April 26, 1990

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:

SUBJECT: AMENDMENT NO. 131 TO PROVISIONAL OPERATING LICENSE NO. DPR-20: PRESSURE-TEMPERATURE LIMITS AND LOW-TEMPERATURE OVERPRESSURE PROTECTION (TAC NOS. 72889 AND 71526)

The Commission has issued Amendment No. 131 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Appendix A Technical Specifications (TS) in response to your application dated September 12, 1989, as supplemented by letters dated September 22, and 25, 1989, and March 2, 1990.

This amendment revises the Appendix A TS relating to Primary Coolant System (PCS) operable components, PCS heatup and cooldown rates, PCS pressure-temperature limits, PCS overpressure protection system set points and operating requirements, and Emergency Core Cooling System (ECCS) operability requirements. The amendment also revises certain related surveillance requirements. Specifically, the amendment modifies TS Sections 3.1, 3.3, 4.1, and 4.6, and Figures 3-1, 3-2, 3-3, and 3-4.

These changes permit the use of a variable set point control-system for low-temperature overpressure protection, account for the use of Regulatory Guide 1.99, Rev. 2, for the determination of PCS heatup and cooldown limits, and allow extending the range of ECCS operation.

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

Sincerely,

original signed by

Albert W. De Agazio, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 131 to License No. DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRSP	PM/PD31:DRSP	(A)D/PD31:DRSP	OGC
MRShuttleworth	ADe Agazio	JThoma	R Bachmann
3/30/90 except TS	4/2/90	4/26/90	4/5/90 w/ insert to SE, p. 3

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

April 26, 1990

Docket No. 50-255

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

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This amendment revises the Appendix A TS relating to Primary Coolant System (PCS) operable components, PCS heatup and cooldown rates, PCS pressure-temperature limits, PCS overpressure protection system set points and operating requirements, and Emergency Core Cooling System (ECCS) operability requirements. The amendment also revises certain related surveillance requirements. Specifically, the amendment modifies TS Sections 3.1, 3.3, 4.1, and 4.6, and Figures 3-1, 3-2, 3-3, and 3-4.

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Sincerely,

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate III-1 Division of
Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
1. Amendment No. 131 to License No. DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

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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.131
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 September 12, 1989, as supplemented by letters dated September 22 and 25, 1989, and March 2, 1990, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131 , 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 and shall be implemented not later than June 1, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 26, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 131

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

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

REMOVE	INSERT
3-1d	3-1d
3-2	3-2
3-3	3-3
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
3-25a	3-25a
3-25b	3-25b
3-25c	3-25c
3-30	3.30
3-33	3-33
4-2	4-2
4-39	4-39
4-41	4-41

3.1 PRIMARY COOLANT SYSTEM (Cont'd)

3.1.1 Operable Components (Cont'd)

h. Forced circulation (starting the first primary coolant pump) shall not be initiated unless one of the following conditions is met:

(1) Primary coolant cold leg temperature is > 430°F.
(2) PCS cold leg temperature is ≤ 430°F and S/G secondary temperature is less than PCS cold leg temperature.
(3) Shutdown cooling is isolated from the PCS AND PCS cold leg temperature is > 210°F AND S/G secondary temperture is less than 100°F higher than PCS temperature.
(4) Shutdown cooling is isolated from the PCS AND PCS cold leg temperature is ≥ 170°F and ≤ 210°F AND S/G secondary temperature is less than 20°F higher than PCS cold leg temperature.
(5) Shutdown cooling is isolated from the PCS AND PCS cold leg temperature is ≥ 120°F and < 170°F AND S/G secondary temperature is less than 100°F higher than PCS cold leg temperature.

i. The PCS shall not be heated or maintained above 325°F unless a minimum of 375 kW of pressurizer heater capacity is available from both buses 1D and 1E. Should heater capacity from either bus 1D and 1E fall below 375 kW, either restore the inoperable heaters to provide at least 375 kW of heater capacity from both buses 1D and 1E within 72 hours or be in hot shutdown within the next 12 hours.

Basis

When primary coolant boron concentration is being changed, the process must be uniform throughout the primary coolant system volume to prevent stratification of primary coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one shutdown cooling or one primary coolant pump is in operation.(1) The shutdown cooling pump will circulate the primary system volume in less than 60 minutes when operated at rated capacity. By imposing a minimum shutdown cooling pump flow rate of 2810 gpm, sufficient time is provided for the operator to terminate the boron dilution under asymmetric flow conditions.(5) The pressurizer volume is relatively inactive, therefore will tend to have a boron concentration higher than rest of the primary coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the primary system during the addition of boron.(2)

Basis (Contd)

The FSAR safety analysis was performed assuming four primary coolant pumps were operating for accidents that occur during reactor operation. Therefore, reactor startup above hot shutdown is not permitted unless all four primary coolant pumps are operating. Operation with three primary coolant pumps is permitted for a limited time to allow the restart of a stopped pump or for reactor internals vibration monitoring and testing.

Requiring the plant to be in hot shutdown with the reactor tripped from the C-06 panel, opening the 42-01 and 42-02 circuit breakers, assures an inadvertent rod bank withdrawal will not be initiated by the control room operator. Both steam generators are required to be operable whenever the temperature of the primary coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

Calculations have been performed to demonstrate that a pressure differential of 1380 psi[3] can be withstood by a tube uniformily thinned to 36% of its original nominal wall thickness (64% degradation), while maintaining:

(1) A factor of safety of three between the actual pressure differential and the pressure differential required to cause bursting.

(2) Stresses within the yield stress for Inconel 600 at operating temperature.

(3) Acceptable stresses during accident conditions.

Secondary side hydrostatic and leak testing requirements are consistent with ASME BPV Section XI (1971). The differential maintains stresses in the steam generator tube walls within code allowable stresses.

The minimum temperature of 100°F for pressurizing the steam generator secondary side is set by the NDTT of the manway cover of + 40°F. /

The transient analyses were performed assuming a vessel flow at hot zero power (532°F) of 124.3×10^6 lb/hr minus 6% to account for flow measurement uncertainty and core flow bypass. A DNB analysis was performed in a parametric fashion to determine the core inlet temperature as a function of pressure and flow for which the minimum DNBR is equal to 1.17. This analysis includes the following uncertainties and allowances: 2% of rated power for power

3.1 PRIMARY COOLANT SYSTEM (Cont'd)

Basis (Cont'd)

measurement; ±0.06 for ASI measurement; ±50 psi for pressurizer pressure; ±7°F for inlet temperature; and 3% measurement and 3% bypass for core flow. In addition, transient biases were included in the derivation of the following equation for limiting reactor inlet temperature:[4]

$$T_{inlet} \leq 543.3 + .0575(P-2060) + 0.00005(P-2060)^{**}2 + 1.173(W-120) - .0102(W-120)^{**}2$$

The limits of validity of this equation are:

$1800 \leq$ Pressure ≤ 2200 Psia
$100.0 \times 10^6 \leq$ Vessel Flow $\leq 130 \times 10^6$ Lb/h
ASI as shown in Figure 3.0

With measured primary coolant system flow rates > 130 M lbm/hr, limiting the maximum allowed inlet temperature to the T_{Inlet} LCO at 130 M lbm/hr increases the margin to DNB for higher PCS flow rates.

The Axial Shape Index alarm channel is being used to monitor the ASI to ensure that the assumed axial power profiles used in the development of the inlet temperature LCO bound measured axial power profiles. The signal representing core power (Q) is the auctioneered higher of the neutron flux power and the Delta-T power. The measured ASI calculated from the excore detector signals and adjusted for shape annealing (Y_I) and the core power constitute an ordered pair (Q,Y_I). An alarm signal is activated before the ordered pair exceed the boundaries specified in Figure 3.0.

The requirement that the steam generator temperature be $\leq$ the PCS temperature when forced circulation is initiated in the PCS ensures that an energy addition caused by heat transferred from the secondary system to the PCS will not occur. This requirement applies only to the initiation of forced circulation (the start of the first primary coolant pump) when the PCS cold leg temperature is < 430°F. However, analysis (Reference 6) shows that under limited conditions when the Shutdown Cooling System is isolated from the PCS, forced circulation may be initiated when the steam generator temperature is higher than the PCS cold leg temperature. /

References

(1) Updated FSAR, Section 14.3.2.
(2) Updated FSAR, Section 4.3.7.
(3) Palisades 1983/1984 Steam Generator Evaluation and Repair Program Report, Section 4, April 19, 1984
(4) ANF-87-150(NP), Volume 2, Section 15.0.7.1
(5) ANF-88-108 /
(6) Consumers Power Company Engineering Analysis EA-A-NL-89-14-1 //

3.1 PRIMARY COOLANT SYSTEM (Continued)

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 or cooldown rate shall be below and to the right of the applicable limit line as shown on Figures 3-1 and 3-2. The average heatup or cooldown rate in any one hour time period shall not exceed the heatup or cooldown rate limit when one or more PCS cold leg is less than the corresponding "Cold Leg Temperature" below.

	*Cold Leg Temperature	Heatup/Cooldown Rate Limit
1	≤ 170°F	20°F/Hr
2.	> 170°F and ≤ 250°F	40°F/Hr
3.	> 250°F and < 350°F	60°F/Hr
4.	≥ 350°F	100°F/Hr

Whenever the shutdown cooling isolation valves (MOV3015 and MOV3016) are open, the primary coolant system shall not be heated at a rate of more than 40°F/Hr. when the "Cold Leg Temperature" is >170°F.

b. Allowable combinations of pressure and temperature for inservice testing during heatup are as shown in Figure 3-3. The maximum heatup and cooldown rates required by Section a. above, are applicable. Interpolation between limit lines for other than the noted temperature change rates is permitted in 3.1.2a.

c. The average heatup or cooldown rates for the pressurizer shall not exceed 200°F/hr in any one hour time period. Whenever the Shutdown Cooling isolation valves (MOV3015 and MOV3016) are OPEN, the pressurizer shall not be heated at a rate of more than 60°F/Hr.

*Use shutdown cooling return temperature if the shutdown cooling system is in operation and all PCP's are off.

3.1.2 Heatup and Cooldown Rates (Continued)

d. 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 Revision 2 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. 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 limit 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,

3.1.2 Heatup and Cooldown Rates (Continued)

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 integrated fast neutron (E > 1 MeV) fluxes of the reactor vessel are calculated using Reference 13, utilitzing DOT III Code with the SAILOR set of cross-sections.

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 and to account for PCP discharge pressure. In addition, for calculational purposes, 5°F was taken as measurement error allowance for calculation of criticality temperature. 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

3.1.2 <u>Heatup and Cooldown Rates</u> (Continued)

<u>Basis</u> (Cont'd)

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 177°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 average heatup and cooldown rates to 100°F/h when all PCS cold legs are $\geq$ 350°F 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 7, 8 and 9 are met. Calculation of average hourly cooldown rate after cooling to a temperature range requiring a lower cooldown rate shall be only from the time the lower cooldown rate is required. The core operational limit applies only when the reactor is critical.

3.1.2 Heatup and Cooldown Rates (Continued)

Basis (Continued)

The heatup and cooldown rate restrictions are consistent with the analyses performed for low temperature overpressure protection (LTOP) (References 13, 14 and 15). Below 430°F, the Power Operated Relief Valves (PORVs) provide overpressure protection; at 430°F or above, the PCS safety valves provide overpressure protection.

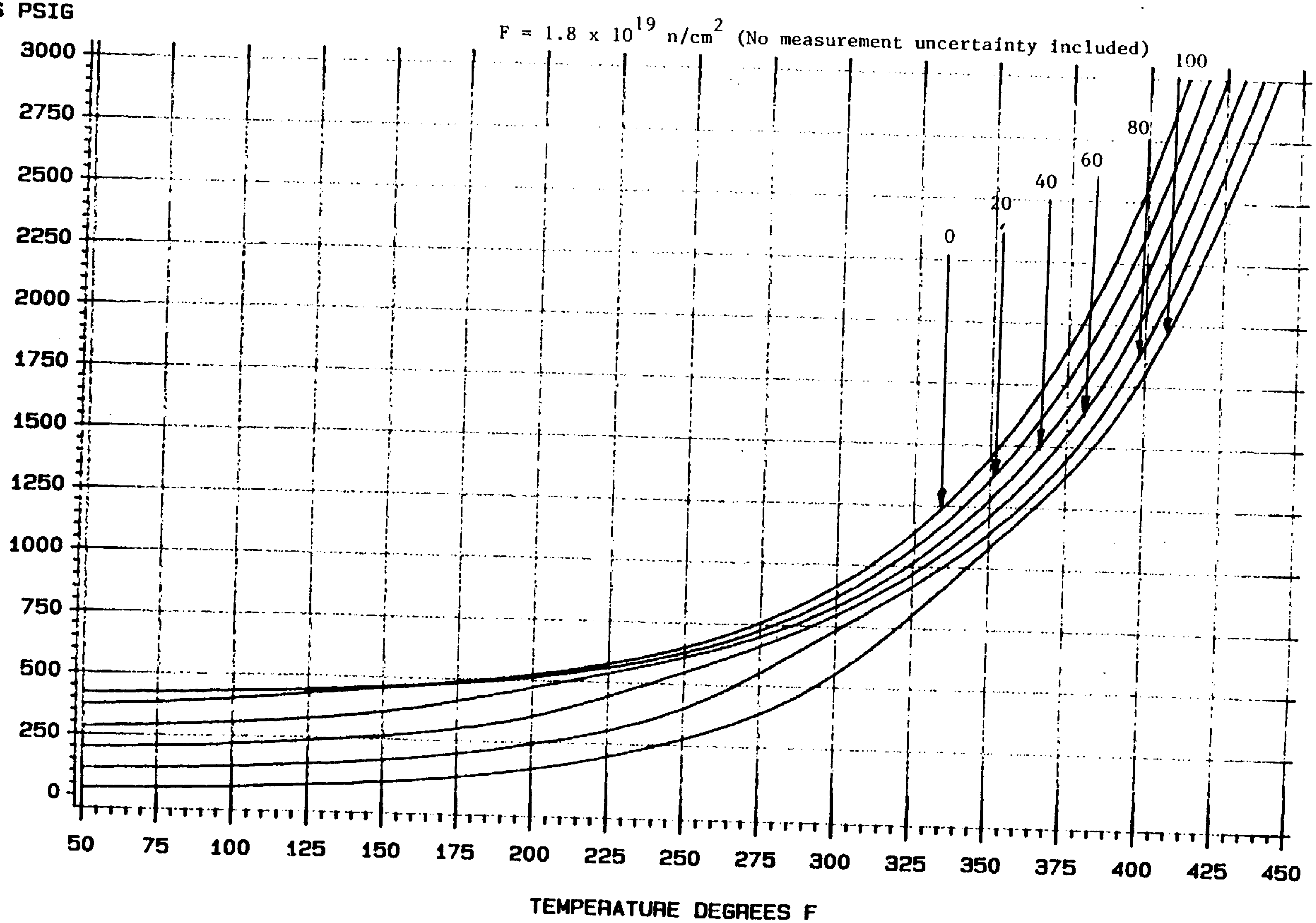
The criticality temperature is determined per Reference 8 and the core operational curves adhere to the requirements of Reference 9. The inservice test curves incorporate allowances for the thermal gradients associated with the heatup curve used to attain inservice test pressure. These curves differ 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) (Deleted)
(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 as amended November 6, 1986.
(10) US Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, May, 1988.
(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.
(13) "Analysis of Fast Neutron Exposure of the Palisades Reactor Pressure Vessel" by Westinghouse Electric Corporation, March 1989.
(14) Consumers Power Company Engineering Analysis EA-FC-809-13, Rev 1 "Pressure Response Effect of VLTOP with Replacement PORVs."
(15) Consumers Power Company Engineering Analysis EA-A-PAL-89-98 "Palisades Pressure and Temperature Limits."

FIGURE 3-1

PALISADES PRESSURE AND TEMPERATURE LIMITS FOR HEATUP



Amendment No. ~~27~~, ~~41~~, ~~88~~, ~~88~~, ~~97~~, ~~117~~, 131

FIGURE 3-2

PALISADES PRESSURE & TEMPERATURE LIMITS FOR COOLDOWN

PRESS PSIG $F = 1.8 \times 10^{19}\ n/cm^2$ (No Measurement Uncertainty Included)

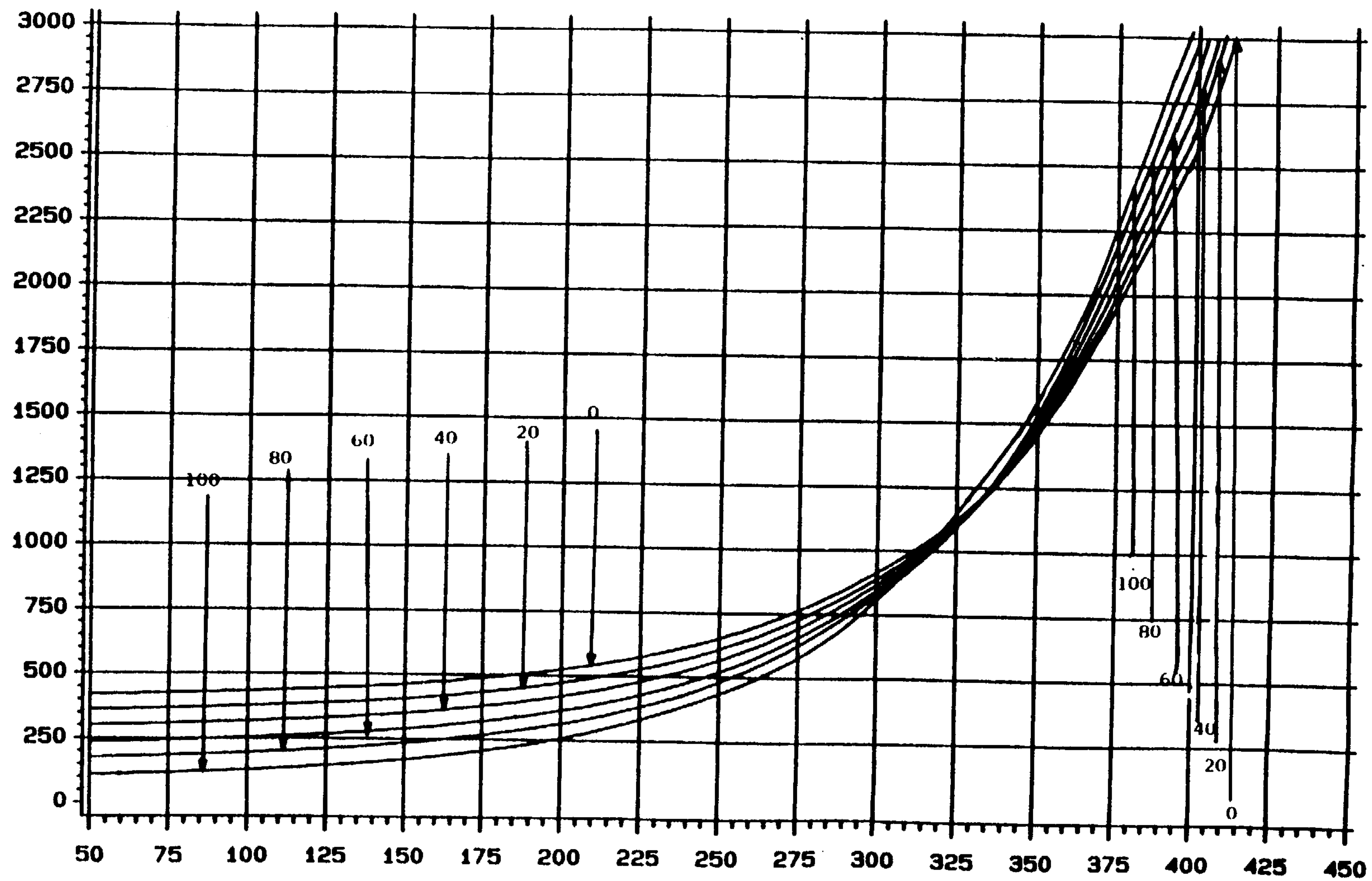


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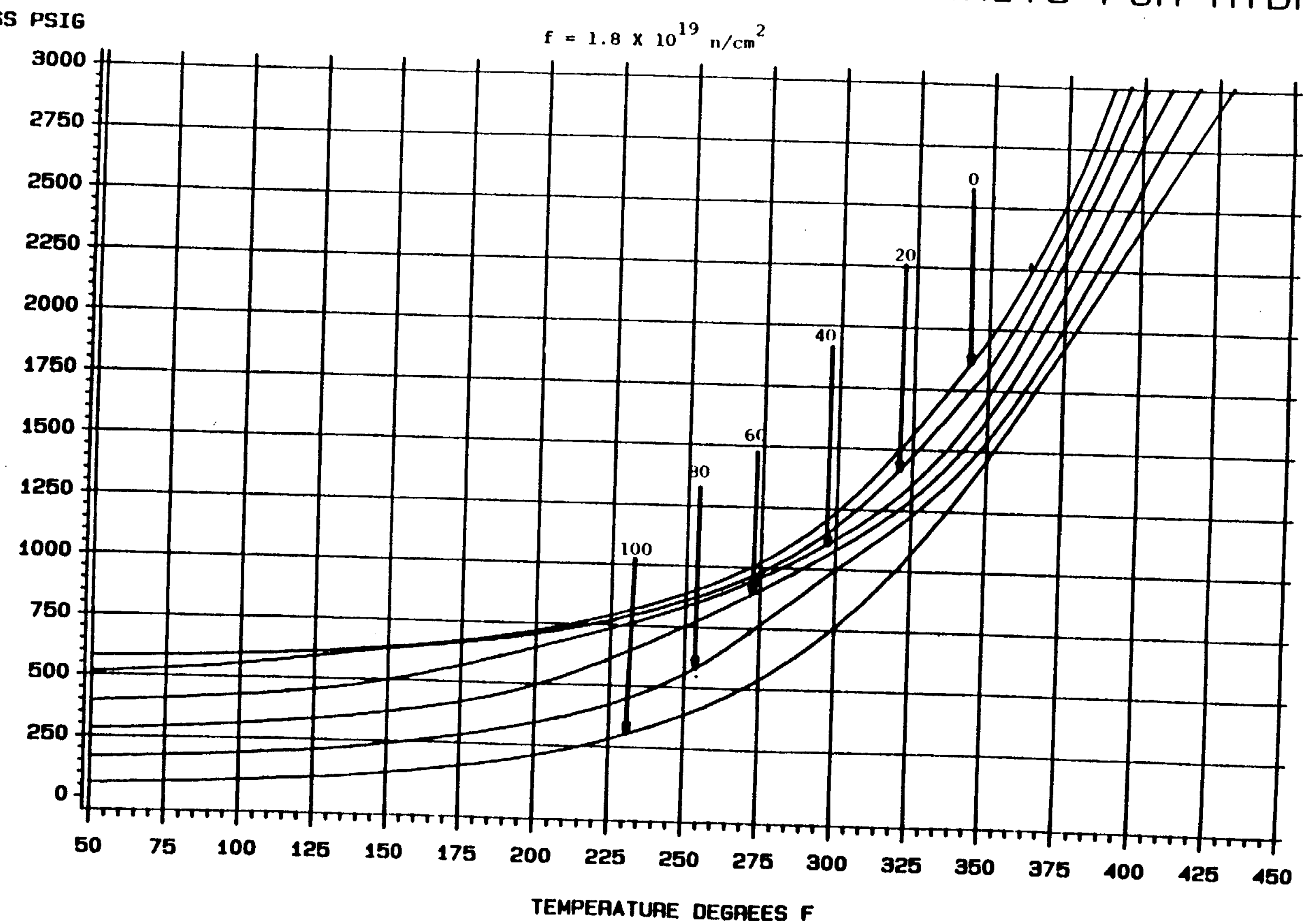
FIGURE 3-3

PALISADES PRESSURE AND TEMPERATURE LIMITS FOR HYDRO



LIMITING CONDITIONS FOR OPERATION

3.1.8.1 REQUIREMENTS

Two power operated relief valves (PORVs) with a lift setting below and/or to the right of the curve in Figure 3-4 shall be operable.

APPLICABILITY: When the temperature of one or more of the primary coolant system cold legs is less than 430°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize within the next 8 hours and either vent the PCS through a $\geq$ 1.3 square inch vent or open both PORV valves and both PORV block valves.

b. With both PORVs inoperable, depressurize within the next 8 hours and either vent the PCS through a $\geq$ 1.3 square inch vent or open both PORV valves and both PORV block valves.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Basis

There are three pressure transients which could cause the PCS pressure to exceed the pressure limits required by 10CFR50 Appendix G. They are: (1) a charging/letdown imbalance, (2) the start of high pressure safety injection (HPSI), and (3) initiation of forced circulation in the PCS when the steam generator temperature is higher than the PCS temperature.

Analysis (Reference 3) shows that when three charging pumps are operating and letdown is isolated and a spurious HPSI occurs between 260°F and 430°F, the PORV setpoints ensure that 10CFR50 Appendix G pressure limits will not be exceeded. Below 260°F, overpressure protection is still provided by the PORVs but HPSI operability is precluded by the limitations of Technical Specification 3.3.2 g. Above 430°F, the pressurizer safety valves prevent 10CFR50 Appendix G limits from being exceeded.

3.1.8. Basis (continued)

Assurance that the Appendix G limits for the reactor pressure vessel will not be violated while operating at low temperature is provided by the variable setpoint of the Low Temperature Overpressure Protection (LTOP) system. The LTOP system is programmed and calibrated to ensure opening of the pressurizer power operated relief valve (PORV) when the combination of primary coolant system (PCS) pressure and temperature is above or to the left of the limit displayed in Figure 3-4. That limit is developed from the more limiting of the heating or cooling limits for the specific temperature of the PCS while heating or cooling at the maximum permissible rate for that temperature. The limit in Figure 3-4 includes an allowance for pressure overshoot during the interval between the time pressurizer pressure reaches the limit, and the time a PORV opens enough to terminate the pressure rise. LTOP is provided by two independent channels of measurement, control, actuation, and valves, either one of which is capable of providing full protection. The actual setpoint of PORV actuation for LTOP will be lowered from the limit of Figure 3-4 to allow for potential instrument inaccuracies, measurement error, and instrument drift. This will ensure that at no time between calibration intervals will the combination of PCS temperature and pressure exceed the limits of Figure 3-4 without PORV actuation.

When the shutdown cooling system is not isolated (MO-3015 and MO-3016 open) from the PCS, assurance that the shutdown cooling system will not be pressurized above its design pressure is afforded by the relief valves on the shutdown cooling system, and the limitations of sections 3.1.1.h., 3.1.2.a & c, and 3.3.2.g.

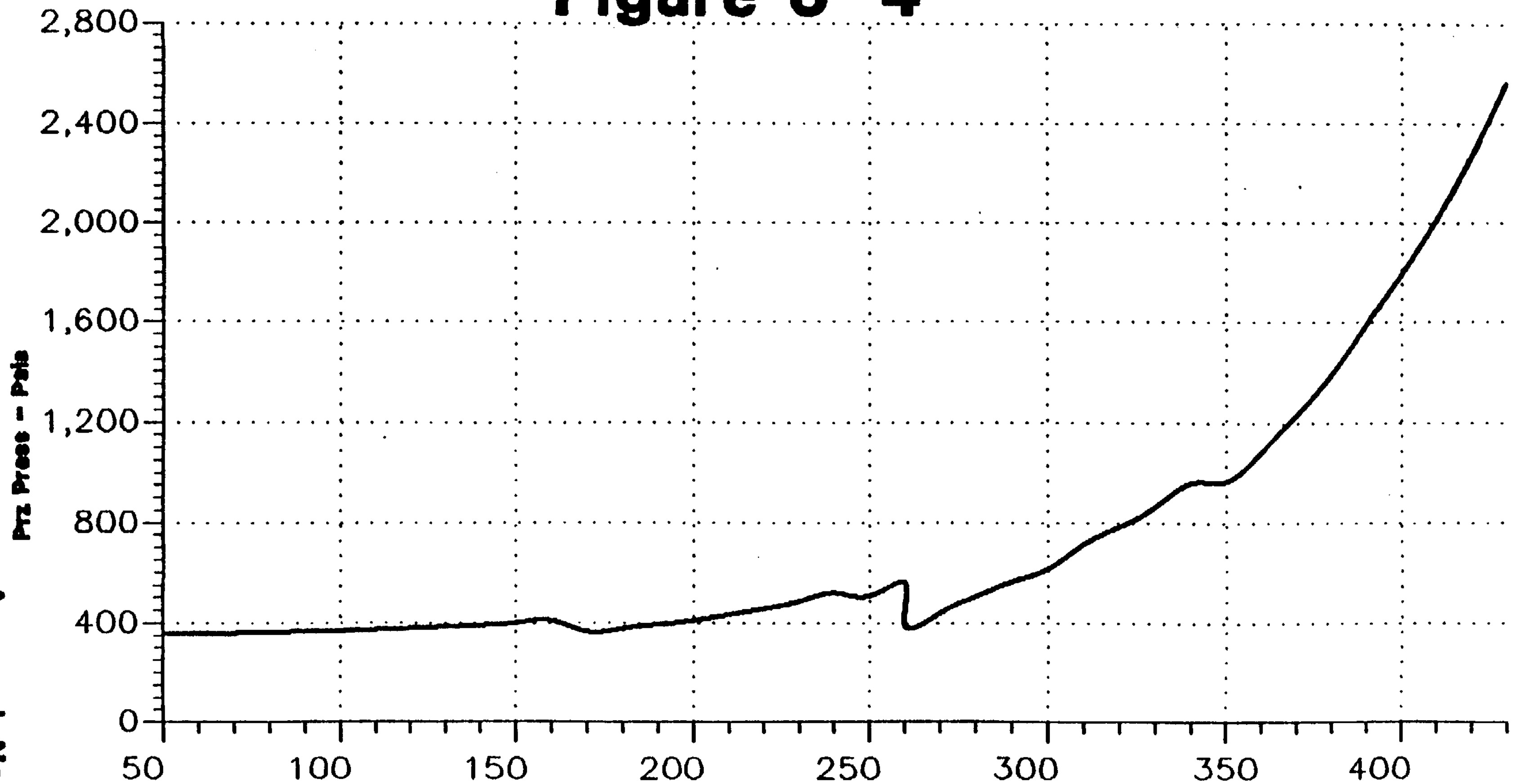
The requirement for the PCS to be depressurized and vented by an opening $\geq$ 1.3 square inches (Reference 4) or by opening both PORV valves and both PORV block valves when one or both PORVs are inoperable ensures that the 10CFR50 Appendix G pressure limits will not be exceeded when one of the PORVs is assumed to fail per the "single failure" criteria 10CFR50 Appendix A, Criterion 34. Since the PORV solenoid is strong enough to overcome spring pressure and valve disc weight, the PORVs may be held open by keeping the control switch in the open position.

References

1. Technical Specification 3.3.2
2. Technical Specification 3.1.2.
3. Consumers Power Company Engineering Analysis EA-FC-809-13, Rev 1
4. "Palisades Plant Overpressurization Analysis" June 1987 and "Palisades Plant Primary Coolant System Overpressurization Subsystem Description" October 1977.

LTOP LIMIT CURVE

Figure 3-4



PCS Degrees F

3.3 EMERGENCY CORE COOLING SYSTEM (Continued)

g. HPSI pump operability shall be as follows:

1) If the reactor head is installed, both HPSI pumps shall be rendered inoperable when:

a. The PCS temperature is < 260°F, or

b. Shutdown cooling isolation valves MO-3015 and MO-3016 are open.

2) Two HPSI pumps shall be operable when the PCS temperature is > 325°F.

3) One HPSI pump may be made inoperable when the reactor is subcritial provided the requirements of Section 3.3.2.c are met.

4) HPSI pump testing may be performed when the PCS temperature is <430°F provided the HPSI pump manual discharge valve is closed.

3.3.3 Prior to returning to the Power Operation Condition after every time the plant has been placed in the Refueling Shutdown Condition, or the Cold Shutdown Condition for more than 72 hours and testing of Specification 4.3.h has not been accomplished in the previous 9 months, or prior to returning the check valves in Table 4.3.1 to service after maintenance, repair or replacement, the following conditions shall be met:

a. All pressure isolation valves listed in Table 4.3.1 shall be functional as a pressure isolation device, except as specified in b. Valve leakage shall not exceed the amounts indicated.

b. In the event that integrity of any pressure isolation valve specified in Table 4.3.1 cannot be demonstrated, at least two valves in each high pressure line having a non-functional valve must be in and remain in, the mode corresponding to the isolated condition.(1)

[1] Motor-operated valves shall be placed in the closed position and power supplies deenergized.

Basis (continued)

demonstrate that the maximum fuel clad temperatures that could occur over the break size spectrum are well below the melting temperature of zirconium (3300°F).

Malfunction of the Low Pressure Safety Injection Flow control valve could defeat the Low Pressure Injection feature of the ECCS; therefore, it is disabled in the 'open' mode (by isolating the air supply) during plant operation. This action assures that it will not block flow during Safety Injection.

The inadvertent closing of any one of the Safety Injection bottle isolation valves in conjunction with a LOCA has not been analyzed. To provide assurance that this will not occur, these valves are electrically locked open by a key switch in the control room. In addition, prior to critical the valves are checked open, and then the 480 volt breakers are opened. Thus, a failure of a breaker and a switch are required for any of the valves to close.

Insuring both HPSI pumps are inoperable when the PCS temperature is < 260°F or the shutdown cooling isolation valves are open eliminates PCS mass additions due to inadvertent HPSI pump starts. Both HPSI pumps starting in conjunction with a charging/letdown imbalance may cause 10CFR50 Appendix G limits to be exceeded when the PCS temperature is < 260°F. When the PCS temperature is ≥ 430°F, the pressurizer safety valves ensure that the PCS pressure will not exceed 10CFR50 Appendix G.

The requirement to have both HPSI trains operable above 325°F provides added assurance that the effects of a LOCA occuring under LTOP conditions would be mitigated. If a LOCA occurs when the primary system temperature is less than or equal to 325°F, the pressure would drop to the level where low pressure safety injection can prevent core damage. Therefore, when the PCS temperature is ≥260°F and ≤325°F operation of the HPSI system would not cause the 10CFR50 Appendix G limits to be exceeded nor is HPSI system operation necessary for core cooling.

HPSI pump testing with the HPSI pump manual discharge valve closed is permitted since the closed valve eliminates the possibility of pump testing being the cause of a mass addition to the PCS.

References

(1) FSAR, Section 9.10.3;
(2) FSAR, Section 6.1,

b. The PCS vent(s) shall be verified to be open at least once per 12 hours when the vent(s) is being used for overpressure protection except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

c. When both open PORV valves are used as an alternative to venting the PCS, then verify both PORV valves and both PORV block valves are open at least once per 7 days.

Basis

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems when the plant is in operation, a checking frequency of once-per-shift is deemed adequate for reactor and steam system instrumentation. Calibrations are performed to insure the presentation and acquisition of accurate information.

The power range safety channels and ΔT power channels are are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at each refueling shutdown interval.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures. Thus, minimum calibration frequencies of one-per-day for the power range safety channels, and once each refueling shutdown for the process system channels, are considered adequate.

The minimum testing frequency for those instrument channels connected to the reactor protective system is based on an estimated average unsafe failure rate of 1.14×10^{-5} failure/hour per channel. This estimation is based on limited operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bonafide signal.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Applicability

Applies to the safety injection system, the containment spray system, chemical injection system and the containment cooling system tests.

Objective

To verify that the subject systems will respond promptly and perform their intended functions, if required.

Specifications

4.6.1 Safety Injection System

a. System tests shall be performed at each reactor refueling interval. A test safety injection signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this test. The system will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing (ie, the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

b Both high pressure safety injection pumps, P-66A and P-66B shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the PCS cold legs is < 260°F or if shutdown cooling valves MO-3015 and MO-3016 are open unless the reactor head is removed.

4.6.2 Containment Spray System

a. System test shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. At least every five years the spray nozzles shall be verified to be open.

c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

Basis (continued)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgment that more frequent testing would not significantly increase the reliability (ie, the probability that the component would operate when required), yet more frequent test would result in increased wear over a long period of time. Verification that the spray piping and nozzles are open will be made initially by a smoke test or other suitably sensitive method, and at least every five years thereafter. Since the material is all stainless steel, normally in a dry condition, and with no plugging mechanism available, the retest every five years is considered to be more than adequate.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the service water system and the containment air coolers. The SI tanks are a passive safety feature. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

With the reactor vessel head installed when the PCS cold leg temperature is less than 260°F, or if the shutdown cooling system isolation valves MO-3015 and MO-3016 are open, the start of one HPSI pump could cause the Appendix G or the shutdown cooling system pressure limits to be exceeded; therefore, both pumps are rendered inoperable.

References

(1) FSAR, Section 6.1.3.
(2) FSAR, Section 6.2.3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 131 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated September 12, 1989 (Ref. 1), and supplemented by letters dated September 22, 1989 (Ref. 2), September 25, 1989 (Ref. 3), and March 2, 1990 (Ref. 4), Consumers Power Company (CPCo) proposed certain revisions to the pressure/temperature (P/T) limits specified in the Palisades Plant Technical Specifications (TS), Sections 3.1, 3.2, and 3.3. The proposed revisions also would change the period of effectiveness of the P/T limits to 1.8E19 n/cm^2 neutron fluence or to about 10 effective full power years (EFPY) (Ref. 2). The proposed P/T limits were developed based on Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the primary coolant system (PCS) during heatup, cooldown, criticality, and hydrotest.

On July 12, 1988, the Commission issued Generic Letter 88-11 which advised the addressees that Regulatory Guide 1.99, Revision 2, became effective as of May 1988, and that the NRC would be using that revision of the guide in reviewing submittals related to pressure/temperature limits and for analyses that require an estimate (except for pressurized thermal shock) of embrittlement of reactor vessel beltline materials. Pressurized water reactor licensees also were reminded that low-temperature overpressure protection (LTOP) setpoints also might need revision because of the revised regulatory guide.

Consumers Power Company provided (Ref. 1 through 4) its updated P/T curves based on the guidance provided in Revision 2 of Regulatory Guide 1.99 and they are applicable until August 1993. CPCo also proposed system modifications to install larger PORVs with variable setpoints for LTOP function. In light of the above changes, CPCo also proposed changes to several other Technical Specifications to assure proper LTOP at the Palisades Plant.

2.0 EVALUATION

2.1 Pressure/Temperature Limit Considerations

The staff uses Appendices G and H of 10 CFR Part 50, the ASTM Standards and the ASME Code which are referenced in Appendices G and H, RG 1.99 (Rev. 2), and Standard Review Plan (SRP) Section 5.3.2 to evaluate the P/T limits. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits, and an acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to 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 prediction of the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). 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.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Palisades reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 1.8E19 n/cm^2 (about 10 EFPY) was the circumferential weld with 0.20% copper (Cu), 0.97% nickel (Ni), and an initial RT_{ndt} of -56°F (Ref. 5, 6, 7, 8).

CPCo has removed two surveillance capsules from Palisades. The results from capsule W-290 were published in Westinghouse report WCAP-10637 (Ref. 9). The results from capsule A-240 were published in Battelle-Columbus Report BCL-585-12 (Ref. 10). Both the surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal (Ref. 5, 11).

For the limiting beltline material, the circumferential welds, the staff calculated the ART to be 233°F at 1/4T (T = reactor vessel beltline thickness) and 177°F at 3/4T. The staff used a neutron fluence of 1.08E19 n/cm^2 at 1/4T and 0.39E19 n/cm^2 at 3/4T (Ref. 12).

CPCo used the method in RG 1.99, Rev. 2, to calculate an ART of 241°F at 1/4T and 177°F at 3/4T for the same limiting circumferential welds. The staff judges that CPCo's ART of 241°F is more conservative than the staff's ART of 233°F and is acceptable. Substituting the ART of 241°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, 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.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. Based on the flange reference temperature of 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. CPCo reported that the unirradiated USE for plate D-3804-2/C-1308-3 (Ref. 9, 10) in the longitudinal direction was 120 ft-lb. Based on a reduction factor of 0.65, the USE in the transverse direction was 78 ft-lb. Using Figure 2 of RG 1.99, Rev. 2, it was found that the expected USE at EOL (3.35E19 n/cm^2) was 43 ft-lb. CPCo has joined the Combustion Engineering Owners' Group to determine the effects of low USE values (less than 50 ft-lb) in beltline materials. Section V.E of Appendix G permits the licensee to continue to operate if it submits a proposed program for satisfying the requirements as set forth in V.C at least 3 years prior to the date when the USE value is predicted to fall below 50 ft. lb. The staff will review the report from the owners' group and monitor CPCo's low USE.

2.2 Low Temperature Overpressure Protection

In the current design, LTOP is provided by two pilot-operated relief valves (PORVs) with a lift setting of 310 psia when the PCS cold-leg temperature is less than 300°F and a lift setting of 575 psia when the PCS cold-leg temperature is less than 430°F. The PORVs must be aligned to the PCS when the PCS is below the specified temperature to provide assurance that the reactor vessel will be operated in the ductile region in accordance with 10 CFR Part 50, Appendix G, during both normal operation and overpressurization events due to equipment malfunction or operator error. Technical Specifications require alignment of the PORVs with proper lift settings to the PCS at PCS temperature below 430°F to prevent exceeding Appendix G P/T limits. When the PCS temperature is above 430°F, the pressurizer safety valves will provide overpressurization protection for the PCS.

CPCo proposes to replace the existing PORVs and block valves with larger valves. The LTOP system will be programmed and calibrated to ensure opening of the PORVs when the combination of reactor coolant pressure and temperature is above or to the left of the limits displayed in Figure 3-4 of the proposed Technical Specifications. This limit curve is developed from the more limiting of the heatup and cooldown limits for the specific PCS temperature while heating or cooling at the maximum permissible rate for that temperature. The limit in Figure 3-4 includes an allowance for pressure overshoot during the interval beginning when a PORV opens to terminate the pressure rise and when the PCS pressure reaches the limit. Either one separately of the two PORVs is sufficient to provide LTOP. The actual PORV lift setting will be lowered from the limit of Figure 3-4 to allow for potential instrument inaccuracies, measurement error, and instrument drift.

For potential mass addition transients leading to PCS overpressurization, CPCo's analysis was performed with several conservative assumptions which include: (1) inadvertent start of charging flow coincident with isolation of letdown, maximum pressurizer heating rate, and maximum PCS heating rate; (2) inadvertent start of HPSI coincident with inadvertent start of charging, maximum pressurizer heating rate and maximum PCS heating; (3) the PCS is water-solid at beginning of the transient; and (4) flow delivery from two HPSI pumps to the PCS is twice that from one HPSI pump.

The results of the mass addition transients indicate for the case of both HPSI pumps starting in conjunction with a charging/letdown imbalance, that 10 CFR

Part 50, Appendix G limits and the shutdown cooling system maximum allowable pressure will be exceeded when the PCS is less than 260°F. Thus, the proposed Technical Specification 3.3.2g requires that if the reactor head is installed, both HPSI pumps shall be rendered inoperable when the PCS temperature is less than 260°F or the shutdown cooling isolation valves are open.

For heat addition transients which lead to PCS pressure overshoot, the current Technical Specifications prohibit starting a reactor coolant pump (RCP) if the steam generator (SG) secondary water temperature is higher than the PCS temperature and the LTOP system is required to be in service. This is to preclude overpressurizing of the PCS due to reverse heat transfer in the SG when the RCP is started. However, the proposed LTOP modifications provide more flexible plant operation limits. The results of CPCo's analysis show that the operator could initiate a RCP when PCS is less than 430°F with shutdown cooling system isolated and the delta-T between the SG and the PCS not exceeding 20°F when the PCS temperature is between 170°F and 210°F and not exceeding 100°F when the PCS temperature is in the rest of temperature regions. These restrictions are specified in the proposed Technical Specification 3.1.1h. CPCo's analysis of heat addition transients was performed with conservative assumptions which include: (1) the PCS is assumed to be water-solid at the initiation of the transient; (2) pressurizer heaters are assumed to be on at the initiation of the transient; and (3) the maximum allowable delta-T between the SG and PCS temperatures exist at the initiation of the transient.

CPCo proposed changes of Technical Specifications 3.1.1h, 3.1.8, 3.3.2g, 4.1.1c, 4.6.1b, Figure 3-4 and their associated bases sections reflect the above discussed LTOP system modifications and the heatup and cooldown rates identified by the updated Figures 3-1, 3-2, and 3-3. The staff finds that they are reasonably conservative and acceptable.

3.0 CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 1.8E19 n/cm^2 neutron fluence or about 10 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. CPCo's submittal also satisfies Generic Letter 88-11 because CPCo used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Palisades Technical Specifications.

The staff will review the owners' group report on the effect of having low USE values before making a decision on CPCo's low predicted USE.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. 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 nor 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 (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: April 26, 1990

REFERENCES

1. Letter from K. Berry of Consumers Power Company to USNRC, "Technical Specification Change Request - LTOP/Variable Setpoint," dated September 12, 1989.

2. Letter from K. Berry of Consumers Power Company to USNRC, "Technical Specification Change Request - LTOP/Variable Setpoint - Revision 1," dated September 22, 1989.

3. Letter from R. Smedley of Consumers Power Company to USNRC, "Technical Specification Change Request - LTOP/Variable Setpoint - Revision 1 - Additional Information," dated September 25, 1989.

4. Letter from R. Smedley of Consumers Power Company to USNRC, "Technical Specification Change Request - LTOP/Variable Setpoint - Revision 2 - Additional Information," dated March 2, 1990.

5. May 23, 1978, Letter from D. P. Hoffman (CP) to D. L. Ziemann (USNRC), subject: Palisades Plant

6. October 12, 1984, Letter from T. U. Marston (EPRI) to J. Tosky (CP), subject: Reactor Vessel and Surveillance Welds for Palisades Plant

7. Attachment III: Summary of Findings Relative to Palisades Plant Reactor Vessel Materials, June 14, 1985

8. October 9, 1984, Letter from N. J. Porter (C-E) to J. B. Tosky (CP), subject: Palisades Vessel Weld Documentation

9. October 31, 1984, Letter from B. D. Johnson (CP) to H. R. Denton (USNRC), subject: Palisades Plant--Test Report of Reactor Vessel Specimen Capsules T-330 and W-290 Removed during the 1983 Refueling Cycle (WCAP-10637)

10. BCL-585-12, Final Report on Palisades Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule A-240, March 13, 1979, Battelle-Columbus Laboratories

11. February 5, 1985, Letter from B. D. Johnson (CP) to Director, USNRC Nuclear Reactor Regulation, subject: Palisades Plant--Reactor Pressure Vessel Surveillance Program Additional Information

12. November 30, 1988, Letter from R. W. Smedley (CP) to USNRC Document Control Desk, subject: Palisades Plant--Compliance with Pressurized Thermal Shock Rule 10CFR50.61 and Regulatory Guide 1.99, Revision 2--Fluence Reduction Status