April 26, 1988

Docket No. 50-298

Mr. George A. Trevors, Division
Manager - Nuclear Support
Nuclear Power Group
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68601

Dear Mr. Trevors:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. 65793)

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications in response to your application dated October 28, 1987 (Change No. 48), as modified by letter dated February 22, 1988.

The amendment changes the Technical Specifications to revise reactor vessel pressure-temperature limits for heatup and cooldown, normal operation and pressure testing. As stated in the enclosed Safety Evaluation, we request that you reevaluate the surveillance program. Please submit plans for such a reevaluation within 180 days.

Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/s/

William O. Long, Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 120 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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JCalvo	ARM/LFMB	OGC-Rockville	DHagan
EJordan	PD4 Plant File		

DOCUMENT NAME: COOPER AMENDMENT TAC 65793

PD4/LA MC for	PD4/PM W	OGC-Rockville	PD4/D MC
PNoonan	WLong:sr		JCalvo
04/25/88	04/15/88	04/20/88	04/25/88

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

April 26, 1988

Docket No. 50-298

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Manager - Nuclear Support
Nuclear Power Group
Nebraska Public Power District
Post Office Box 499
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William O. Long

William O. Long, Project Manager
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Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
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cc w/enclosures:
See next page

Mr. George A. Trevors
Nebraska Public Power District

Cooper Nuclear Station

cc:
Mr. G. D. Watson, General Counsel
Nebraska Public Power District
P. O. Box 499
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Cooper Nuclear Station
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Dennis Grams, Director
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Lincoln, Nebraska 68509-8922

Mr. Larry Bohlken, Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, Nebraska 68305

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 218
Brownville, Nebraska 68321

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. Harold Borchart, Director
Division of Radiological Health
Department of Health
301 Centennial Mall, South
P. O. Box 95007
Lincoln, Nebraska 68509-5007

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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District (the licensee) dated October 28, 1987, and modified by letter dated February 22, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 120 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 26, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages
132
133
146
147
154
155
156
157

3.6 Primary System Boundary

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

2. During operation where the core is critical or during heatup by non-nuclear means or cooldown following shutdown, the reactor vessel metal and fluid temperatures shall be at or above the temperatures shown on the limiting curves of Figures 3.6.1.a or 3.6.1.b.

3. The reactor vessel metal temperatures for the botton head region and beltline region shall be at or above the temperatures shown on the limiting curves of Figure 3.6.2 during inservice hydrostatic or leak testing. The Adjusted Reference Temperature (ART) for the beltline region must be determined from the appropriate beltline curve (8, 10, or 12 EFPY) depending on the current accumulated number of effective full power years (EFPY). The ART curve for the bottom head is valid to 12 EFPY.

4.6 Primary System Boundary

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 50°F.

   a. Bottom head drain.

   b. Recirculation loops A and B.

2. Reactor vessel temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

3. Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-73 to the degree possible.

3.6.A (cont'd.)

4.6.A (cont'd.)

The schedule for withdrawal of the remaining two capsules is based on ASTM E185-82 and is as follows:

Second Capsule: 15 EFPY
Third Capsule: 32 EFPY

4. The Reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 80°F.

4. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

5. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

6. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

6. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

Thermal and Pressurization Limitations

The requirements for the reactor vessel have been identified by evaluating the need for its integrity over the full spectrum of plant conditions and events.

This is accomplished through the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the reactor vessel. The limits expressed in the technical specification for the applicable operating states are taken from the actual Nuclear Safety Operational Requirements for the reactor vessel as given in Subsection IV-2.8 of the Updated Safety Analysis Report.

The components of the nuclear system pressure boundary are constructed so that its initial maximum nil-ductility transition temperature (RT NDT) is not greater than 40°F, as cited in Subsection IV-2.5 of the Updated Safety Analysis Report. The heatup-cooldown and hydrostatic test minimum pressurization temperatures were calculated to comply with the recommendations of Appendix G of Section III, ASME Boiler and Pressure Vessel Code, 1972 Summer Addendum.

The temperature versus pressure limits when critical which are presented in Figure 3.6.1.b assure compliance with Appendix G of 10CFR50.

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the O-rings of the head and vessel flange. The reactor vessel head flange and head are constructed so that their initial maximum NDTT is 20°F, as cited in Paragraph IV-2.5 of the Updated Safety Analysis Report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 80°F (20°F + 60°F). The total integrated neutron flux in the head flange region will be less than that at the core mid-plane level by a factor of 10^{-3} or 10^{-4}, therefore, the maximum calculated fluence in the head flange region will be far below 1×10^{17} nvt. With such a low total integrated neutron flux in the head flange region, there will be no detectable or significant NDTT shift, and the minimum stud tightening temperature remains at 80°F.

The reactor vessel is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for a pressure of 1250 psig. The pressure limit of 1035 psig represents the maximum expected operating pressure in the steam dome when the station is operating at design thermal power. Observation of this limit assures that the operator remains within the envelope of conditions considered by Chapter 14 of the Updated Safety Analysis Report.

Stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. The specific conditions analyzed included a maximum of 120 cycles of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F. The expected number of normal heatup and cooldown cycles to which the vessel will be subjected is 80.

As described in the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The first surveillance capsule was removed at 6.8 EFPY of operation and base metal, weld metal and HAZ specimens were tested. In addition, flux wires were tested to experimentally determine the integrated neutron flux (fluence) at the surveillance capsule location. The test results are presented in General Electric Report MDE-103-0986. Measured shifts in RT_{NDT} of the base metal and weld metal were compared to predicted values per Regulatory Guide 1.99, Revision 1. The measured values were higher than predicted, so the 1.99 methods were modified to reflect the surveillance data. The test results for the flux wires were used with analytically determined lead factors to determine the peak end-of-life (EOL) fluence at the 1/4 T Vessel wall depth. The value corresponding to 40 years operation (32 EFPY) is 1.5×10^{18} n/cm^2.

The adjusted reference temperature (ART) of a beltline material is defined as the initial RT_{NDT} plus the RT_{NDT} shift due to irradiation. The curves of Figures 3.6.1.a and 3.6.1.b reflect a beltline ART of 110°F, making them valid for operation up to 12 EFPY. Figure 3.6.2, the pressure test curve, includes curves with ART values for 8, 10 and 12 EFPY to provide more flexibility in pressure testing. Figure 3.6.2 also has a separate curve for the bottom head region. The bottom head curve does not shift with increased operation. Therefore, the bottom head temperature can be monitored against lower temperature requirements than the beltline during pressure testing.

B. Coolant Chemistry

Materials in the primary system are primarily Type-304 stainless steel and Ziracloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams[1], where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel, and at high oxygen concentration little chloride is required to cause cracking. These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

[1] W. L. Williams, Corrosion 13, 1957, p. 539t.

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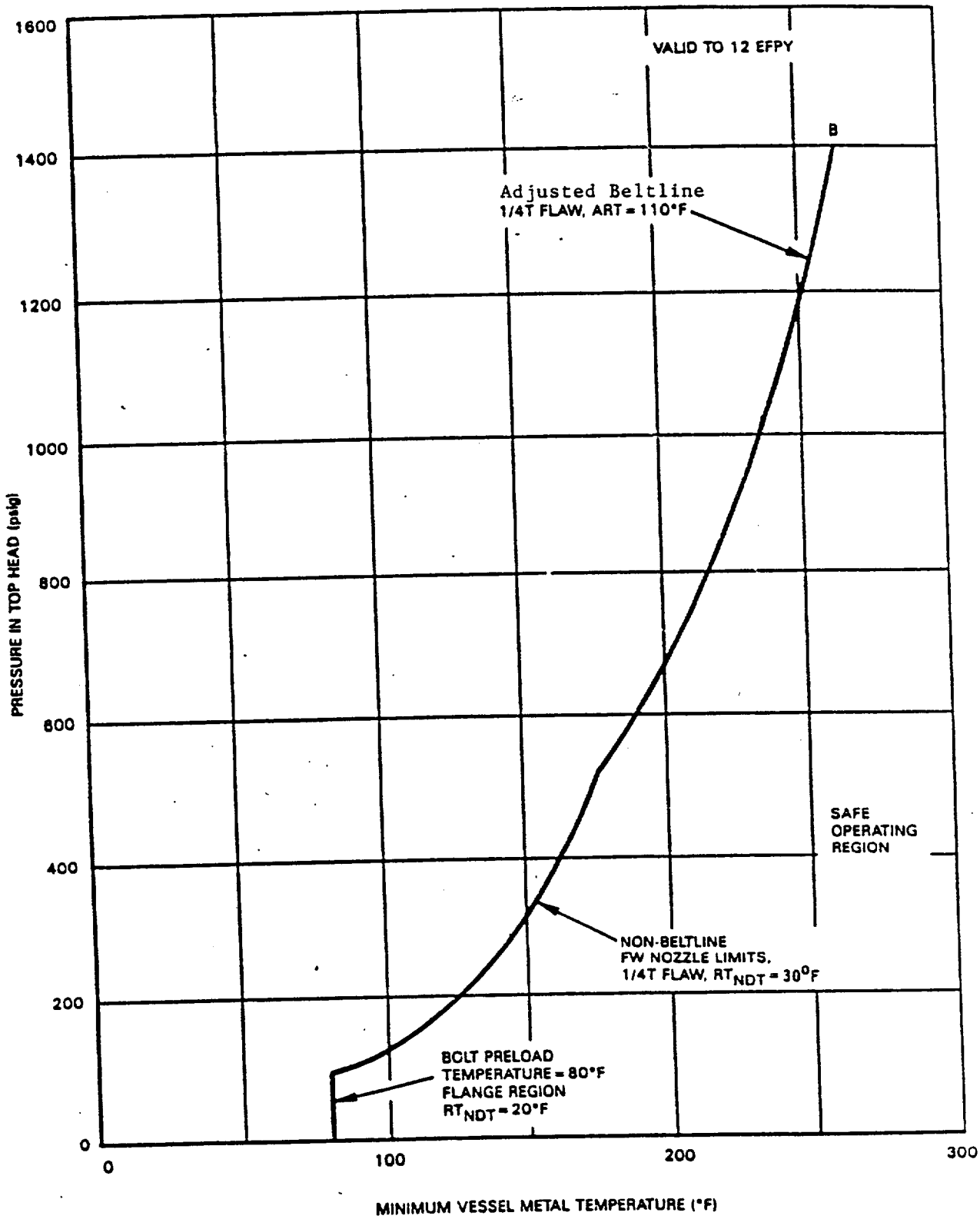

Figure 3.6.1.a Minimum Temperature for Non-Nuclear Heatup or Core Cooldown Following Nuclear Shutdown

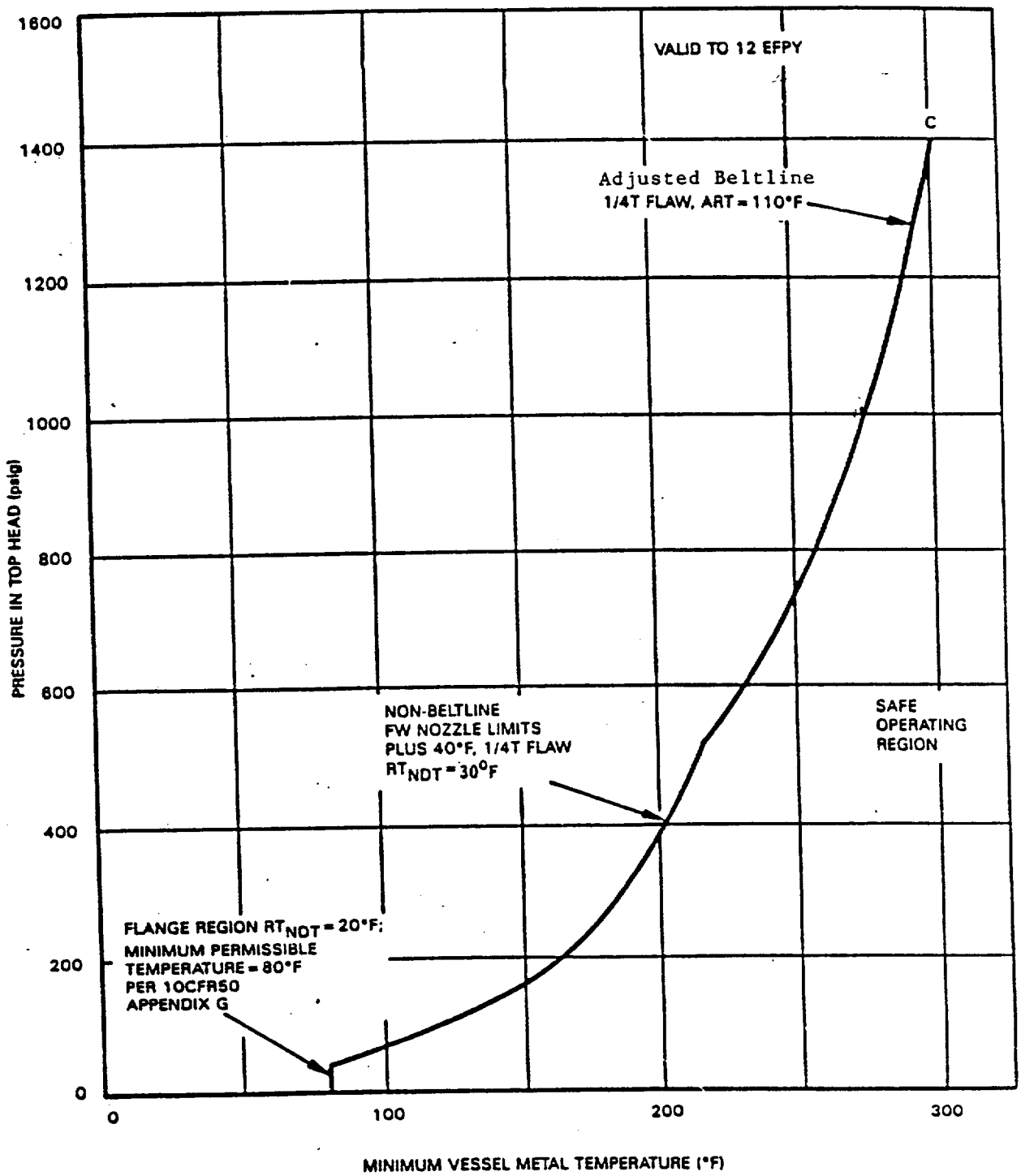

Figure 3.6.1.b Minimum Temperature for Core Operation (Criticality) - Includes 40°F Margin Required by 10CFR50 Appendix G

Amendment No. 120

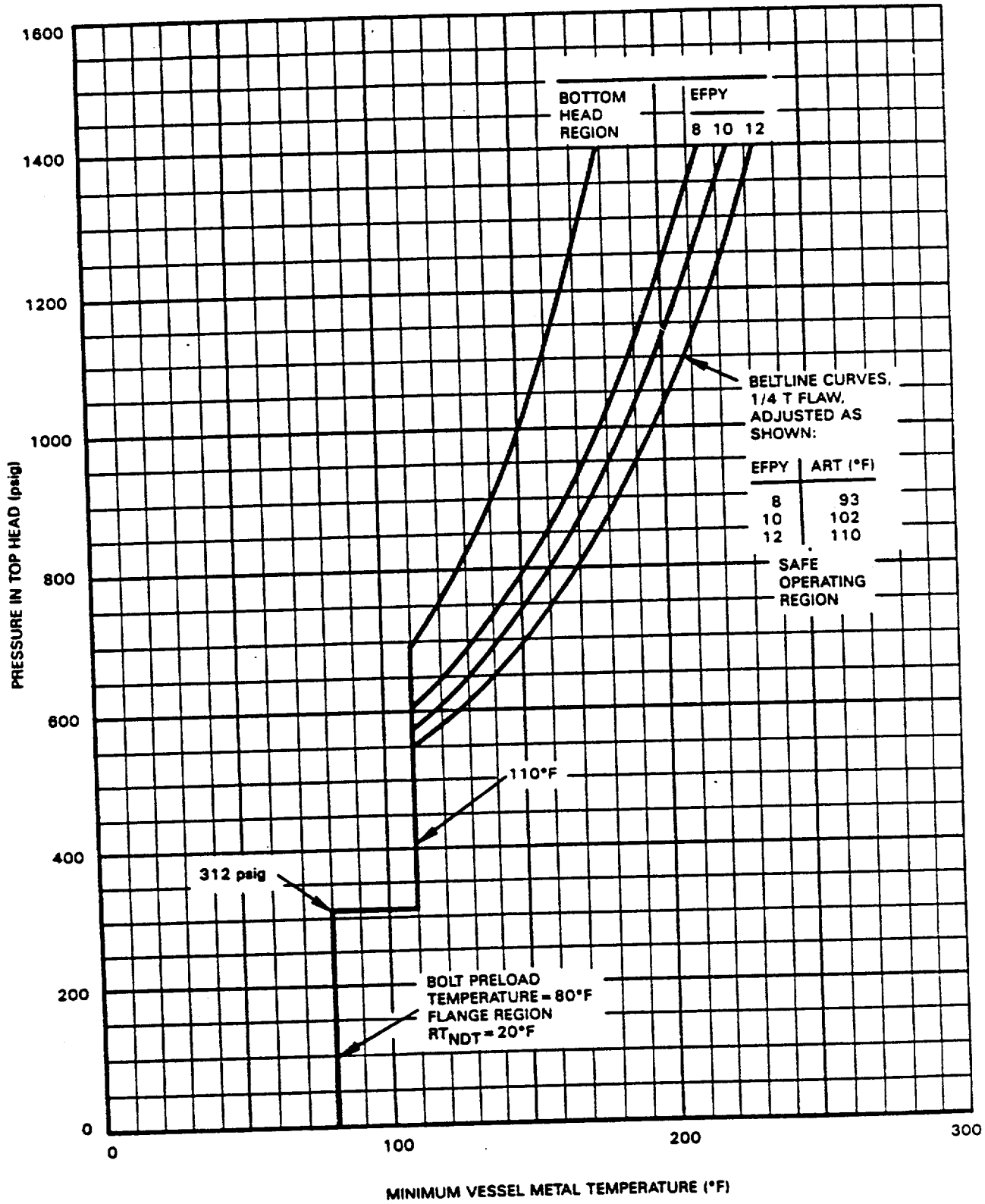

Figure 3.6.2 Minimum Temperature for Pressure Tests Such as Required by Section XI

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.

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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 changes 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.

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