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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-316

Introduction

In a letter from R. F. Hering to H. R. Denton dated February 14, 1985, the Indiana & Michigan Electric Company requested an amendment to the DCCNP-2 Technical Specifications. The amendment proposes revised reactor coolant pressure-temperature limits, which will be applicable through twelve (12) effective full power years (EFPY). In support of this amendment, the licensee referenced a Southwest Research Institute report entitled, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 2 Analysis of Capsule Y." This document was transmitted to the staff in a letter from M. P. Alexich to H. R. Denton dated July 20, 1984.

Discussion

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The DCCNP-2 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the initial RT_{NDT} for each reactor vessel material. Technical Specification Table B 3/4.4-1 reports the initial RT_{NDT} for materials in the closure flange and beltline regions of the DCCNP-2 vessel using the method recommended by the staff in Standard Review Plan, Section 5.3.2, Branch Technical Position MTEB 5-2 entitled, "Fracture Toughness Requirements." This method results in an initial RT_{NDT} for the limiting closure flange region material of 33°F, an initial RT_{NDT} for the limiting beltline weld metal of -35°F and an initial RT_{NDT} for the limiting beltline plate of 58°F.

The method recommended by the staff for predicting the increase in RT_{NDT} resulting from neutron irradiation damage is documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of copper and phosphorus in the beltline material. The predicted amount of neutron fluence is dependent upon the neutron flux. The neutron flux is dependent upon the core design. The DCCNP-2 core design was changed to a low leakage core following the second fuel cycle. The licensee plans to utilize low leakage cores for the remaining life of DCCNP-2 plant. Using flux wire measurements and a two dimensional discrete ordinate transport calculation, the Capsule Y Test Report indicates that the peak inside surface neutron flux during core cycle 3 (low leakage core) is calculated to be 1.59 X 10^{10} n/cm^2/sec (E >1MeV) and the peak inside surface neutron flux during core cycles 1 and 2 is calculated to be 1.98 X 10^{10} n/cm^2/sec (E >1MeV). This results in a predicted neutron fluence for twelve EFPY of 3.8 x 10^{18} n/cm^2 (E >1MeV) at the 1/4 T beltline location and 9.4 X 10^{17} n/cm^2 (E >1MeV) at the 3/4 T beltline location.

The amounts of copper and phosphorus in the beltline materials in the DCCNP-2 reactor vessel are reported in FSAR Appendix Q, Question 121.2. Using the method recommended in Regulatory Guide 1.99, Rev. 1 for predicting neutron irradiation damage, the limiting material in the DCCNP-2 reactor vessel would be plate heat no. C5556-2. The material used in the reactor vessel surveillance program is from plate heat no. C5521-2. Both plates heat no. C5556-2 and C5521-2 were supplied by Lukens Steel, have been heat treated to an equivalent microstructure and have equivalent chemical composition. Hence, the test results from the surveillance material could be used to demonstrate the effect that neutron irradiation would have on the limiting beltline material. In Table 1 we have compared the amount of increase in RT_{NDT} resulting from neutron irradiation damage observed on capsule material from plate heat no. C5521-2 to that predicted by the formula in Regulatory Guide 1.99, Rev. 1. This comparison indicates that at low neutron fluence, the formula in the guide is nonconservative. However, in instances such as this, the guide recommends that neutron irradiation damage be estimated by a straight-line interpolation on a logarithmic plot between credible surveillance points and extrapolation from the lower surveillance data point, using the slope of the family of lines in Figure 1 of the guide. This is the method used by the licensee to predict neutron irradiation damage to the limiting beltline material.

Evaluation/Conclusion

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage to the limiting beltline material was estimated using the extrapolation method recommended in Regulatory Guide 1.99, Rev. 1. Our conclusion is that the proposed pressure temperature limits meet the safety margins of Appendix G, 10 CFR 50 for (12) twelve EFPY and may be incorporated into the DCCNP-2 technical specifications.

TABLE I

INCREASE IN REFERENCE TEMPERATURE, RT_{NDT}, FOR SURVEILLANCE MATERIAL HEAT NO. C5521-2

Neutron Fluence (n/cm^2, E >1 MeV)	Increase in RT_{NDT} (°F) Observed from Capsule Test Data	Increase in RT_{NDT} (°F) Predicted by Regulatory Guide 1.99, Rev. 1
2.7×10^{18}	80	65
7.0×10^{18}	100	105

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

Conclusion

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

Dated: June 27, 1985

Principal Contributors:

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