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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

In a letter from L. G. Kuncl to USNRC dated October 28, 1987 and supplemented by a letter from G. A. Trevors to USNRC dated February 22, 1988, the Nebraska Public Power District (the licensee) proposed to amend Facility Operating License No. DPR-46 for the Cooper Nuclear Generating Station (Cooper). The amendment proposes to revise the reactor coolant system pressure-temperature limits and surveillance capsule withdrawal schedule, which are contained in Section 3.6 and 4.6 of the Cooper Technical Specifications (TS). The RT_{NDT} Shift Curve (Figure 3.6.1) is to be deleted, Non-Nuclear Heatup/Cooldown Curve (Figure 3.6.1.a) and Core Critical Curve (Figure 3.6.1.b) are to be applicable for 12 effective full power years (EFPY), and Pressure Test Curves are to be applicable for 8, 10 and 12 EFPY (Figure 3.6.2). The licensee proposes to revise the capsule withdrawal schedule to require withdrawal of the next capsule at 15 EFPY, and the remaining capsule at 32 EFPY. The bases for these changes are the test results from the Cooper surveillance program, which are contained in a letter from G. A. Trevors to USNRC dated July 6, 1987.

2.0 DISCUSSION

Pressure-Temperature Limits: Pressure-Temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR Part 50, which became effective on July 16, 1983. Pressure-Temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR Part 50 are dependent upon the initial reference temperature (RT_{NDT}) for the limiting materials in the beltline, and closure flange regions of the reactor vessel and the increase in reference temperature resulting from neutron irradiation damage to the limiting beltline material. The Cooper reactor vessel was procured to earlier ASME Code requirements, which did not specify fracture toughness testing to determine the initial RT_{NDT} for each vessel material. Appendix G, 10 CFR Part 50 indicates that vessels fabricated to earlier ASME Code requirements must provide supplementary data and analyses to demonstrate that the vessel material's fracture toughness data and material analysis requirements are equivalent to that specified in later editions of the ASME Code.

The Cooper reactor vessel was fabricated by Combustion Engineering (CE). The beltline was fabricated by welding plates together and the closure flange regions were fabricated by welding plates and forgings together.

The initial RT_{NDT} for plate materials was determined by extrapolating the existing data using a calculation method developed by General Electric (GE). The GE method is based on test results from 24 plates reported in Welding Research Council Bulletin 217 and from 22 plates reported in the LaSalle FSAR. The initial RT_{NDT} for the forging materials was determined using the method recommended by the staff in NRC Branch Technical Position MTEB 5-2. This branch technical position is documented in Standard Review Plan 5.3.2, "Pressure-Temperature Limits" of NUREG-0800, Rev. 1, July 1981. The initial RT_{NDT} used for weld metals was the two standard deviation upper bound value used by the staff for CE weld metals in SECY-82-465, "Pressurized Thermal Shock." These methods result in an initial RT_{NDT} for the limiting beltline base metal and weld metal of 14°F, and -22°F, respectively, and an initial RT_{NDT} for the limiting closure flange region material of 20°F.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee by extrapolating the surveillance data at the rate documented in Regulatory Guide (R.G.) 1.99, Rev. 1, "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 residual elements in the beltline materials. The neutron fluence predictions were upper bound estimates, which were calculated using measurements from passive neutron flux monitors and by analysis, which was made with the DOT two-dimensional discrete ordinate code and the SN1D one-dimensional computer code. Inputs into the analysis included 26 neutron energy groups, cross-sections from ENDF B-IV, P3 expansion of the scattering cross section, average measured neutron spectra for BWRs at the GE Test Reactor at Vallecitos, and power distributions representative of time-averaged conditions derived from Cooper plant specific cycles. The neutron spectra used in the analysis are documented in NEDO-24793, which is contained in the licensee's letter dated February 22, 1988.

The measured increase in reference temperature for the Cooper surveillance materials are compared in the table below, to the values predicted using R.G. 1.99, Rev. 1 and R.G. 1.99, Rev. 2. which has been approved and is awaiting publication as a final guide. The increase in reference temperature measured from the surveillance material significantly exceeds the values predicted using the formula in R.G. 1.99, Rev. 1. The increase in reference temperature for the weld metal is less than the value predicted using the method in R.G. 1.99, Rev. 2. The increase in reference temperature for the plate material is slightly greater than the value predicted using the method in R.G. 1.99, Rev. 2. The surveillance data indicates that R.G. 1.99, Rev. 1 underpredicts the effect of neutron irradiation on the Cooper beltline material, while R.G. 1.99, Rev. 2 conservatively predicts the effect of neutron irradiation of the beltline weld metal and slightly underpredicts the effect of neutron irradiation on the Cooper beltline plate material.

Surveillance Capsule Test Results

Material	Increase in Ref. Temp. Measured from Surveillance Material (°F)	Increase in Ref. Temp. Predicted by R.G. 1.99, Rev. 1 (°F)	*Increase in Ref. Temp. Predicted by R.G. 1.99, Rev. 2 (°F)	Ratio of Measured to Predicted by R.G. 1.99, Rev. 1 (°F)
Plate	74	31	69	2.39
Weld Metal	55	34	83	1.62

*Increase in Ref. Temp. are mean plus two standard deviation values

The Pressure-Temperature limits proposed by the licensee were calculated using the increase in reference temperature formula in R.G. 1.99, Rev. 1 with a correction to account for the underpredication of this method compared to the surveillance material. The formula in R.G. 1.99, Rev. 1 consists of a chemistry factor and a fluence factor. The licensee increased the chemistry factor in this formula by the ratio of the measured increase in reference temperature from the surveillance material to the values predicted by the formula in R.G. 1.99, Rev. 1 (this ratio is reported in the last column in Table I). The licensee's method of predicting the increase in reference temperature results in adjusted reference temperature (ART) values for the limiting beltline material of 110°F, 102°F, and 93°F at 12 EFPY, 10 EFPY and 8 EFPY, respectively. The ART is the sum of the initial RT_{NDT} and the increase in reference temperature resulting from neutron irradiation. The licensee indicates that this method results in a predicted final end-of-life ART (the ART at 32 EFPY) value of 171°F for the limiting beltline material.

The ART values for the limiting beltline material using the formula in R.G. 1.99, Rev. 2 are 95°F, 91°F, and 86°F at 12 EFPY, 10 EFPY and 8 EFPY, respectively. The final ART value for the limiting beltline material using the formula in R.G. 1.99, Rev. 2 is 144°F. Since the ART values used by the licensee to calculate the proposed Pressure-Temperature limits are greater than the values predicted using the formula in R.G. 1.99, Rev. 2, the proposed Pressure-Temperature limits will meet R.G. 1.99, Rev. 2.

To confirm that the Pressure-Temperature limits proposed by the licensee will meet the safety margins of Appendix G, 10 CFR Part 50 for the proposed operating periods, 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 staff's calculation includes the licensee's ART values. Our calculations confirm that the proposed Pressure-Temperature limits meet the safety margins of Appendix G, 10 CFR Part 50 for the operating periods identified on the curves.

Surveillance Program: The amendment request includes editorial charges to the Technical Specifications requirements for the withdrawal schedule for the remaining two capsules. The existing Technical Specifications define two withdrawal schedules, both stated in terms of service life. One schedule is for use based on the adjusted reference temperature not exceeding 100 deg. F over the life of the vessel. The second schedule is to be used in the event surveillance specimens indicate a shift of the Charpy V-notch fracture energy curve greater than predicted. Since the test data indicates a shift greater than predicted, the Technical Specifications are being "cleaned-up" to reflect use of the latter schedule in accordance with requirements. This is a simple editorial change to the Technical Specifications involving no change in the surveillance program requirements. Also, the Technical Specifications are being revised to specify capsule removal intervals in terms of "EFPY", instead of "service life". Since the service life is defined by ASTM E-185-82 as 32 EFPYs, and the revised intervals in terms of EFPYs correspond to the original intervals in terms of service life, this is also a simple editorial change in actual surveillance program requirements. Because these changes are considered editorial they are acceptable.

The staff believes that changes should be made to the surveillance program. The present withdrawal schedule is based on original assumptions that (1) the increase in reference temperature resulting from neutron exposure would be less than 100 deg. F., and (2) the surveillance specimens would receive greater fluence than the vessel wall. The licensee's analysis indicates that the surveillance specimen neutron exposure lags the vessel wall material. The licensee's analysis and an independent staff analysis indicate that, as noted above, the increase in reference temperature will be greater than 100 deg. F at end of life. Since the original assumptions were incorrect, the surveillance plan should be revised. Appendix H, 10 CFR Part 50 requires, to the extent practical, that the capsule withdrawal program meet the requirements of ASTM E-185-82. When the predicted increase in reference temperature is greater than 100°F, ASTM E 185-82 recommends that 4 capsules be withdrawn and the surveillance program monitor the long term effects of neutron irradiation. The proposed technical specifications indicate that the removal and analysis of the remaining capsules is for the second to be removed at 15 EFPY, and the third and last to be removed at 32 EFPY. Our findings are that irradiation damage is in excess of predictions using Regulatory Guide 1.99 Rev. 2 criteria, irradiation damage will exceed 100 deg. F, and the surveillance specimens are receiving less exposure than the vessel wall. Therefore, to assure maintenance of safety margins beyond 12 EFPY and to support possible life extension, the staff recommends that the schedule for withdrawal of the second capsule should be accelerated to 12 EFPY and the schedule for withdrawal of the third should be determined based on the findings of analysis of the second capsule. In addition, the licensee should begin planning for possible insertion of a fourth capsule into the Cooper reactor vessel, perhaps with reconstituted specimens from the first capsule. The staff will request the licensee to review the surveillance program in consideration of these recommendations.

3.0 EVALUATION

Based on our review, we find that:

1. The data and analysis provided by the licensee demonstrate that the vessel material's fracture toughness is equivalent to that specified in later editions of the ASME Code and the initial RT_{NDT} values proposed for the Cooper reactor vessel material are acceptable for use in calculating the Cooper Pressure-Temperature limits.

2. The licensee's method of calculating neutron fluence is acceptable and may be used to predict the increase in reference temperature resulting from neutron irradiation.

3. Since ART values used to calculate the Pressure-Temperature limits were derived from the surveillance material test results and are greater than the values calculated using the method documented in R.G. 1.99, Rev. 2, the proposed Pressure-Temperature limits adequately account for neutron irradiation.

4. Based on the above findings and the staff's confirmatory calculations, the proposed Pressure-Temperature limits meet the safety margins of Appendix G, 10 CFR Part 50.

5. The proposed changes to the capsule withdrawal schedule are editorial in nature.

6. The proposed Pressure-Temperature limits and capsule withdrawal schedule may be incorporated into the Cooper Technical Specifications. However, the surveillance program should be promptly reevaluated to permit early development of plans to assure maintenance of safety margins for operation beyond 12 EFPY.

4.0 ENVIRONMENTAL CONSIDERATION

The 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 exposures. 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. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 26, 1988

Principal Contributors: B. Elliot/W. Long/L. Lambros