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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

1.0 INTRODUCTION AND BACKGROUND

By letter dated August 30, 1990, Duke Power Company (the licensee) proposed amendments for McGuire Nuclear Station, Units 1 and 2. The proposed amendments would replace the existing McGuire Units 1 and 2 reactor coolant system heatup and cooldown curves (Technical Specification (TS) Figures 3.4-2, 3.4-3, 3.4-4, and 3.4-5), as well as revise the reactor vessel surveillance capsule withdrawal schedule (TS Table 4.4-5).

The proposed pressure/temperature (P/T) limits for McGuire Units 1 and 2 are valid for 10 effective full power years (EEPY). Both sets of P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. The P/T limits provide up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the NRC staff uses the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 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 TSs for the operation of the plant. In particular, 10 CFR 50.36 (c)(2) requires that limiting conditions of operation be included in the TSs. The P/T limits are among the limiting conditions of operation in the TSs for all commercial nuclear plants in the U.S. Appendices G and H to 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 of constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G to 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 to 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the condition of vessel embrittlement 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 RG 1.99, Rev. 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 to 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 that are made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

2.1 Pressure Vessel Irradiation

The McGuire Unit 2 reactor pressure vessel irradiation analysis updates the calculated neutron fluence to the reactor pressure vessel using measurements obtained from surveillance capsule X. The licensee compared the surveillance capsule measurements to the results of the irradiation calculations.

Both forward and adjoint type calculations were carried out by the licensee in (R,O) geometry using the DOT two-dimensional discrete ordinates code with the SAILOR cross section library. The SAILOR cross section library is ENDF/B-IV based. The anisotropic scattering is treated with a P_3 expansion approximation and the angular quadrature used an S_8 approximation. The adjoint calculations relate the fast neutron flux (E greater than 1.0 MeV) to the surveillance capsule and several azimuthal locations of the pressure vessel inner radius. The importance functions generated from the adjoint analyses, combined with cycle specific source distributions provided absolute neutron exposure at all locations of interest for the first five cycles of irradiation. These cycle specific values include the increased neutron yield per fission due to plutonium buildup as a function of burnup. The cycle specific power distributions were obtained from reload cycle design reports. These values are sufficiently close to the real distribution and are acceptable.

In addition to the E greater than 1.0 MeV fluence, the licensee's calculation includes E greater than 0.1 MeV and the pressure vessel iron displacements per atom (dpa). The licensee's pressure vessel irradiation calculations were carried out with the two-dimensional DOT code, the SAILOR cross section set, and with acceptable approximation; thus, we find them acceptable.

2.2 Neutron Dosimetry

The activation of the passive neutron dosimeters contained in surveillance capsule X was determined by the licensee using established ASTM procedures. The capsule irradiation history was obtained from NUREG-0020. The energy response for each monitor was obtained from ENDF/B-V dosimetry data. The energy spectrum in the location of the dosimeter from an initial estimated value was iteratively adjusted to yield the dosimeter measured activity values. Comparison of the calculated and measured final values of the activities (and corresponding neutron flux values) are in reasonable agreement. The dosimetry was performed with acceptable ASTM standards, and the calculations were performed with acceptable dosimetry data and methods; thus, we find them acceptable.

2.3 McGuire Unit 1 Embrittlement

The NRC staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the McGuire 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 10 EFPY for McGuire 1 was the intermediate shell longitudinal weld seam M1.22 with 0.21% copper (Cu), 0.88% nickel (Ni), and an initial RT_{ndt} of -50°F.

The licensee has removed two surveillance capsules from Unit 1. The results from Capsules U and X in Unit 1 were published in Westinghouse reports WCAP-10786 and WCAP-12354, respectively. The surveillance capsules contained Charpy impact specimens and tensile specimens which were made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, longitudinal weld seam M1.22, the staff calculated the ART to be 164°F at 1/4T (T = reactor vessel beltline thickness) and 110°F at 3/4T. The staff used a neutron fluence of 4.17E18 n/cm^2 at 1/4T and 1.48E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because the limiting material was not in the surveillance capsules.

The licensee calculated the ARTs of 165.5°F at 1/4T and 113°F at 3/4T for the same limiting material, longitudinal weld seam M1.22. The licensee's ARTs are more conservative than the staff's ARTs and, therefore, are acceptable. Substituting the ART of 165.5°F into the 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 to 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor closure vessel 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 that are 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 40°F, 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 Unit 1, based on data from Surveillance Capsule U withdrawn at 5.9 EFPY, the measured Charpy USE is 75 ft-lb for the intermediate shell longitudinal weld (M1.22) metal. This is a 33% reduction from the unirradiated value of 112 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of weld metal M1.22 at the end of life will be above 50 ft-lb and, therefore, is acceptable.

2.4 McGuire Unit 2 Embrittlement

The NRC staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the McGuire 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 10 EFPY was the lower shell forging with 0.15% copper (Cu), 0.80% nickel (Ni), and an initial RT_{ndt} of -30°F.

The licensee has removed two surveillance capsules from Unit 2. The results from Capsules V and X were published in Westinghouse reports WCAP-11029 and WCAP-12556, respectively. The surveillance capsules contained Charpy impact specimens and tensile specimens which were made from base metal, weld metal, and HAZ metal. The licensee proposed to change the withdrawal schedule of Capsules U and W. The staff has determined that the revision is acceptable because the revised withdrawal schedule satisfies ASTM E185-82.

For the limiting beltline material, the lower shell forging, the staff calculated the ART to be 89.5°F at 1/4T (T = reactor vessel beltline thickness) and 60.6°F for 3/4T at 10 EFPY. The staff used a neutron fluence of 3.97E18 n/cm^2 at 1/4T and 1.43E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because the limiting material was not in the surveillance capsule.

The licensee calculated an ART of 90°F at 1/4T for the same limiting forging. The staff judges that the licensee's ART of 90°F is more conservative than the staff's ART of 89.6°F, and it is acceptable. Substituting the ART of 90°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 to 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor closure vessel 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 that are 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 1°F, the staff has determined that the proposed Unit 2 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 Unit 2, based on data from Surveillance Capsule V at 4.2 EFPY, the measured Charpy USE is 85 ft-lb for the intermediate shell forging (05) metal. This is a 9.6% reduction from the unirradiated value of 94 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of shell forging 05 at the end of life will be above 50 ft-lb and, therefore, is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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. 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 these amendments.

4.0 CONCLUSION

The Commission's proposed determination that the amendments involve no significant hazards consideration was published in the Federal Register (55 FR 38599) on September 19, 1990. The Commission consulted with the State of North Carolina. No public comments were received, and the State of North Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Da[illegible] November 15, 1990