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

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.158 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO.148 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated March 1, 1991, as superseded September 6, 1991, the Tennessee Valley Authority (the licensee) submitted a request for changes to the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications (TS). The requested changes would revise the pressure/temperature (P/T) limits in the Sequoyah Units 1 and 2 Technical Specifications, Section 3.4. The proposed P/T limits are valid for 16 effective full power years (EFPY) and were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends that RG 1.99, Revision 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society of Testing Materials (ASTM) Standards and the American Society of Mechanical Engineeers (ASME) Codes, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

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Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittiement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees use the methods in Regulatory Guide 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Sequoyah 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 16 EFPY in Sequoyah 1 was the lower shell forging with 0.13% copper (CU), 0.76% nickel (Ni), and an initial RT(ndt) of 73°F; the material with the highest ART at 16 EFPY for Sequoyah 2 was the weld between the intermediate and lower shell forgings with 0.13% copper (Cu), 0.11% nickel (Ni), and an initial RT(ndt) of -4°F.

So far, the licensee has removed two surveillance capsules from each unit. The results from capsules T and U of Sequoyah 1 were published in Westinghouse report WCAP-10340 and Southwest Research Institute Report SwRI 06-8651. The results from capsules T and U of Sequoyah 2 were published in Westinghouse Report WCAP-10509 and Southwest Research Institute Report SwRI 17-8851. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material in Sequoyah 1, the lower shell forging, the staff calculated the ART to be 194.4°F at 1/4T (T = reactor vessel beltline thickness) and 165.6°F for 3/4T at 16 EFPY. The staff used a neutron fluence of 1.16E19 n/sq.cm at 1/4T and 4.11E18 n/sq.cm at 3/4T. The ART was determined by the least squares extrapolation method using the surveillance data from Sequoyah 1. The least squares method is described in Section 2.1 of RG 1.99, Revision 2.

For the limiting beltline material in Sequoyah 2, the weld between the intermediate and lower shell forgings, the staff calculated the ART to be 141.5°F at 1/4T and 103.1°F for 3/4T at 16 EFPY. In this case, the staff used a neutron fluence of 5.15 E18 n/sq.cm at 1/4T and 1.83E18 n/sq.cm at 3/4T. The ART was determined from the Sequoyah 2 surveillance data in the same way.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 1.95°F at 16 EFPY at 1/4T for the same limiting lower shell forging for Sequoyah 1, and an ART of 1.42°F at 16 EFPY at 1/4T for the same limiting weld between the intermediate and lower shell forgings for Sequoyah 2. Since the licensee's ARTs of 195°F and 142°F are almost the same as the staff's ARTs of 194.4°F and 141.5°F, they are acceptable. Substituting the ARTs of 194.4 °F and 141.5°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.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 reference temperature of -49°F for Sequoyah 1 and -13°F for Sequoyah 2, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. For Sequoyah 1, the measured Charpy USE is 58 ft-lb for the lower shell forging, based on data from surveillance capsule U withdrawn at 2.92 EFPY. This is a 19.4% reduction from the unirradiated value of 72 ft-lb. Using the method in RG 1.99, Revision 2, the staff calculated that the Charpy USE of the lower shell forging at the end of life will be 55.2 ft-lb. Data from capsule T was discounted because of change in USE was found there. For Sequoyah 2, the material with the lowest initial USE is the intermediate shell forging with 93 ft-lb. Using the Figure 2 of RG 1.99, Revision 2, the staff calculated that the EOL USE at 1/4T will be 67.9 ft-lb. This number has not been adjusted by surveillance data because applying capsule data would give higher EOL USEs. Since both numbers are greater than 50 ft-lb, they are acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16

EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy Generic Letter 88-11 because the method in RG 1.99, Revision 2, was used to calculate the ARTs. Hence, the proposed P/T limits are acceptable for use in the Sequoyah Units 1 and 2 Technical Specifications.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 13669 and 56 FR 49928). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: March 31, 1992