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Docket No. 50-255

MAR 26 1980

Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Bixel:

The Commission has issued the enclosed Amendment No. 55 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your request dated July 2, 1979, and supplement thereto dated November 6, 1979.

This amendment changes the Technical Specifications to conform the operating limits to the requirements of 10 CFR Part 50, Appendix G for operation to 5.5×10^6 MWdt.

Copies of our Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Thomas M. Novak for

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 55 to
   License No. DPR-20
2. Safety Evaluation
3. Notice

cc w/enclosure:
See next page

2-29-80

8004080402

OFFICE	DOR:ORB # 2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:AD/SEP	
SURNAME	RDSilver:ah	HSmith		DLZiemann	RHVollmer	
DATE	2/25/80	2/25/80	2/ /80	3/5/80	3/6/80	

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

March 6, 1980

Docket No. 50-255

Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Bixel:

The Commission has issued the enclosed Amendment No. 55 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your request dated July 2, 1979, and supplement thereto dated November 6, 1979.

This amendment changes the Technical Specifications to conform the operating limits to the requirements of 10 CFR Part 50, Appendix G for operation to 5.5×10^6 MWdt.

Copies of our Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

for Thomas V. Wambach

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 55 to
   License No. DPR-20
2. Safety Evaluation
3. Notice

cc w/enclosure:
See next page

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One First National Plaza
Chicago, Illinois 60670

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Van Buren County, Michigan 49043

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Room 1 - Capitol Building
Lansing, Michigan 48913

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

Charles Bechhoefer, Esq., Chairman
Atomic Safety and Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. George C. Anderson
Department of Oceanography
University of Washington
Seattle, Washington 98195

Dr. M. Stanley Livingston
1005 Calle Largo
Santa Fe, New Mexico 87501

Resident Inspector
c/o U. S. NRC
P. O. Box 87
South Haven, Michigan 49090

Palisades Plant
ATTN: Mr. J. G. Lewis
Plant Manager
Covert, Michigan 49043

*w/cy of CPC filings dtd. 7/2/79 and 11/6/79

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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 55
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 July 2, 1979, as supplemented November 6, 1979, 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.

8004080405

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 Appendix A, as revised through Amendment No. 55, 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

Thomas V. Wambach

for Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 6, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 55

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the following pages and by inserting the enclosed pages. The revised pages contain the captioned amendment number and 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 (Figure 3-1)
3-10	3-10 (Figure 3-2)
3-11	3-11 (Figure 3-3)
3-12	3-12
3-13	3-13
3-14 through 3-16	---*

*Next page is 3-17 (Section 3.1.4).

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 the initial

3.1.2 Heatup and Cooldown Rates (Cont'd)

(2) (Cont'd)

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 6.75×10^{18} nvt is realized at the inner vessel wall at the beltline region, the following basis is established: 3.64×10^{19} nvt calculated at the reactor vessel beltline for 2540 MW_t for 40 years at a 80% load factor. This conversion has resulted in a correlation of 1.227×10^{12} nvt per 1 MWd_t.

(3) The limit lines in Figures 3-1 through 3-3 shall be moved parallel to the temperature axis in the direction of increasing temperature a distance associated with the RT_{NDT} increase during the period since the curves were last constructed. The RT_{NDT} increase will be based upon surveillance program testing of the specimens in the initial surveillance capsule.

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 testing of base line specimens associated with the reactor surveillance program indicates that the vessel weld has the highest RT_{NDT} of plate, weld and HAZ specimens at the fluence to which the Figures 3-1, 3-2 and 3-3 apply. (4,6) The unirradiated RT_{NDT} has been determined to be 0°F. (3,8) An RT_{NDT} of 0°F is used as an unirradiated value to which irradiation effects are added. In addition, this plate has been 100% volumetrically inspected by ultrasonic test using both

3.1.2 Heatup and Cooldown Rates (Cont'd)

Basis (Cont'd)

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 for 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 maximum integrated fast neutron ($E > 1$ MeV) exposure of the reactor vessel is computed to be 3.64×10^{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] The actual shift in RT_{NDT} will be established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4-11 of the FSAR. 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

3.1.2 Heatup and Cooldown Rates (Cont'd)

Basis (Cont'd)

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, $5^{\circ}F$ 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 $60^{\circ}F$ per hour.

The revised pressure-temperature limits are applicable to reactor vessel inner wall fluences of up to 6.75×10^{18} nvt or approximately 5.5×10^{6} MWd of thermal reactor power. The application of appropriate fluence attenuation factors at the 1/4 and 3/4 thickness locations results in fluences of 3.8×10^{18} nvt and $.95 \times 10^{18}$ nvt, respectively. From Reference 6, these fluences are extrapolated to RT_{NDT} shifts of $150^{\circ}F$ and $75^{\circ}F$, respectively, for the limiting weld

3.1.2 Heatup and Cooldown Rates (Cont'd)

Basis (Cont'd)

material. The criticality condition which defines a temperature below which the core cannot be made critical (strictly based upon fracture mechanics' considerations) is 286°F. The most limiting wall location is at 1/4 thickness. The minimum criticality temperature 286°F is the minimum permissible temperature for the inservice system hydrostatic pressure test. That temperature is calculated based upon 2100 psig operation pressure.

The restriction of heatup and cooldown rates to 100°F/h and the maintenance of a pressure-temperature relationship to the right of 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 1/4T 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.

3.1.2 Heatup and Cooldown Rates (Cont'd)

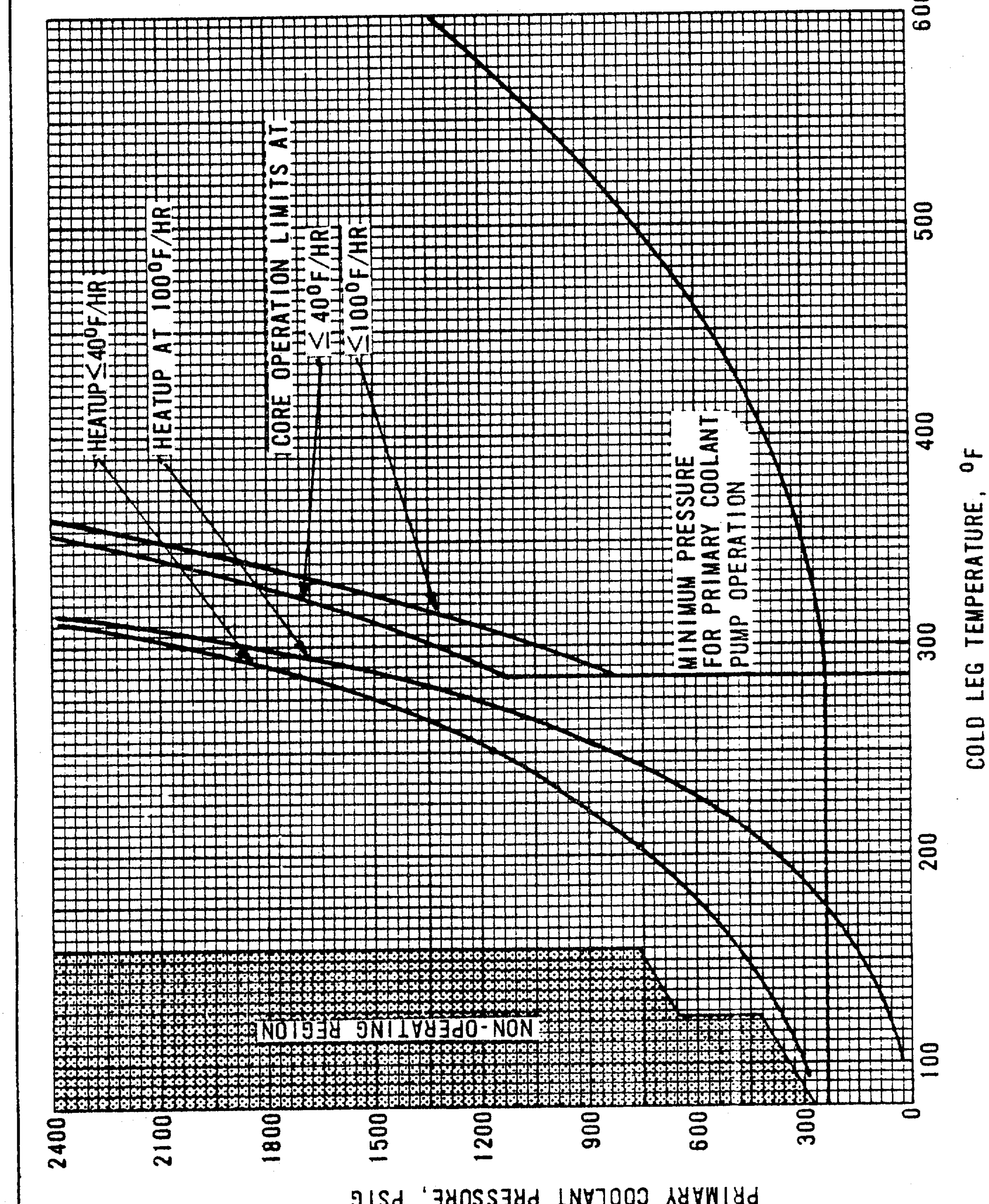
References

(1) FSAR, Section 4.2.2

(2) ASME Boiler and Pressure Vessel Code, Section III, N-415

(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.

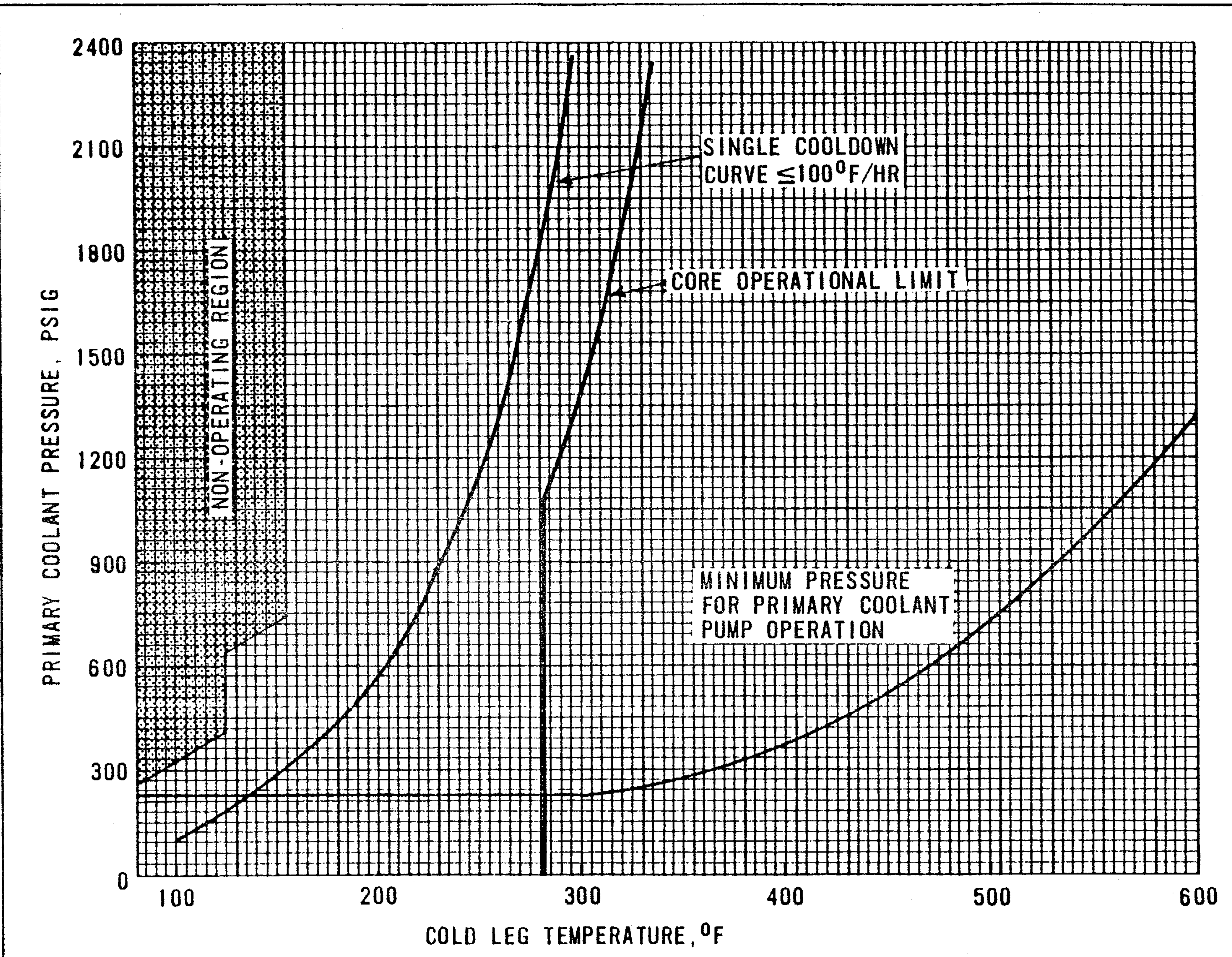


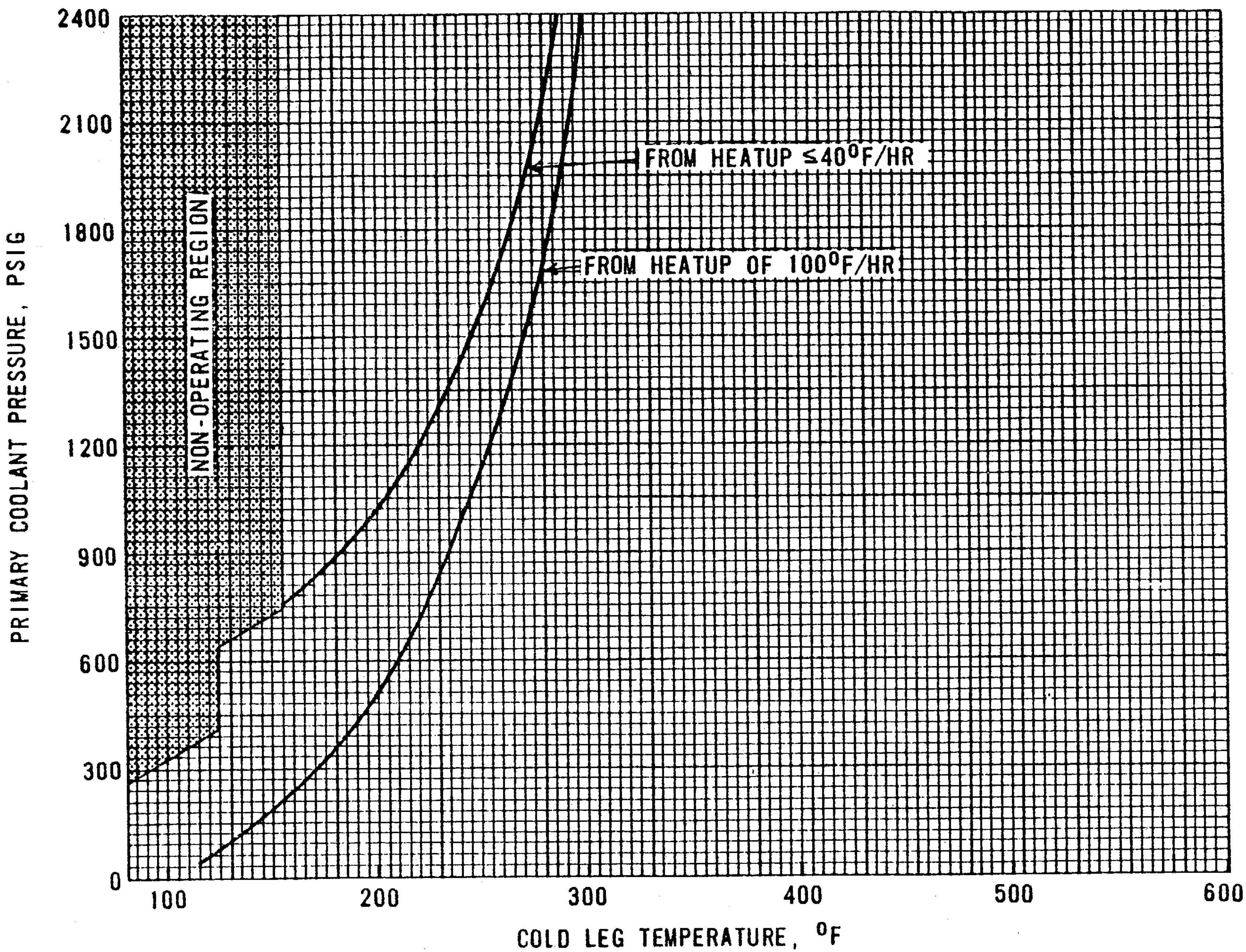
PALISADES PLANT TECH SPEC	PRESSURE-TEMPERATURE LIMITS FOR HEATUP-TO 5.5 X 10^6 MWDT FIGURE 3-1	DATE: AMEND. NO: ~~27~~, ~~41~~, 55

PALISADES PLANT
TECH SPEC

PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN-TO 5.5×10^6 MWDT
FIGURE 3-2

DATE:
AMEND. NO: 27, 41, 55





PALISADES PLANT
TECH SPEC

PRESSURE-TEMPERATURE LIMITS
INSERVICE TEST-TO 5.5 X 10^6 MWDT
FIGURE 3-3

DATE:
AMEND. NO: ~~27~~, ~~41~~, 55

3.1.2 Heatup and Cooldown Rates (Cont'd)

References (Cont'd)

(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," August 31, 1973.

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 286°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.

3.1.3 Minimum Conditions for Criticality (Cont'd)

Basis (Cont'd)

The moderator coefficient at lower temperatures will be less negative or more positive than at operating temperature.[1,2] It is, therefore, prudent to restrict the operation of the reactor when primary coolant temperatures are less than normal operating temperature ($\geq 525^{\circ}F$). 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 $525^{\circ}F$ 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 $286^{\circ}F$ 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 55 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated July 2, 1979, as revised November 6, 1979, Consumers Power Company (the licensee) requested an amendment to the Technical Specifications appended to License No. DPR-20 for the Palisades Plant. The amendment would conform the operating limits to the requirements of 10 CFR Part 50, Appendix G "Fracture Toughness Requirements" for operation to 5.5×10^6 MWdt.

2.0 DISCUSSION

10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The pressure-temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel beltline region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principle effect of the neutron irradiation is that it causes the vessel material nil-ductility temperature (RT_{NDT}) to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation induced increase in RT_{NDT} by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure-temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel are maintained within acceptable limits.

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The magnitude of the shift in RT_{NDT} is proportional to the neutron fluence that the materials are subjected to. The shift in RT_{NDT} can be predicted from Regulatory Guide 1.99. To check the validity of the predicted shift in RT_{NDT}, a reactor vessel material surveillance program is required. Surveillance specimens are periodically removed from the vessel and tested. The results of these tests are compared to the predicted shifts in RT_{NDT}, and the pressure-temperature operating limits are revised accordingly.

3.0 EVALUATION

Revised operating limits for Palisades are proposed for operation through 5.5×10^6 MWd of thermal reactor power. The limiting vessel material is weld metal having an initial value of RT_{NDT} of 0°F. At 5.5×10^6 MWd_t the maximum fluence on the vessel wall at the 1/4 T* location is estimated to be 3.78×10^{18} n/cm^2. Based on the test results performed on Palisades material surveillance specimens(1) and the damage predictions (increase in RT_{NDT}) in Regulatory Guide 1.99, this fluence causes the RT_{NDT} of the limiting weld metal to increase to 150°F. The revised operating limits are based on this value of RT_{NDT}.

We have reviewed the proposed operating limits for Palisades and have performed independent calculations to verify compliance with Appendix G, 10 CFR Part 50. We conclude that the limiting material is weld metal that will have an RT_{NDT} of 150°F at 5.5×10^6 MWd_t.

As part of our review of the operating limits, we also reviewed the Palisades material surveillance program. To date one surveillance capsule has been removed from the vessel and tested. Our estimates of radiation damage are based on the results of tests on materials contained in this capsule. We find that the Palisades surveillance program continues to be acceptable and in accordance with Appendix H, 10 CFR Part 50.

We conclude that the proposed operating limits are acceptable and are in conformance with Appendix G, 10 CFR Part 50 for operation to 5.5×10^6 MWd_t. Conformance with Appendix G to 10 CFR Part 50 in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requirements to NRC General Design Criterion 31, Appendix A, 10 CFR Part 50. We, therefore, find the proposed changes, as modified, acceptable.

---

* 1/4 T is one-fourth the thickness of the vessel wall, measured from the inside.

(1) "Palisades Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule A-240", March 13, 1979, submitted by Consumers Power Company letter of July 2, 1979.

4.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not involve a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: March 6, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 55 to Provisional Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised the Technical Specifications for operation of the Palisades Plant (the facility) located in Covert Township, Van Buren County, Michigan. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications to conform the operating limits to the requirements of 10 CFR Part 50, Appendix G for operation to 5.5×10^6 MWdt.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this action was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 2, 1979, and supplement thereto dated November 6, 1979, (2) Amendment No. 55 to License No. DPR-20, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan 49006. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 6th day of March, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

Thomas V. Wambach, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Hazel

Pressure Vessel Surveillance final

William O. Miller, Chief
License Fee Management Branch, ADM

Date: 7/20/79
Amended Form Date: 3/6/80

FACILITY AMENDMENT CLASSIFICATION - DOCKET NO(S). 50-255

Licensee: Consumers Power Co.

Plant Name and Unit(s): Palisades

License No(s): DPR-20 Mail Control No: 79-331

Request Dated: 7/2/79, 11/6/79 (suppl) Fee Remitted: Yes ✓ No ___

✓ Assigned TAC No: 8625

Licensee's Fee Classification: Class I ___, II ___, III ✓, IV ___, V ___, VI ___, None ___.

Amendment No. 55 Date of Issuance 3/6/80

(Appendix G Requirements)

[X] 1. This request has been reviewed by DOR/PM in accordance with Section 170.22 of Part 170 and is properly categorized.

[ ] 2. This request is incorrectly classified and should be properly categorized as Class ___. Justification for classification or reclassification: ___

[X] 3. ~~Additional information is required to properly categorize the request~~:
Our original fee position is hereby affirmed. RSilver (R Silver) for DLZ 3/5/80

[ ] 4. This request is a Class ___ type of action and is exempt from fees because it:

(a) ___ was filed by a nonprofit educational institution,

(b) ___ was filed by a Government agency and is not for a power reactor,

(c) ___ is for a Class ___ (can only be a I, II, or III) amendment which results from a written Commission request dated ___ for the application and the amendment is to simplify or clarify license or technical specifications, has only minor safety significance, and is being issued for the convenience of the Commission, or

(d) ___ other (state reason therefor): ___

HS 7/13/79

RS

R Silver

Richard J. Silver for D. L. Z.

Division of Operating Reactors/Project Management

[ ] The above request has been reviewed and is exempt from fees.

William O. Miller, Chief
License Fee Management Branch

Date

LFMB 6/78

... REQUIRED: LTTR 39 ENCL 38