DCR 016

August 21, 1986

Docket No. 50-255

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Mr. Kenneth W. Berry
Director, Nuclear Licensing
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 97 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated March 17, 1986.

This amendment changes the Technical Specifications by revising pressure-temperature limits for heatup, cooldown and hydrostatic tests of the reactor vessel to account for radiation effects. This amendment also extends the limits from 6.6 effective full power years to approximately 9.0 EFPY.

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

Sincerely,

/S/

Thomas V. Wambach, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 97 to License No. DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

PBD#8	PWR#8 JVW	PBD#8 JVW	PBD#8 AT	OGC-Bethesda	FOB
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Mr. Kenneth W. Berry
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. 97
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 March 17, 1986, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 97 , 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

A C Thadani

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: August 21, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 97

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
3-4	3-4
3-5	3-5
3-6	3-6
3-7	3-7
3-8	3-8
3-9	3-9
3-10	3-10
3-11	3-11
3-12	3-12
3-13	3-13

3.1 PRIMARY COOLANT SYSTEM (Cont'd)

3.1.2 Heatup and Cooldown Rates

The primary coolant pressure and the system heatup and cooldown rates shall be limited in accordance with Figure 3-1, Figure 3-2 and as follows:

a) Allowable combinations of pressure and temperature for any heatup rate shall be below and to the right of the limit lines as shown on Figure 3-1. The average heatup rate shall not exceed 100°F/h in any one-hour time period.

b) Allowable combinations of pressure and temperature for any cooldown rate shall be below and to the right of the limit lines as shown on Figure 3-2. The average cooldown rate shall not exceed 100°F/h in any one-hour time period.

c) Allowable combinations of pressure and temperature for inservice testing from heatup are as shown in Figure 3-3. Those curves include allowances for the temperature change rates noted above. Interpolation between limit lines for other than the noted temperature change rates is permitted in 3.1.2a, b or c.

d) The average heatup and cooldown rates for the pressurizer shall not exceed 200°F/h in any one-hour time period.

e) Before the radiation exposure of the reactor vessel exceeds the exposure for which the figures apply, Figures 3-1, 3-2 and 3-3 shall be updated in accordance with the following criteria and procedure:

(1) US Nuclear Regulatory Commission Regulatory Guide 1.99 has been used to predict the increase in transition temperature based on integrated fast neutron flux and surveillance test data. If measurements on the irradiated specimens show increase above this curve, a new curve shall be constructed such that it is above and to the left of all applicable data points.

(2) Before the end of the integrated power period for which Figures 3-1, 3-2 and 3-3 apply, the limit lines on the figures shall be updated for a new integrated power period. The total integrated reactor thermal power from start-up to the end of the new power period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV). Such a conversion shall be made consistent with the dosimetry evaluation of capsule W-290[12].

3.1.2 Heatup and Cooldown Rates (Cont'd)

(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[8].

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. 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 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°F.[5]

As a result of fast neutron irradiation in this 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.

3.1.2 Heatup and Cooldown Rates (Cont'd)

Basis (Cont'd)

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 flux variation. The predicted RT_{NDT} shift for the base metal has been predicted based upon surveillance data and the US NRC Regulatory Guide.[10] 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 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

Basis (Cont'd)

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 mechanics 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 1.8×10^{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 241°F and 183°F, respectively, for the limiting weld material. The criticality condition which defines a temperature below which the core cannot be made critical (strictly based upon fracture mechanics' considerations) is 371°F. The most limiting wall location is at 1/4 thickness. The minimum criticality temperature, 371°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.

3.1.2 Heatup and Cooldown Rates (Cont'd)

Basis (Cont'd)

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] 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.

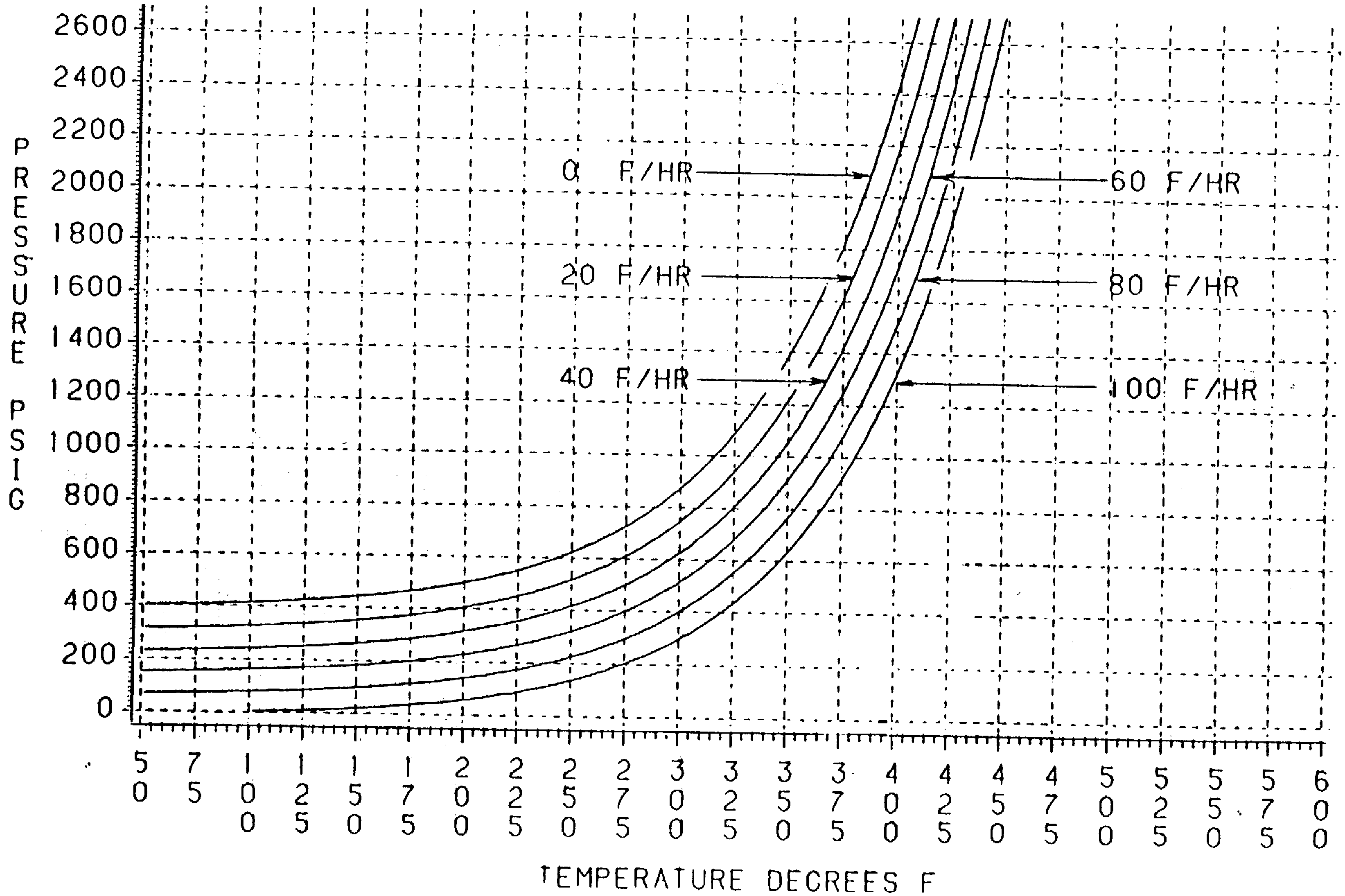
(5) FSAR, Section 4.2.4.

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

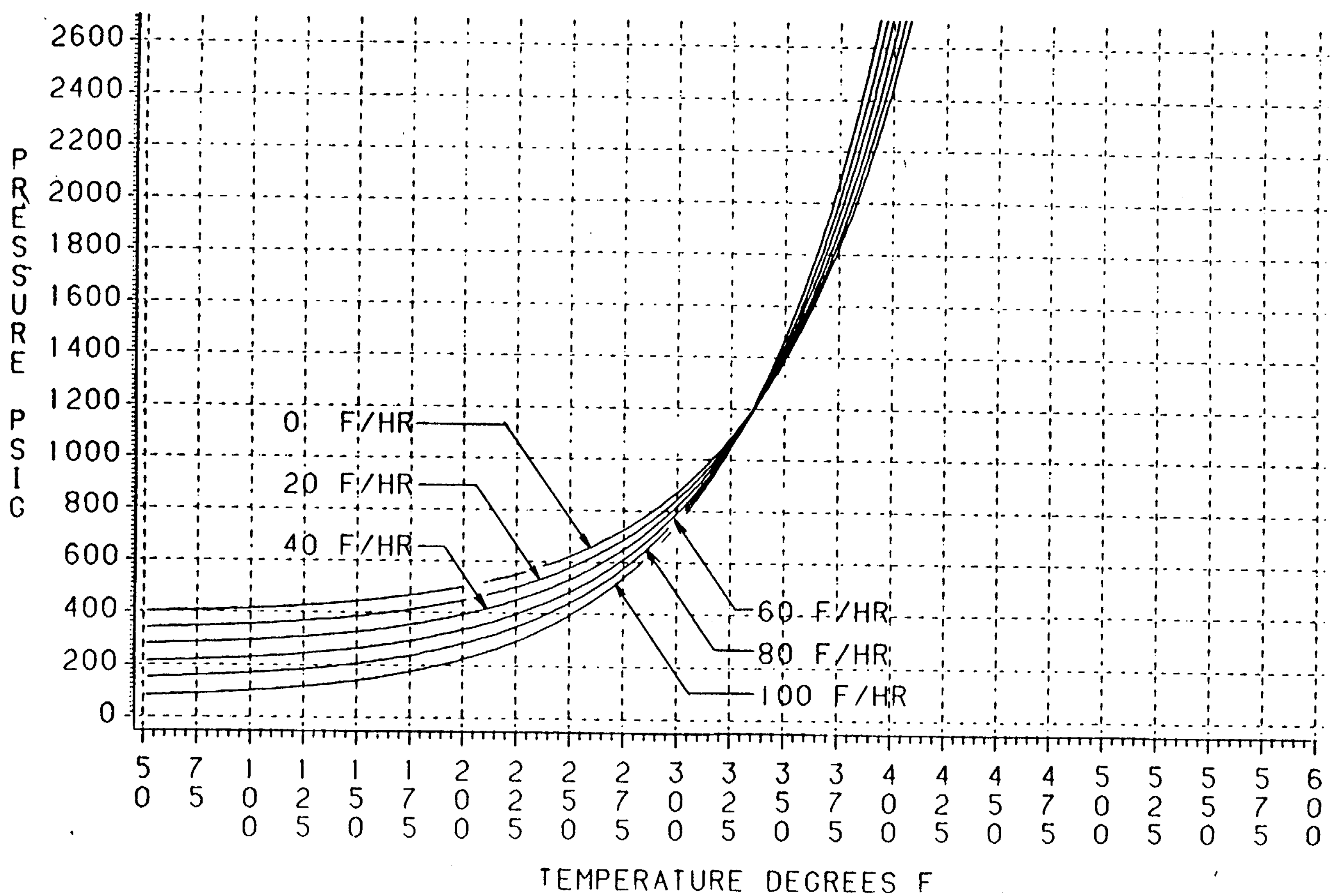
(7) ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure," 1974 Edition.

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

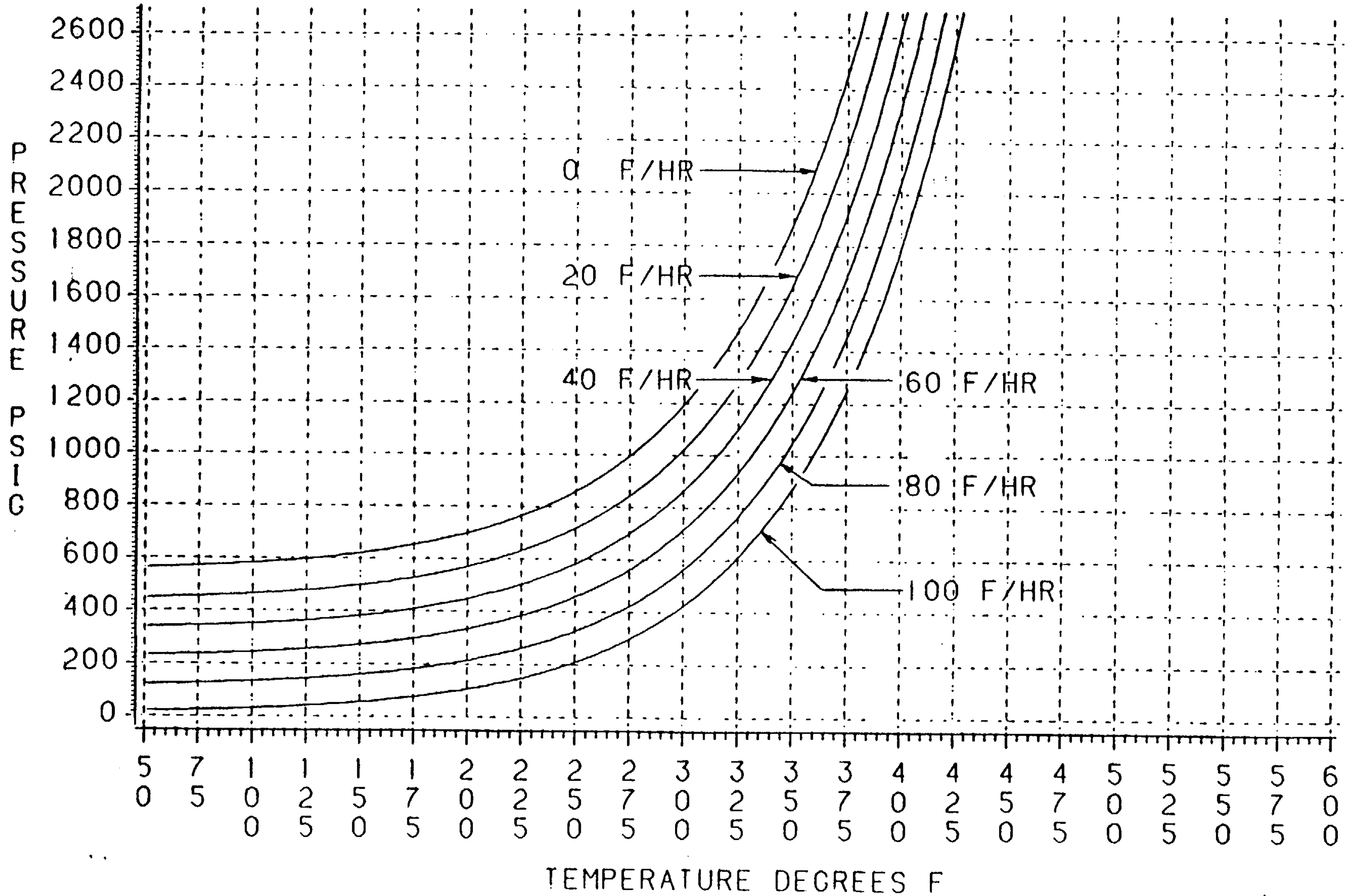
(9) 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," May 31, 1983.



Palisades Plant Pressure/Temperature Limits for Heatup
for Fluence to 1.8×10^{19} nvt
Figure 3-1



Palisades Plant Pressure/Temperature Limits for Cooldown
for Fluence to 1.8×10^{19} nvt
Figure 3-2



Palisades Plant Pressure/Temperature Limits for Hydro for Fluence to 1.8×10^{19} nvt

Figure 3-3

3.1.2 Heatup and Cooldown Rates (Cont'd)

References (Cont'd)

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

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

(12) "Analysis of Capsules T-330 and W-290 from the Consumers Power Company Palisades Reactor Vessel Radiation Surveillance Program," WCAP-10637, September 1984.

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 371°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.

The moderator coefficient at lower temperatures will be less negative or more positive than at operating temperature.[1,2] It is, therefore, 3.1.3 Minimum Conditions for Criticality (Cont'd)

Basis (Cont'd)

prudent to restrict the operation of the reactor when primary coolant temperatures are less than normal operating temperature ($\geq$ 525°F). Assuming the most pessimistic rods out moderator coefficient, the maximum potential reactivity insertion that could result from depressurizing the coolant from 2100 psia to saturation pressure at 525°F is 0.1%$\Delta\rho$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(3) and the small integrated $\Delta\rho$ would limit the magnitude of a power excursion resulting from a reduction of moderator density. The requirement that the reactor is not to be made critical below 371°F provides increased assurance that the proper relationship between primary coolant pressure and temperature will be maintained relative to the RT_{NDT} of the primary coolant system pressure boundary material. Heatup to this temperature will be accomplished by operating the primary coolant pumps.

If the shutdown margin required by Specification 3.10.1 is maintained, there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.

Normal water level is established in the pressurizer prior to the withdrawal of control rods or the dilution of boron so as to preclude the possible overpressurization of a solid primary coolant system.

References

(1) FSAR, Table 3-2
(2) FSAR, Table 3-6
(3) FSAR, Table 3-3

(Next page is 3-17)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 97 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated March 17, 1986, the Consumers Power Company (the licensee) requested a change to the Palisades Technical Specifications, Sections 3.1.2 and 3.1.3. The proposed revisions provide new reactor vessel pressure-temperature limits for heat-up, cool-down and the inservice hydrostatic test.

The last surveillance capsule report submitted to the staff by the licensee was Westinghouse Report 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 in a letter from B. D. Johnson to H. R. Denton dated October 31, 1984.

2.0 EVALUATION

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

The Palisades reactor vessel was procured to ASME Code requirements that were in effect at the time of procurement. However, Code requirements at that time did not specify fracture toughness testing to determine the RT_{NDT} for each of the materials needed to make the reactor vessel. Hence, the initial RT_{NDT} for materials in the closure flange and beltline region of the reactor vessel could not be determined in accordance with the test requirements of the current ASME Code. Therefore, the initial RT_{NDT} for these materials is 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

* Formulae contained within this text are defined in the enclosed table.

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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 MTEB5-2, "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 and uses a RT_{NDT} of 60°F for the closure flange forgings.

The limiting materials in the 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 Modified (Mn Mo Ni) weld wires. However, records clearly indicate that the RACO 3 weld wire had the addition of Nickel-200 and not the Mil B-4 Modified. 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 82-465, "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 surveillance weld was fabricated using heats of wire, which were different from those used in the fabrication of the 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 reactor vessel beltline. During fabrication of the reactor vessel, chemical analyses of the beltline welds were not performed. However, the licensee in Attachment III to their June 14, 1985 letter to the NRC has established the amounts of cooper 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 Palisades beltline material. Since the amount of copper and nickel should be the same within a heat of weld wire and the weld 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 Palisades beltline weld should provide reliable estimates for the amounts of copper and nickel in the Palisades beltline welds.

The licensee's proposed pressure-temperature limits at 9 Effective Full Power Years (EFPY) have been calculated using a neutron fluence of 1.80×10^{19} n/cm^2 (E>1MeV). The amount of time required to accumulate this neutron fluence incident at the inner diameter of the reactor pressure vessel is dependent upon a radiological evaluation of the core and the reactor vessel. Report WCAP-10637 contains a description of the radiological analyses performed by Westinghouse on the Palisades core and vessel. These analyses result 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 6.56×19^{19} n/cm^2 (E>1MeV), when the axial peaking factor at the core midplane is 1.20. The licensee has evaluated the previous core peaking factors at Palisades and found them to be 1.15. An axial peaking factor of 1.15 yields an end of life peak neutron fluence of 6.29×10^{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 5.48×10^{19} n/cm^2 (E>1MeV). Since this peak neutron fluence from the Capsule W-290 dosimetry is less than the 6.29×10^{19} n/cm^2 (E>1MeV) calculated using the Westinghouse radiological analysis, the value of 6.29×10^{19} n/cm^2 (E>1MeV) will conservatively estimate the end of life neutron fluence for the Palisades reactor pressure vessel.

The longitudinal weld is taken as the most limiting material. The amount of copper and nickel in the longitudinal weld is 0.19% by weight and 1.1% by weight, respectively. The inside surface fluence is taken as 1.8×10^{19} n/cm^2 (E>1MeV). The reactor vessel inside radius is 86 inches, and the outside radius is 94.5 inches which yields reactor vessel wall thickness and weld thickness of 8.5 inches.

Flaws are postulated on the inside surface and the outside surface of the vessel or weld. Distance is measured from the inner radius of the vessel outward. The flaws on the inside surface and outside surface are referred to by location as 1/4 thickness and 3/4 thickness, respectively.

Recently, Draft Regulatory Guide 1.99, Revision 2, was revised to incorporate comments from the public. No substantive changes in the Regulatory Guide occurred. However, the formula for attenuation of neutron fluence through the vessel wall was revised. Based on Regulatory Guide 1.99, Revision 2, the shift in RT_{NDT} at 1/4 thickness and 3/4 thickness for the longitudinal weld computed by the staff and the licensee differ by a nominal 2% or less. This difference in ΔRT_{NDT} is due to the method used to attenuate neutron fluence through the vessel wall. The staff finds the adjusted RT_{NDT} computed by the licensee acceptable.

The criteria from Section III, ASME Code, Article G-2000, Vessels, was used to determine the measured temperature during an inservice hydrostatic test ($K_{IR} \geq 1.5K_{Im} + K_{It}$). At the flaw on the inside surface of the longitudinal weld, the temperatures computed by the staff and the licensee are essentially the same for the hydrostatic test pressure of 2310 psig. The minimum criticality temperature of 371°F computed by the licensee is acceptable. The family of curves for pressure-temperature limits for inservice hydrostatic testing is acceptable to the staff.

The criteria from Section III, ASME Code, Article G-2000, Vessels, was also used to determine the measured temperature during various heat-up and cool-down rates ($K_{IR} \geq 2K_{Im} + K_{It}$) of the reactor vessel. For heat-up, either the flaw in the longitudinal weld on the inside surface (1/4 thickness) or the outside surface (3/4 thickness) is limiting. For cool-down, the flaw in the longitudinal weld on the inside surface (1/4 thickness) is limiting. The family of heat-up and cool-down curves (0°F/hr through 100°F/hr) computed by the licensee either conservatively bound the values calculated by the staff or differ by 2% or less. The differences in computed temperature values between the licensee and the staff are due to (1) the expressions used to attenuate the neutron fluence, (2) the methodology used to determine the temperature gradient through the vessel wall, and (3) whether K_{It} is disregarded when it would be conservative to do so. Thus, the staff finds the pressure-temperature limits for the family of heat-up and cool-down curves presented by the licensee to be acceptable.

Figures 3-1 and 3-2 of the licensee's submittal do not include pressure-temperature limits for heat-up and cool-down during core operations; that is, these figures are for the reactor noncritical during heat-up and cool-down. Pressure-temperature limits for heat-up and cool-down during core operations are obtained by adding 40°F to the temperature values in Figures 3-1 and 3-2. The resulting temperature must be greater than or equal to the minimum criticality temperature, 371°F.

In the submittal of March 17, 1986 the licensee requested changes to the Technical Specifications, Sections 3.1.2 and 3.1.3. The staff has reviewed each of the proposed changes and finds them all acceptable.

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan Section 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 1.99, Revision 2. Amounts of copper and nickel reported in Attachment III to the licensee's letter dated June 14, 1985 and an end of life peak neutron fluence of 6.29×10^{19} n/cm^2 (E>1MeV) were used. Our conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR Part 50, for 9 (EFPY) based on a fluence of 1.8×10^{19} n/cm^2 (E>1MeV) and may be incorporated into the plant's technical specifications.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: August 21, 1986

Principal Contributor:
Robert J. Wright

Enclosure

FORMULAE DEFINITIONS

(1) NDT = Nil-Ductility transition

(2) Rt_{NDT} = Reference Temperature (^{o}F) as defined in the ASME code.

(3) ΔRT_{NDT} = Reference Temperature Shift (^{o}F)

(4) K_{Im} = Membrane Hoop Stress Intensity Factor KSI $\sqrt{in}$

(5) K_{IR} = Reference Stress Intensity Factor KSI $\sqrt{in}$

(6) K_{It} = Thermal Stress Intensity Factor KSI $\sqrt{in}$