October 13, 1992

Docket No. 50-298

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

Dear Mr. Horn:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. M84240)

The Commission has issued the enclosed Amendment No. 155 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 July 28, 1992.

The amendment changes the Technical Specifications to validate the existing pressure vs. temperature operating limit curves for Cooper beyond the current 12 effective full-power years and remove the vessel material surveillance capsule withdrawal schedule from the Cooper Technical Specifications in accordance with the guidance in Generic Letter 91-01.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Roby B. Bevan, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 155 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	NRC/Local PDR	PD4-1 Reading	G. Hill (4)
M. Virgilio	J. Larkins	P. Noonan	
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Wanda Jones(MS7103)	C. Grimes(MS11E22)	PD4-1 Plant File	
OPA(MS2G5)	OC/LFMB(MS4503)	P. Harrell, RIV	
R. Bevan (2)	B. Boger		

Minor editorial corrections DONE RBB 10/8

OFC	LA:PD4-1	PM:PD4-1	OGC	D:PD4-1
NAME	PNoonan	RBevan:pk	MZOBLER	JLarkins
DATE	10/5/92	10/5/92	10/7/92	10/13/92

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

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Project Directorate IV-1
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Mr. Guy R. Horn
Nuclear Power Group Manager

Cooper Nuclear Station

cc:

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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. 155
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 July 28, 1992, 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, 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.

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. 155, 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

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 13, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 155

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
132	132
133	133
147	147
154	154
155	155
156	156
157	157
158	158

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6 Primary System Boundary	4.6 Primary System Boundary
Applicability:	Applicability:
Applies to the operating status of the reactor coolant system.	Applies to the periodic examination and testing requirements for the reactor cooling system.
Objective:	Objective:
To assure the integrity and safe operation of the reactor coolant system.	To determine the condition of the reactor coolant system and the operation of the safety devices related to it.
Specification:	Specification:
A. Thermal and Pressurization Limitations	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.	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. During operation where the core is critical or during heatup by nonnuclear 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. This specification applies when the reactor vessel head is tensioned.	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. 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 (13, 18, or 21 EFPY) depending on the current accumulated number of effective full power years (EFPY).	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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.A (cont'd.)	4.6.A (cont'd.)
	The reactor vessel surveillance specimens shall be removed and examined to determine changes in their material properties as required by 10 CFR 50 Appendix H.
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.

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 which was in effect at that time. 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.

Subsequent to this evaluation, the NRC issued Regulatory Guide 1.99, Revision 2. This revision requires that two surveillance capsules be tested before the test results are factored into the adjusted reference temperature (ART) shift predictions. The adjusted reference temperature of a beltline material is defined as the initial RT_{NDT} plus the RT_{NDT} due to irradiation. Therefore, the curves developed from the initial surveillance capsule testing were re-evaluated in accordance with the guidance provided in Regulatory Guide 1.99 Revision 2. Based strictly on the chemistry factors provided in Regulatory Guide 1.99, Revision 2, and considering each beltline material chemistry and peak fluence at a given EFPY, the pressure-temperature curves in Figures 3.6.1.a and 3.6.1.b, which reflect a beltline ART of 110°F, were determined to be valid for 21 EFPY. Figure 3.6.2, the pressure test curve, was re-evaluated in like manner and includes curves for 13, 18 and 21 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. The surveillance capsule withdrawal schedule for the remaining specimens is located in Section IV.2.7 of the CNS USAR.

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.

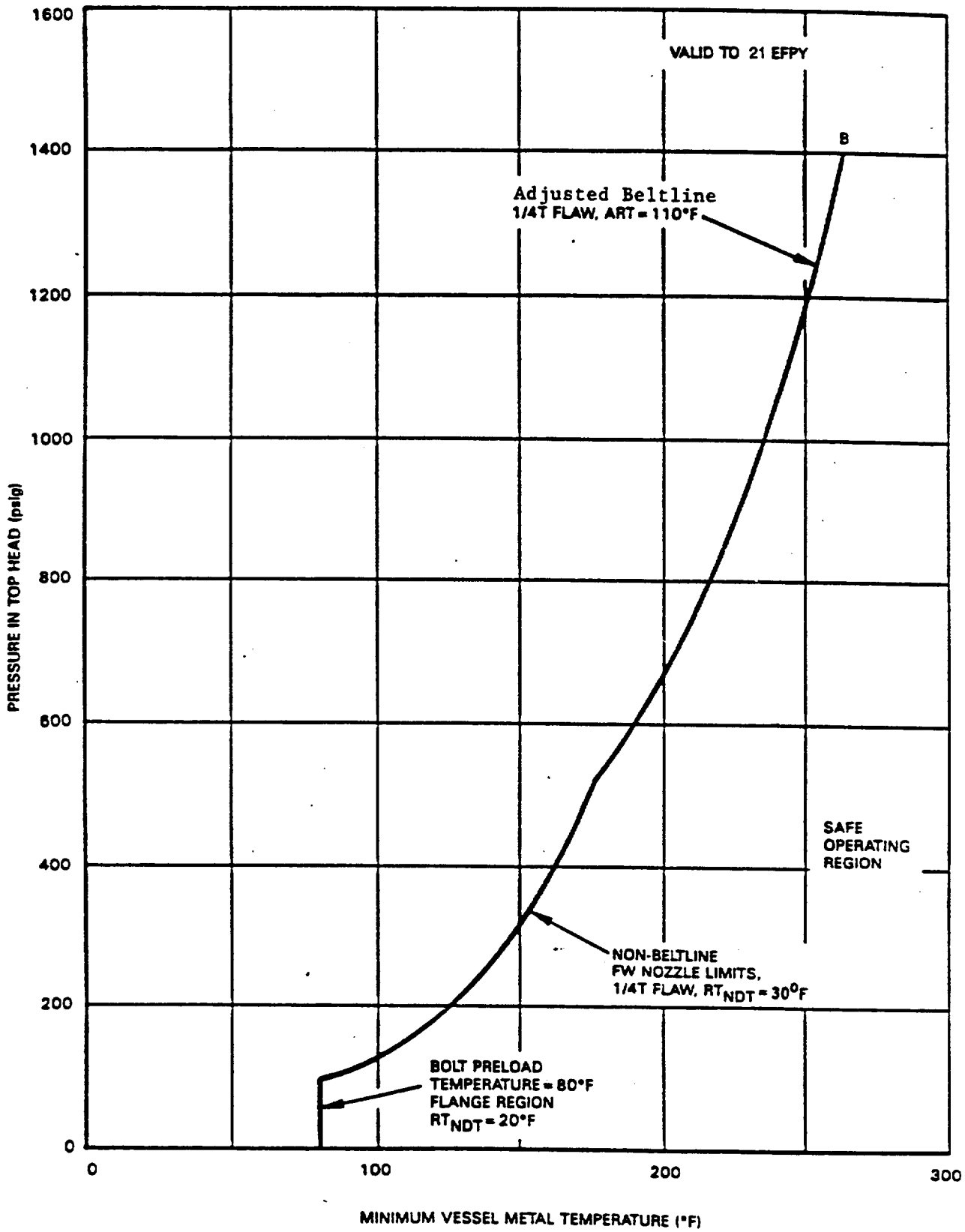

Figure 3.6.1.a Minimum Temperature for Non-Nuclear Heatup or Core Cooldown Following Nuclear Shutdown

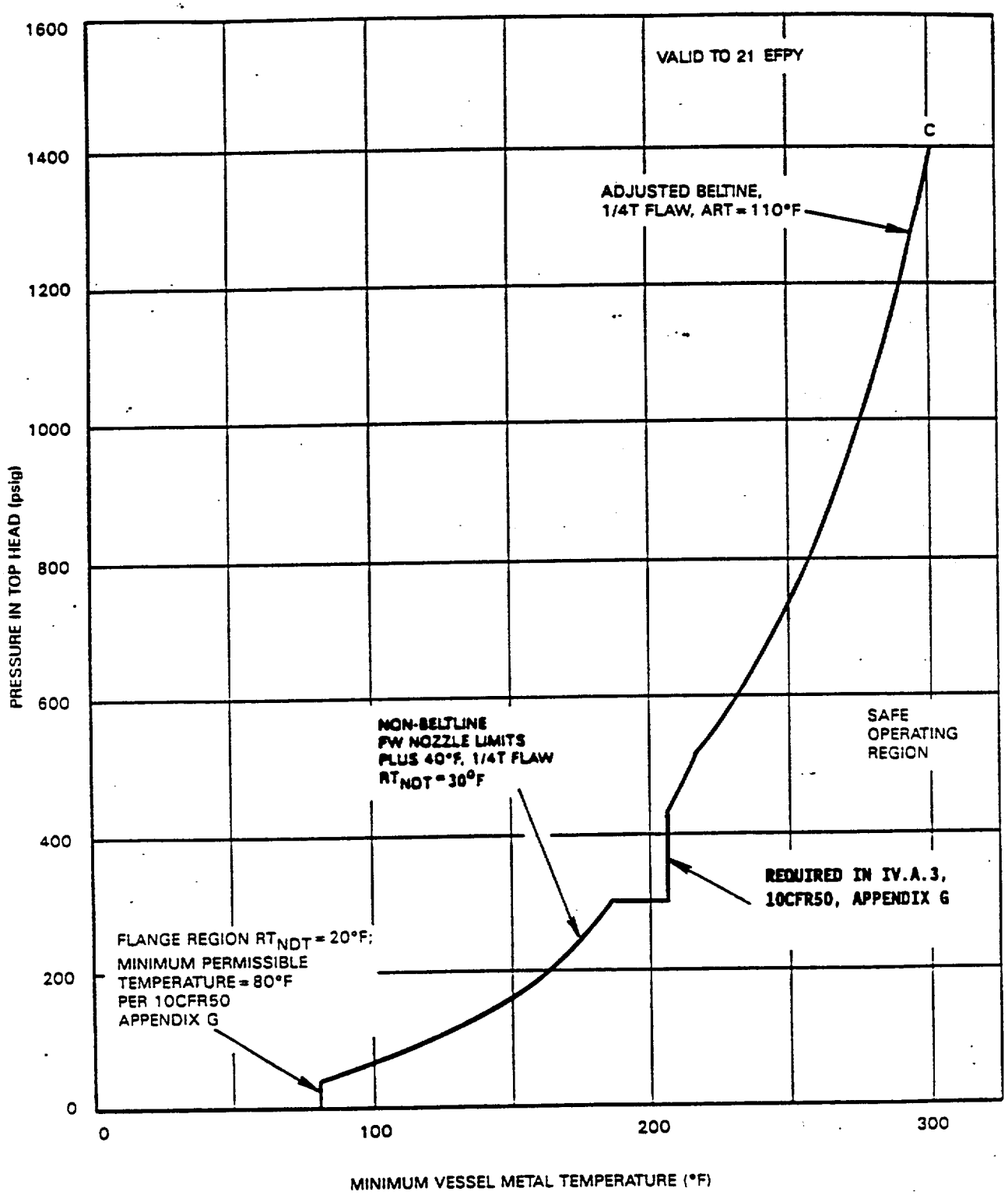

Figure 3.6.1.b Minimum Temperature for Core Operation (Criticality) - Includes 40°F Margin Required by 10CFR50 Appendix G

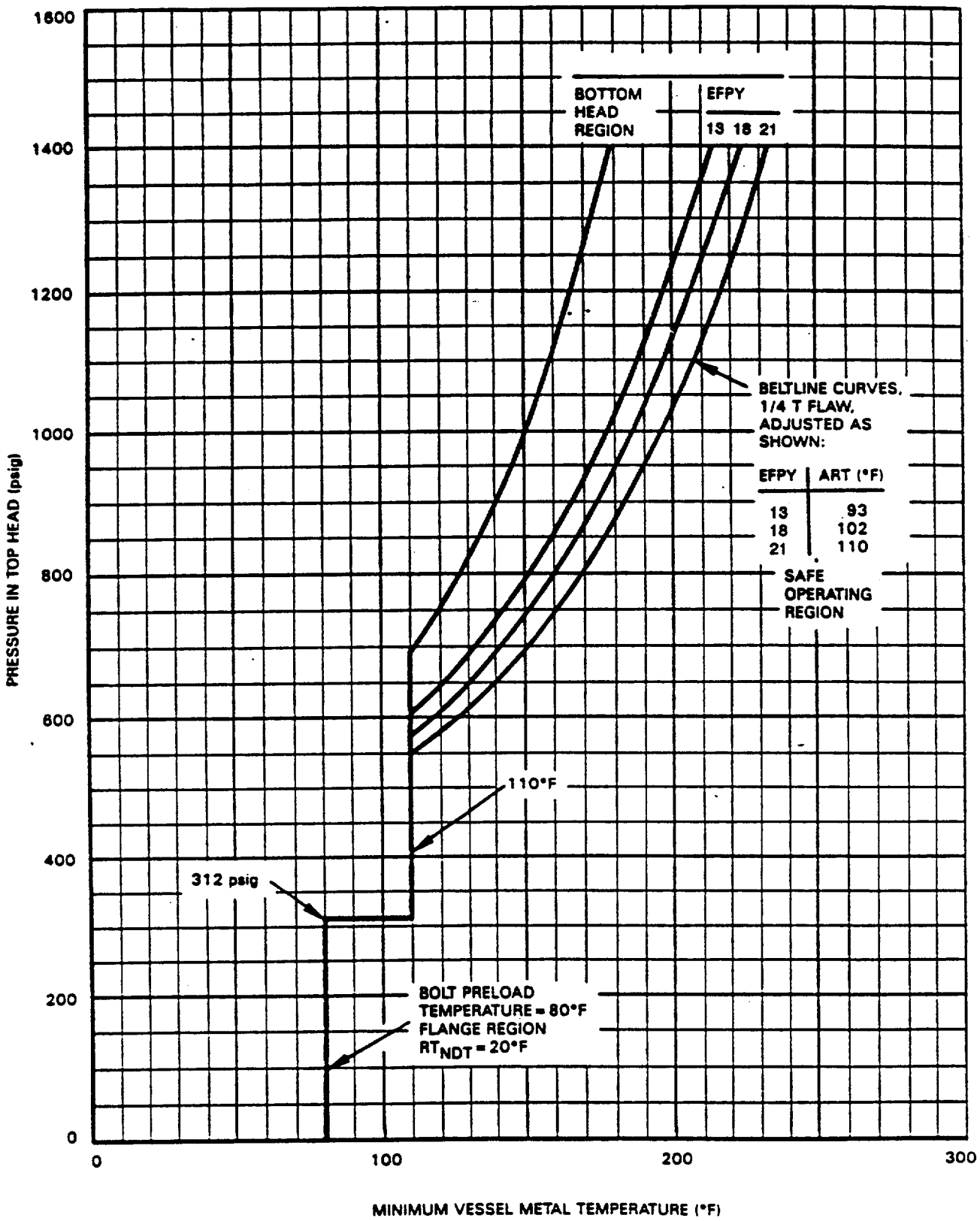

Figure 3.6.2 Minimum Temperature for Pressure Tests Such as Required by Section XI

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated July 28, 1992, Nebraska Public Power District (the licensee) submitted a request for changes to the Cooper Nuclear Station (Cooper) Technical Specifications (TS). The requested changes would validate the existing pressure vs. temperature operating limit curves for Cooper beyond the current 12 effective full-power years and remove the vessel material surveillance capsule withdrawal schedule from the Cooper TS in accordance with the guidance in Generic Letter 91-01.

The licensee proposed the following: 1) revise pressure/temperature (P/T) limits for heatup, cooldown, and criticality to 21 effective full-power years (EFPY), 2) revise P/T limits for hydrostatic or leak test to 13, 18, and 21 EFPY, and 3) relocate the capsule withdrawal schedule for the surveillance of reactor vessel materials to the Updated Safety Analysis Report (USAR).

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials under the American Society of Mechanical Engineers (ASME) Code and, in particular, that the beltline materials in the surveillance capsules be tested under Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to American Society for Testing and Materials (ASTM) Standards. These tests define the extent 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 and permittees use the methods in Regulatory Guide (RG) 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. RG 1.99 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.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Cooper reactor vessel under RG 1.99, Rev. 2. The staff determined that the limiting material with the highest ART at 13 and 18 EFPY is the lower shell plate, G-2803-1, with 0.2% Copper (Cu), 0.67% Nickel (Ni), and initial RTndt of 14°F. The limiting material for 21 EFPY is the axial welds in the lower shell with 0.35% Cu, 1.00% Ni, and an initial RTndt of -50°F. The licensee applied maximum default values in RG 1.99 to the copper and nickel contents of the axial weld because actual contents are unavailable.

At the 1/4T location (T = reactor vessel beltline thickness), the staff calculated the ART of 92.4°F, 100.7°F, and 107.6°F for 13, 18, and 21 EFPY, respectively. The staff extrapolated a neutron fluence of 1.7E18 at end of life to various EFPYs and locations to calculate the ARTs.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 93°F, 102°F, and 110°F at 1/4T for 13, 18, and 21 EFPY, respectively. The licensee's ARTs and the staff's ARTs differ because the licensee rounded off its ARTs conservatively.

Substituting the staff's ARTs into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, criticality, 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.A.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. Paragraph IV.A.3 of Appendix G states that "...an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload...." Based on the reference temperature of 20°F for the reactor closure flange at the Cooper Nuclear Station, the staff determined that the bolt preload temperature of 80°F in the proposed P/T limits satisfies Section IV.A.2 of Appendix G.

The staff also verified that the proposed P/T limits for the bottom head region of the reactor vessel are acceptable because they satisfy Standard Review Plan (SRP) 5.3.2.

The licensee removed the first surveillance capsules during Reload 9, Cycle 10 refueling outage in 1985 at about 6.8 EFPY. The results from Capsule No. 1 were published in General Electric report MDE-103-0986. Capsule No. 2 was removed during the Reload 14, Cycle 15 refueling outage in late 1991 at about 11 EFPY. The licensee has yet to publish the test results of Capsule No. 2. This is acceptable because Appendix H to 10 CFR Part 50 allows one year for the test results to be reported for NRC review. The licensee has stated that it will re-evaluate the P/T limits based on test results of Capsule No. 2 and will propose changes to the P/T limits and the withdrawal schedule for the remaining capsules in the vessel if warranted.

Based on the NRC recommendation during its review of Amendment No. 120, the licensee has committed to reconstitute the specimens from the removed Capsule No. 2 and re-insert the reconstituted specimens into the reactor vessel cavity during the Reload 15, Cycle 16 refueling outage. The reconstituted specimens will be located in Capsule No. 4. The licensee also committed to schedule the withdrawal of Capsule No. 3 based on the test results of Capsule No. 2 in order to meet the intent of ASTM E-185-82.

The licensee proposed to relocate the capsule withdrawal schedule for the surveillance of reactor vessel materials from the Cooper Technical Specifications to the USAR. The staff finds this proposal acceptable because it satisfies Generic Letter 91-01.

The staff concludes that the proposed P/T limits for heatup, cooldown, and criticality are acceptable through 21 EFPY and the proposed limits for hydrotest or leak test are acceptable for 13, 18, and 21 EFPY because they all conform to the requirements of Appendix G of 10 CFR Part 50 and Generic Letter 88-11. Hence, the proposed P/T limits may be incorporated into the Cooper Technical Specifications.

The licensee proposed relocation of the surveillance capsule withdrawal schedule from the Cooper Technical Specifications to the USAR is also acceptable because the relocation satisfies Generic Letter 91-01.

3.0 STATE CONSULTATION

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

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 and changes in surveillance requirements. 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 (57 FR 40214). 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 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.

Principal Contributor: J. Tsao

Date: October 13, 1992