August 21, 1985

Docket No. 50-255
LS05-85-08-025

See Correction letter of 9/10/85

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Mr. David J. VandeWalle
Director, Nuclear Licensing
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: REVISED PRESSURE/TEMPERATURE LIMITS

Re: Palisades Plant

The Commission has issued the enclosed Amendment No. 89 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment is in response to your application dated June 14, 1985.

This amendment changes the Technical Specifications for the Palisades Plant to provide new, more restrictive pressure-temperature limits for heatup, cooldown and hydrostatic test to account for the effects of irradiation of the reactor vessel materials for 6.6 effective full power years of operation.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 3, 1985 (50 FR 27504). No comments or requests for hearing were received.

A copy of our related Safety Evaluation is enclosed. This action will appear in the Commission's biweekly notice publication in the Federal Register.

Sincerely,

Original signed by:

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

SE01
DSU USE 04

Enclosures:
1. Amendment No.89 to License No. DPR-20
2. Safety Evaluation

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PDR ADOCK 05000255
P PDR

cc w/enclosures:
See next page

DL: ORB #5	DL: ORB #5	OELD	DL: ORB #5	DL: ORB
CJamerson	TWambach:jb		JZwolinski	GD DCrutchfield
8/8/85	8/8/85	8/14/85	8/20/85	8/20/85

Mr. David J. VandeWalle
Consumers Power Company

Palisades Plant

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

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 89
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated June 14, 1985 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 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 3.B. of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 21, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 89

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3-5 thru 3-12	3-5 thru 3-12

3.1.2 Heatup and Cooldown Rates (Contd)

(2) (Contd)

surveillance program capsule which was removed at the beginning of the Cycle 3. For purposes of determining fluence at the reactor vessel beltline until a fluence of 1.3×10^{19}nvt is realized at the inner vessel wall at the beltline region, the following basis is established: 5.9×10^{19}nvt calculated at the reactor vessel beltline for 2540 MW_t for 40 years at an 80% load factor. This conversion has resulted in a correlation of 1.989×10^{12}nvt per 1 MWd_t.

(3) The limit lines in Figures 3-1, 3-2 and 3-3 are based on the requirements of Reference 9, Paragraphs IV.A.2 and IV.A.3. These lines reflect a preservice hydrostatic test pressure of 2400 psig and a vessel flange material reference temperature of 60°F.

Basis

All components in the primary coolant system are designed to withstand the effects of cyclic loads due to primary system temperature and pressure changes.(1) These cyclic loads are introduced by normal unit load transients, reactor trips and start-up and shutdown operation. During unit start-up and shutdown, the rates of temperature and pressure changes are limited. A maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.(2)

The reactor vessel plate and material opposite the core has been purchased to a specified Charpy V-Notch test result of 30 ft-lb or greater at an NDTT of + 10°F or less. The vessel weld has the highest RT_{NDT} of plate, weld and HAZ materials at the fluence to which the Figures 3-1, 3-2 and 3-3 apply.(10) The unirradiated RT_{NDT} has been determined to be -56°F.(11) An RT_{NDT} of -56°F is used as an unirradiated value to which irradiation effects are added. In addition, the plate has been 100% volumetrically inspected by ultrasonic test using

3.1.2 Heatup and Cooldown Rates (Contd)

Basis (Contd)

both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other primary coolant system components, meets the appropriate design code requirements and specific component function and has a maximum NDTT of $+40^{\circ}F$.[5]

As a result of fast neutron irradiation in the region of the core, there will be an increase in the RT with operation. The techniques used to predict the integrated fast neutron (E > 1 MeV) fluxes of the reactor vessel are described in Section 3.3.2.6 of the FSAR and also in Amendment 13, Section II, to the FSAR.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift from a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron (E > 1 MeV) exposure of the reactor vessel is computed to be 5.9×10^{19} nvt for 40 years' operation at 2540 MW_t and 80% load factor. The predicted RT_{NDT} shift for the base metal has been predicted based upon surveillance data and the appropriate US NRC Regulatory Guide.[6] To compensate for any increase in the RT caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown.

Reference 7 provides a procedure for obtaining the allowable loadings for ferritic pressure-retaining materials in Class 1 components. This procedure is based on the principles of linear elastic fracture mechanics and involves a stress intensity factor prediction which is a lower bound of static, dynamic and crack arrest critical values. The stress

3.1.2 Heatup and Cooldown Rates (Contd)

Basis (Contd)

intensity factor computed[7] is a function of RT_{NDT}, operating temperature, and vessel wall temperature gradients.

Pressure-temperature limit calculational procedures for the reactor coolant pressure boundary are defined in Reference 8 based upon Reference 7. The limit lines of Figures 3-1 through 3-3 consider a 54 psi pressure allowance to account for the fact that pressure is measured in the pressurizer rather than at the vessel beltline. In addition, for calculational purposes, 5°F and 30 psi were taken as measurement error allowances for temperature and pressure, respectively. By Reference 7, reactor vessel wall locations at 1/4 and 3/4 thickness are limiting. It is at these locations that the crack propagation associated with the hypothetical flaw must be arrested. At these locations, fluence attenuation and thermal gradients have been evaluated. During cooldown, the 1/4 thickness location is always more limiting in that the RT_{NDT} is higher than that at the 3/4 thickness location and thermal gradient stresses are tensile there. During heatup, either the 1/4 thickness or 3/4 thickness location may be limiting depending upon heatup rate.

Figures 3-1 through 3-3 define stress limitations only from a fracture mechanic's point of view.

Other considerations may be more restrictive with respect to pressure-temperature limits. For normal operation, other inherent plant characteristics may limit the heatup and cooldown rates which can be achieved. Pump parameters and pressurizer heating capacity tends to restrict both normal heatup and cooldown rates to less than 60°F per hour.

The revised pressure-temperature limits are applicable to reactor vessel inner wall fluences of up to 1.3×10^{19}nvt. The application of appropriate fluence attenuation factors (Reference 10) at the 1/4 and 3/4 thickness locations results in RT_{NDT} shifts of 223°F and 170°F, respectively, for the limiting weld material. The criticality condition

3.1.2 Heatup and Cooldown Rates (Contd)

Basis (Contd)

which defines a temperature below which the core cannot be made critical (strictly based upon fracture mechanics' considerations) is 352°F. The most limiting wall location is at 1/4 thickness. The minimum criticality temperature, 352°F is the minimum permissible temperature for the inservice system hydrostatic pressure test. That temperature is calculated based upon 2310 psig inservice hydrostatic test pressure.

The restriction of heatup and cooldown rates to 100°F/h and the maintenance of a pressure-temperature relationship under the heatup, cooldown and inservice test curves of Figures 3-1, 3-2 and 3-3, respectively, ensures that the requirements of References 6, 7, 8 and 9 are met. The core operational limit applies only when the reactor is critical.

The criticality temperature is determined per Reference 8 and the core operational curves adhere to the requirements of Reference 9. The inservice test curves incorporate allowances for the thermal gradients associated with the heatup curve used to attain inservice test pressure. These curves differ from heatup curves only with respect to margin for primary membrane stress.[7] For heatup rates less than 60°F/h, the hypothetical 0°F/h (isothermal heatup) at the 1/4 T location is controlling and heatup curves converge. Cooldown curves cross for various cooldown rates, thus a composite curve is drawn. Due to the shifts in RT_{NDT}, NDTT requirements associated with nonreactor vessel materials are, for all practical purposes, no longer limiting.

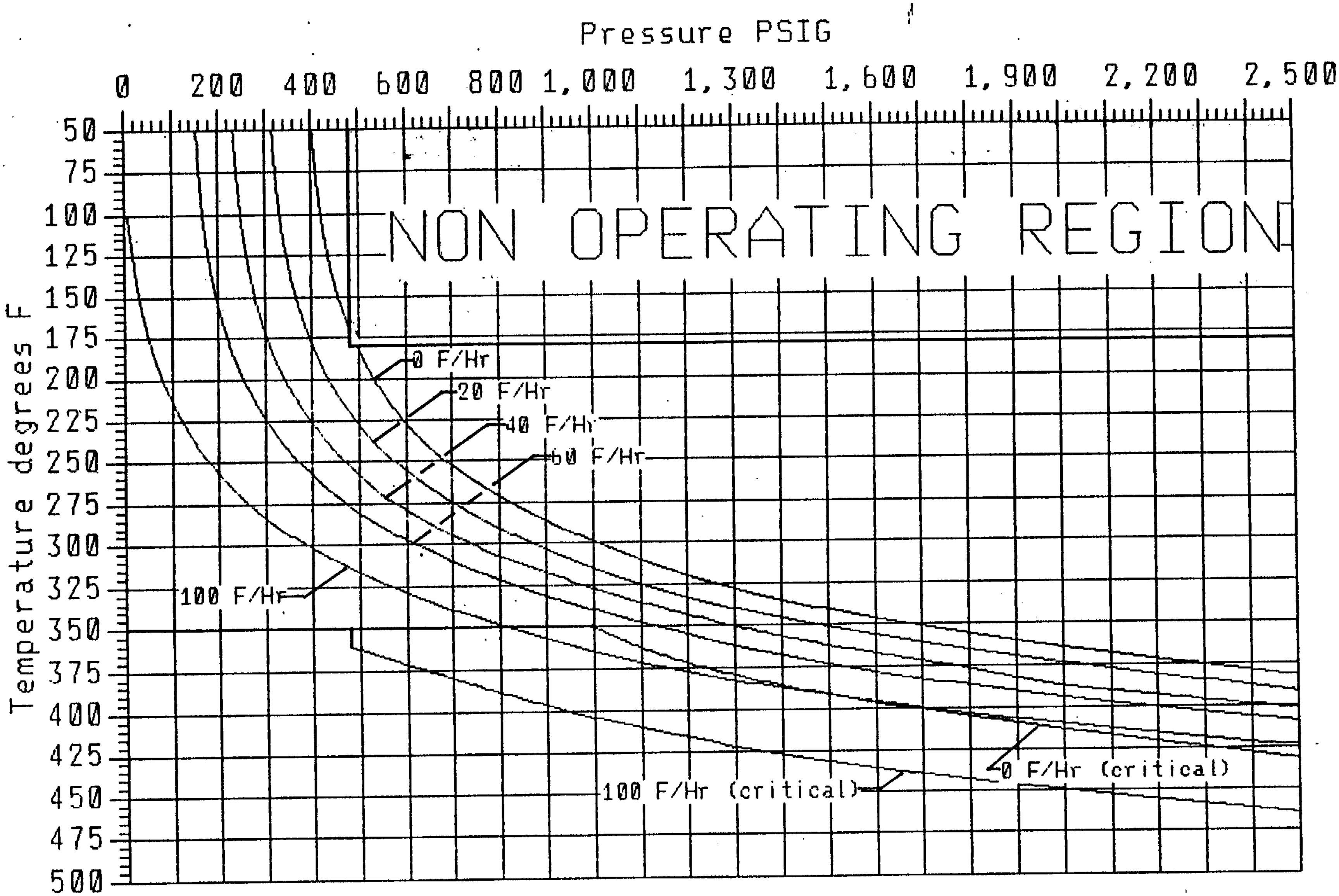
References

(1) FSAR, Section 4.2.2.

(2) ASME Boiler and Pressure Vessel Code, Section III, A-2000.

(3) Battelle Columbus Laboratories Report, "Palisades Pressure Vessel Irradiation Capsule Program: Unirradiated Mechanical Properties," August 25, 1977.

(4) Battelle Columbus Laboratories Report, "Palisades Nuclear Plant Reactor Vessel Surveillance Program: Capsule A-240," March 13, 1979, submitted to the NRC by Consumers Power Company letter dated July 2, 1979.

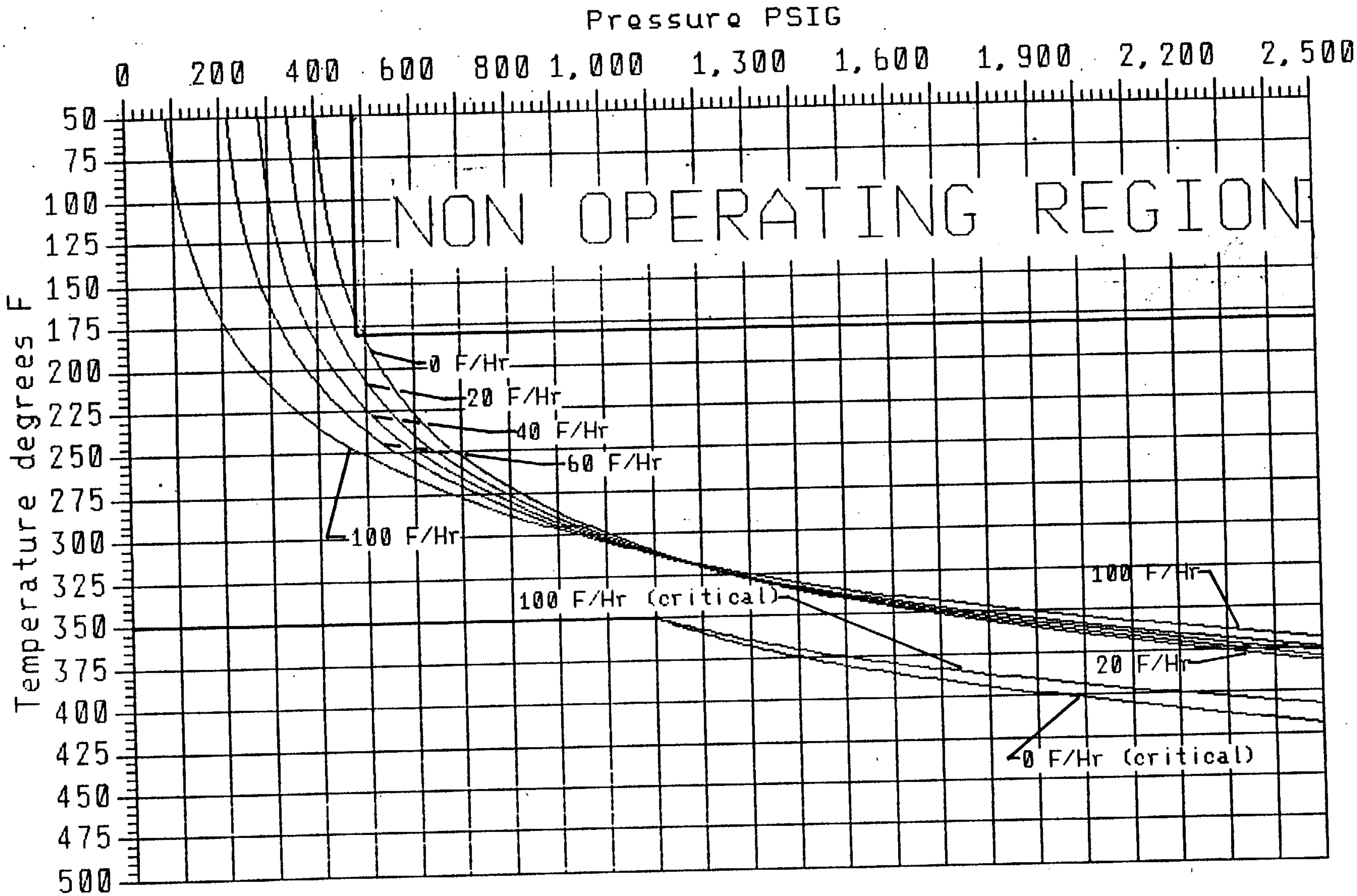


PALISADES PLANT TECH. SPEC.

PRESSURE - TEMPERATURE LIMITS FOR HEATUP - TO 1.3 X 10^{19} nvt

DATE: AMEND. NO.

FIGURE 3-1



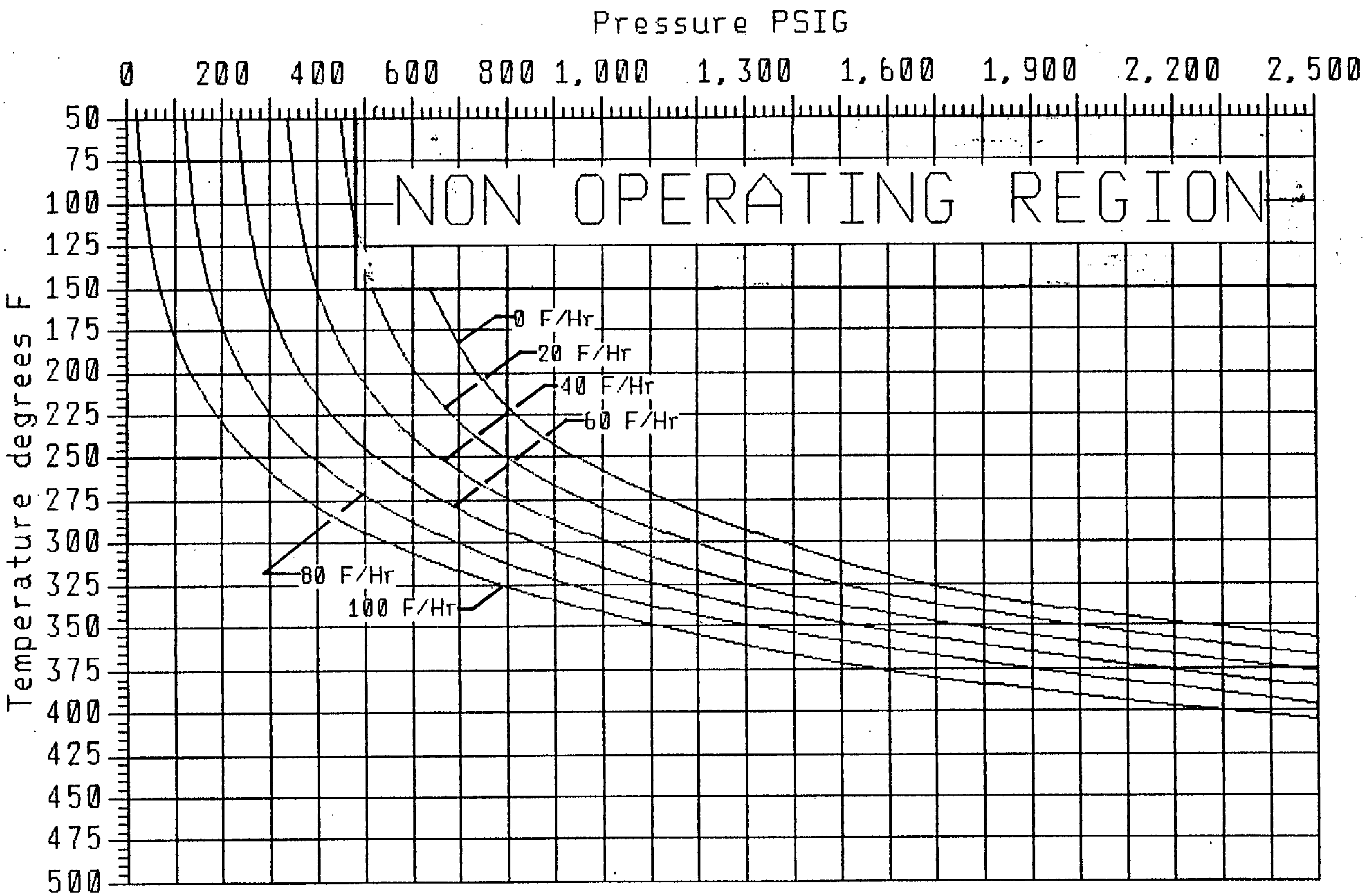
PALISADES PLANT
TECH. SPEC.

PRESSURE - TEMPERATURE LIMITS
FOR COOLDOWN - TO 1.3 X 10^{19} nvt

DATE:
AMEND. NO.

FIGURE 3-2

Amendment No. 27, 41, 58, 89



PALISADES PLANT TECH. SPEC.

PRESSURE - TEMPERATURE LIMITS FOR HYDRO TEST - TO 1.3 X 10^{19} nvt

DATE:
AMEND. NO.

FIGURE 3-3

3.1.2 Heatup and Cooldown Rates (Contd)

References (Contd)

(5) FSAR, Section 4.2.4.

(6) US Nuclear Regulatory Commission, Regulator Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," July, 1975.

(7) ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure," 1974 Edition.

(8) US Atomic Energy Commission Standard Review Plan, Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits."

(9) 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," May 31, 1983.

(10) US Nuclear Regulatory Commission, Regulatory Guide 1.99 Draft Revision 2, April, 1984.

(11) Combustion Engineering Report CEN-189, December, 1981.

3.1.3 Minimum Conditions for Criticality

a) Except during low-power physics test, the reactor shall not be made critical if the primary coolant temperature is below 525°F.

b) In no case shall the reactor be made critical if the primary coolant temperature is below 352°F.

c) When the primary coolant temperature is below the minimum temperature specified in "a" above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

d) No more than one control rod at a time shall be exercised or withdrawn until after a steam bubble and normal water level are established in the pressurizer.

e) Primary coolant boron concentration shall not be reduced until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all control rods withdrawn.[1] However, the uncertainty of the calculation is such that it is possible that a slightly positive coefficient could exist.

Amendment No. ~~27~~, ~~41~~, ~~55~~, 89

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 89 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT (PNP)

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated June 14, 1985 , the Consumers Power Company (CPC) submitted a request for changes to the Palisades Plant Technical Specification Sections 3.1.2 and 3.1.3.

The amendment provides new reactor vessel pressure-temperature limits for heat-up, cooldown and hydrostatic test. The last surveillance capsule report submitted to the staff by the licensee was Westinghouse WCAP 10637, entitled "Analysis of Capsule T-330 and W-290 from the Consumers Power Company Palisades Reactor Vessel Radiation Surveillance Program." This report was submitted to the NRC by letter dated October 31, 1984.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 3, 1985 (50 FR 27504). No comments or requests for hearing were received.

2.0 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 increase in RT_{NDT} resulting from neutron irradiation damage to the beltline materials.

The PNP reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the RT_{NDT} for each of the reactor vessel materials. Hence, the initial RT_{NDT} for materials in the closure flange and beltline region of the PNP reactor vessel could not be determined in accordance with the test requirements of the ASME Code. Therefore, the initial RT_{NDT} for these materials must be estimated from test data from other similar materials used for fabrication of reactor vessels in the nuclear industry.

The licensee indicates that the limiting closure flange region materials were forgings, which were fabricated to ASME Code SA 508 C12 requirements. The licensee has estimated the RT_{NDT} for these materials in accordance with Branch Technical Position - MTEB 5-2, "Fracture Toughness Requirements," which are contained in NUREG-0800, "USNRC Standard Review Plan 5.3.2, Pressure-temperature Limits". This branch technical position provides conservative estimates of RT_{NDT} for reactor vessel materials. This branch technical position results in an RT_{NDT} for the closure flange forgings of 60°F.

The limiting materials in the PNP reactor vessel beltline are weld metals, which were fabricated by Combustion Engineering using the submerged arc weld process with RACO 3 and MIL B-4 Mod (Mn Mo Ni) weld wires. The RACO 3 submerged arc welds were fabricated using a second wire of pure nickel, identified as Ni 200. In all submerged arc welds the flux utilized was Linde 1092. The initial RT_{NDT} for these weld materials was estimated by the licensee as -56°F with a standard deviation of 17°F. These initial RT_{NDT} and standard deviation values were recommended by the staff in Commission Report SECY 84-465, "Pressurized Thermal Shock" for welds fabricated by Combustion Engineering using Linde 1092 flux.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the method documented in Draft Regulatory Guide 1.99, Revision 2, "Radiation Damage to Reactor Vessel Materials." Although this regulatory guide is only a draft, its methodology is considered by the staff to be the most up-to-date method for predicting neutron irradiation damage. This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of copper and nickel in the beltline material.

The licensee has conducted a detailed search of vessel and surveillance fabrication records at Combustion Engineering to determine the heats of wire used in their reactor vessel beltline and their surveillance welds. As a result of this search, the licensee indicates that the PNP surveillance weld was fabricated using heats of wire, which were different from those used in fabrication of the PNP beltline welds.

The search confirmed that RACO 3 heat numbers W5214 and 34B009 and MIL B-4 Mod (Mn Mo Ni) heat number 27204 were utilized to fabricate the PNP reactor vessel beltline. During fabrication of the PNP reactor vessel, chemical analyses of the PNP beltline welds were not performed. However, the licensee in Attachment III to its June 14, 1985 memorandum has established the amounts of copper and nickel in each of the beltline welds. The amounts of copper and nickel were estimated from chemical analyses of reactor vessel surveillance welds and other nuclear vessel welds, which were fabricated by Combustion Engineering using the same heats of weld wire as the PNP beltline material. Since the amount of copper and nickel should be consistent within a heat of weld wire and the wire is the source of copper and nickel in a weld, the use of chemical analyses from surveillance welds and other nuclear vessel welds fabricated with the same heats of wire as the PNP beltline weld should provide reliable estimates for the amounts of copper and nickel in the PNP beltline welds.

The licensee's proposed pressure-temperature limits have been calculated using a neutron fluence of 1.30 x 10^{19} n/cm^2 (E > 1MeV). The amount of time corresponding to this neutron fluence incident on the reactor vessel is dependent upon a radiological evaluation of the core and the PNP vessel. Report WCAP-10637 contains a description of the radiological analyses performed by Westinghouse on the PNP core vessel. This analysis results in a lead factor of 1.28 between the capsule and the vessel location receiving the highest neutron flux. The Westinghouse radiological calculation predicts the end of life (2530 MWt for 32 effective full-power years) peak neutron fluence to be 6.56 x 10^{19} n/cm^2 (E > 1Mev), when the axial peaking factor at the core midplane is 1.20. The licensee has evaluated its previous core peaking factors and determined that the axial peaking factor for the midplane of the core was 1.15. This decrease in the axial peaking factor causes the end of life peak neutron fluence to be reduced to 6.29 x 10^{19} n/cm^2 (E > 1MeV).

Report WCAP-10637 contains the Westinghouse analysis of the dosimetry in Surveillance Capsule W-290. The calculated peak neutron fluence at the end of life using the results from the Capsule W-290 dosimetry and the predicted lead factor of 1.28 is 5.38 x 10^{19} n/cm^2 (E > 1MeV). Since the peak neutron fluence from the Capsule W-290 dosimetry is less than that calculated using the Westinghouse radiological analysis with 1.15 axial peaking factor, the Westinghouse calculated value will conservatively estimate the end of life neutron fluence for the PNP reactor vessel.

3.0 EVALUATION

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 beltline materials was estimated using the method documented in Draft Regulatory Guide 1.99, Working Paper F dated May 21, 1985. The inputs used were the amounts of copper and nickel reported in Attachment III to the licensee's letter dated June 14, 1985 and the calculated end of life peak neutron fluence of 6.29 x 10^{19} n/cm (E > 1MeV). The pressure temperature curves submitted by the licensee have been accepted by the staff; however, the licensee's submittal would permit use of these curves without reassessment for a period of 9 effective full power years. Upon review by the staff, a mathematical error was found which, when corrected, would require that these curves should be used only for 6.6 effective full power years before they are reassessed. The licensee has accepted the staff assessment. This change does not change the substance of the amendment requested by the licensee and there are no other differences between the amendment requested by the licensee and the amendment authorized by the staff. Our conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR Part 50 for 6.6 effective full power years and may be incorporated into the plant's technical specifications.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of facility components 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 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.

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

6.0 ACKNOWLEDGEMENT

This Safety Evaluation has been prepared by B. J. Elliot.

Dated: August 21, 1985.