agreed with those in the licensee's updated responses to GL 92-01 for BFN Units 1, 2, and 3. The staff used the material properties to perform an independent calculation of the ART values for the limiting materials using RG 1.99, Revision 2. Based on the staff's calculation, the staff verified that the licensee's calculated ARTs for BFN Units 1, 2, and 3 are acceptable.

Substituting the ARTs of BFN Units 1, 2, and 3 limiting materials into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, criticality, and inservice hydrostatic test satisfy the requirements in paragraphs IV.A.2 and IV.A.3 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 preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload 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 20°F for Unit 1, 22°F for Unit 2, and 10°F for Unit 3 provided by the licensee, the staff has determined that the proposed P/T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

The staff has performed an independent analysis to verify the licensee's proposed BFN Units 1, 2, and 3 P/T limits. The staff concludes that the proposed P/T limits for heatup, cooldown, inservice hydrostatic test and criticality are valid for 12 effective full power years, because: (1) the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11, and (2) the material properties and chemistry used in calculating the P/T limits are consistent with or conservative compared to data submitted under GL 92-01; hence, the proposed BFN Units 1, 2, and 3 P/T limits may be incorporated in the TS. In addition, the proposed changes in the Bases section of the TS are consistent with the P/T limits changes and; therefore, are acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. 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 (60 FR 29888). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(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.

6.0 CONCLUSION

The Commission has concluded, based upon 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: September 13, 1995