Docket No. 50-255

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Consumers Power Company
ATTN: Mr. David Bixel
Nuclear Licensing Administrator
212 West Michigan Avenue
Jackson, Michigan 49201

MAY 16 1978

Gentlemen:

The Commission has issued the enclosed Amendment No. 41 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your requests dated February 22, 1978, as supplemented by letter dated March 7, 1978.

This amendment changes the Palisades Technical Specifications relating to the limits on primary coolant pressure and temperature for normal reactor operation, heatup and cooldown operations and tests.

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

Sincerely,

Original signed by
Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 41 to License No. DPR-20
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

DLZ

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*(w/cy. of CPC filings 2/22/78 & 3/7/78)

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230 South Dearborn Street
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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. 41
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 February 22, 1978, as supplemented by letter dated March 7, 1978, 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 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. 41, 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

Dennis L. Ziemann

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 41

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
3-10	3-10
3-10a	-
3-11	3-11
3-12	3-12
3-13	3-13
-	3-14 (Blank Page)
3-15	3-15
3-16	3-16

3.1 PRIMARY COOLANT SYSTEM (Contd)

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.

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.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.2 Heatup and Cooldown Rates (Contd)

(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 surveillance program capsule to be removed before the beginning of the Cycle 3. For purposes of determining fluence at the reactor vessel beltline for the present fuel cycle, the following basis was 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 1.23×10^{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.

3.1.2 Heatup and Cooldown Rates (Contd)

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 plate has the highest RT_{NDT} of plate, weld and HAZ specimens. The RT_{NDT} has been determined to be 0°F.(3) 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 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.(4)

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT 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

3.1.2 Heatup and Cooldown Rates (Contd)

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.[5] The predicted RT_{NDT} shift for a given fluence and copper-phosphorus weight percent has been made from a correlation for that purpose.[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 NDTT 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.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.2 Heatup and Cooldown Rates (Contd)

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 belt line. In addition, for calculational purposes, 5°F and 30 psi were taken as measurement error allowances for temperature and pressure, respectively. By Reference 7, reactor vessel wall locations at 1/4 and 3/4 thickness are limiting. It is at these locations that the crack propagation associated with the hypothetical flaw must be arrested. At these locations, fluence attenuation and thermal gradients have been evaluated. During cooldown, the 1/4 thickness location is always more limiting in that the RT_{NDT} is higher than that at the 3/4 thickness location and thermal gradient stresses are tensile there. During heatup, either the 1/4 thickness or 3/4 thickness location may be limiting depending upon heatup rate.

Figures 3-1 and 3-2 define stress limitations only from a fracture mechanic's point of view.

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

The revised pressure-temperature limits are applicable to reactor vessel inner wall fluences of up to 3.7×10^{18} nvt or approximately 3.0×10^{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

3.1.2 Heatup and Cooldown Rates (Contd)

of 2.26×10^{18} nvt and 5.2×10^{17} nvt, respectively. From Reference 6, these are consistent with RT_{NDT} shifts of 114°F and 55°F, respectively. The criticality condition which defines a temperature below which the core cannot be made critical (strictly based upon fracture mechanics' considerations) is RT_{NDT} + 132°F. The most limiting wall location is at 1/4 thickness. The minimum criticality temperature (246°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 1/4 T location is controlling and heatup curves converge. Cooldown curves cross for various cooldown rates, thus a composite curve is drawn. Due to the shifts in RT_{NDT}, NDTT requirements associated with nonreactor vessel materials are, for all practical purposes, no longer limiting.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.2 Heatup and Cooldown Rates (Contd)

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) FSAR, Section 4.2.4

(5) FSAR, Amendment 15

(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."

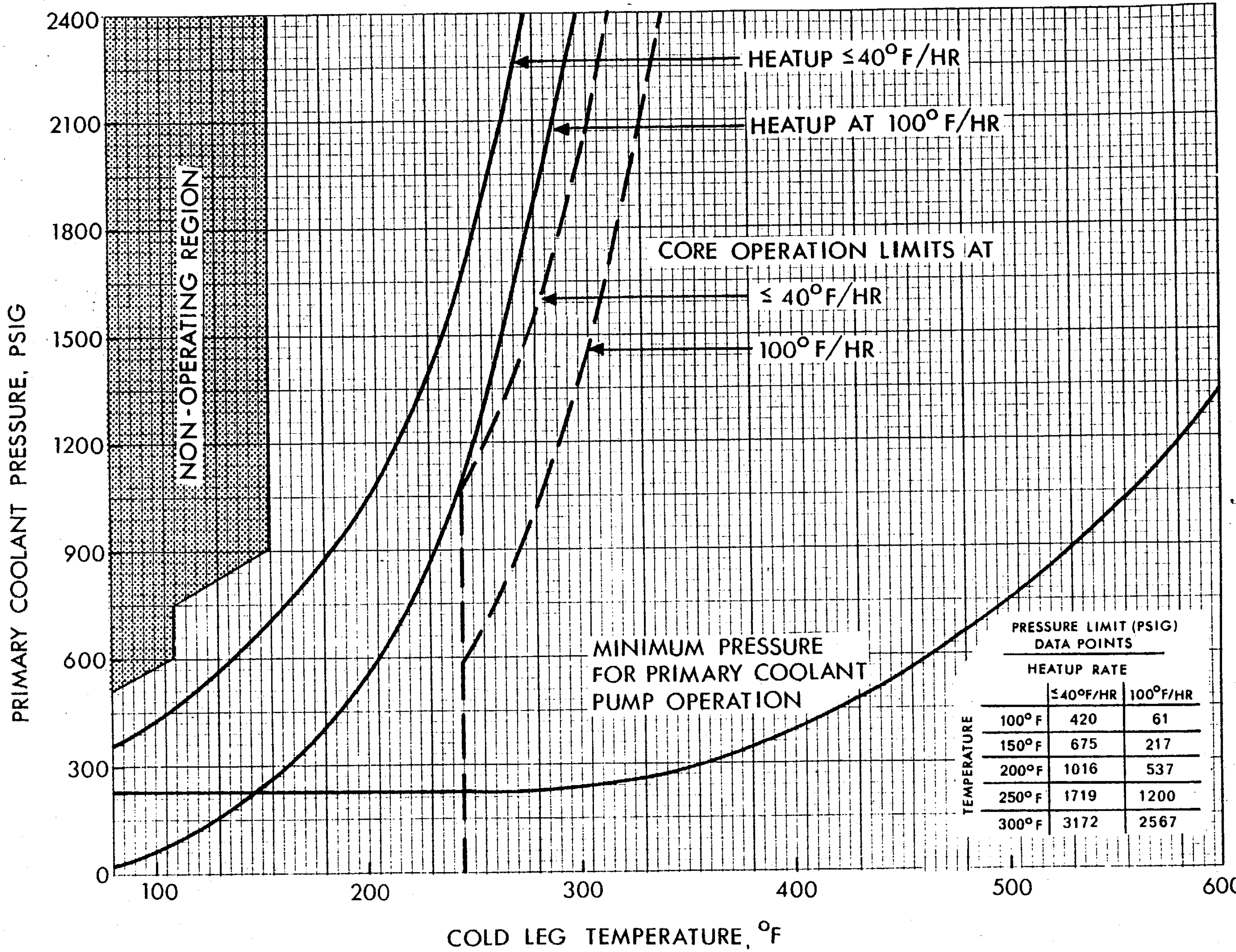
(9) 10 CF Part 50, Appendix G, "Fracture Toughness Requirements," August 31, 1973.

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 RT_{NDT} + 132°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.

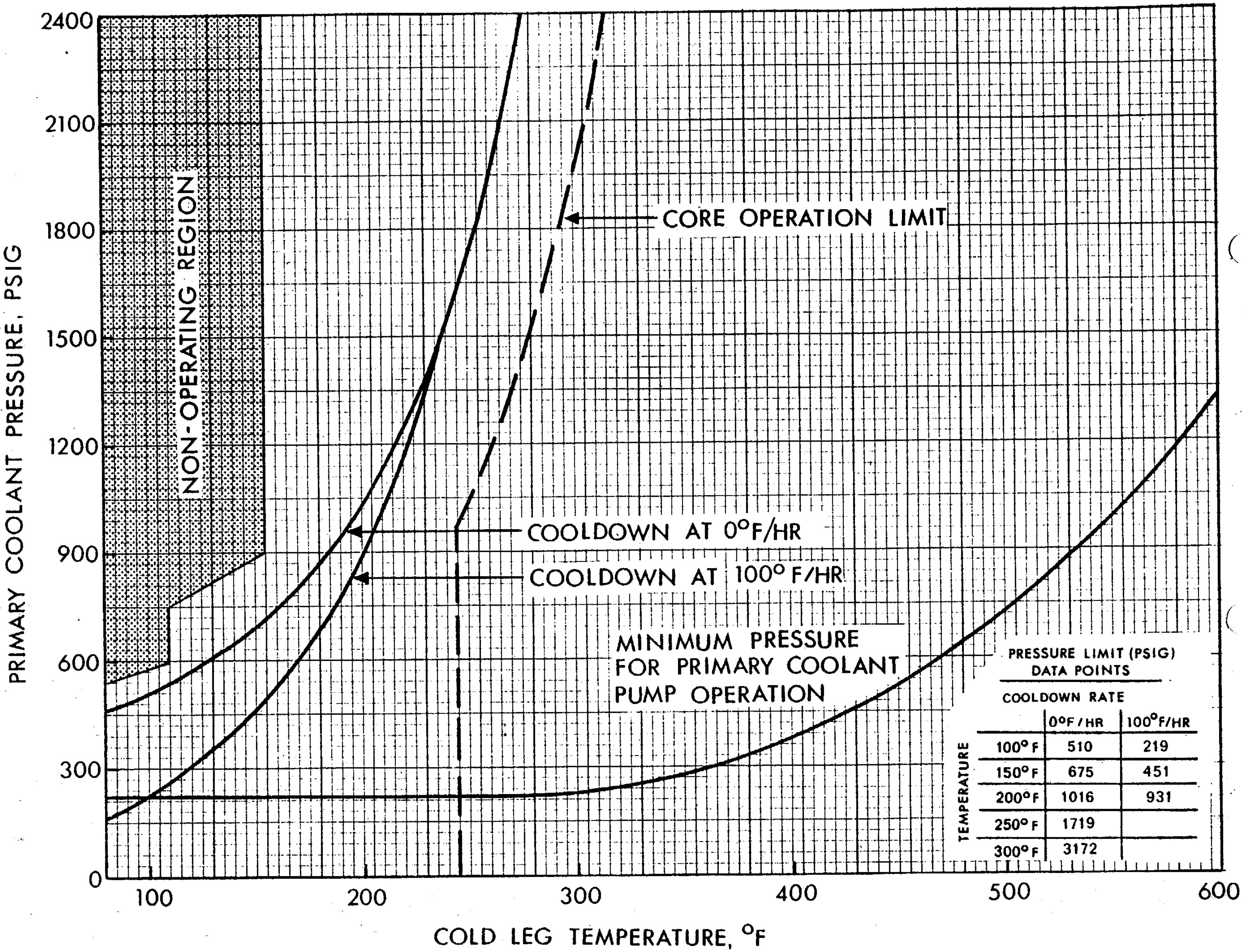


PRESSURE LIMIT (PSIG) DATA POINTS

TEMPERATURE	HEATUP RATE ≤40°F/HR	HEATUP RATE 100°F/HR
100° F	420	61
150° F	675	217
200° F	1016	537
250° F	1719	1200
300° F	3172	2567

PRESSURE-TEMPERATURE LIMITS FOR HEATUP - TO 3.0×10^6 MWDT
FIGURE 3-1

AMEND.NO: 27, 41



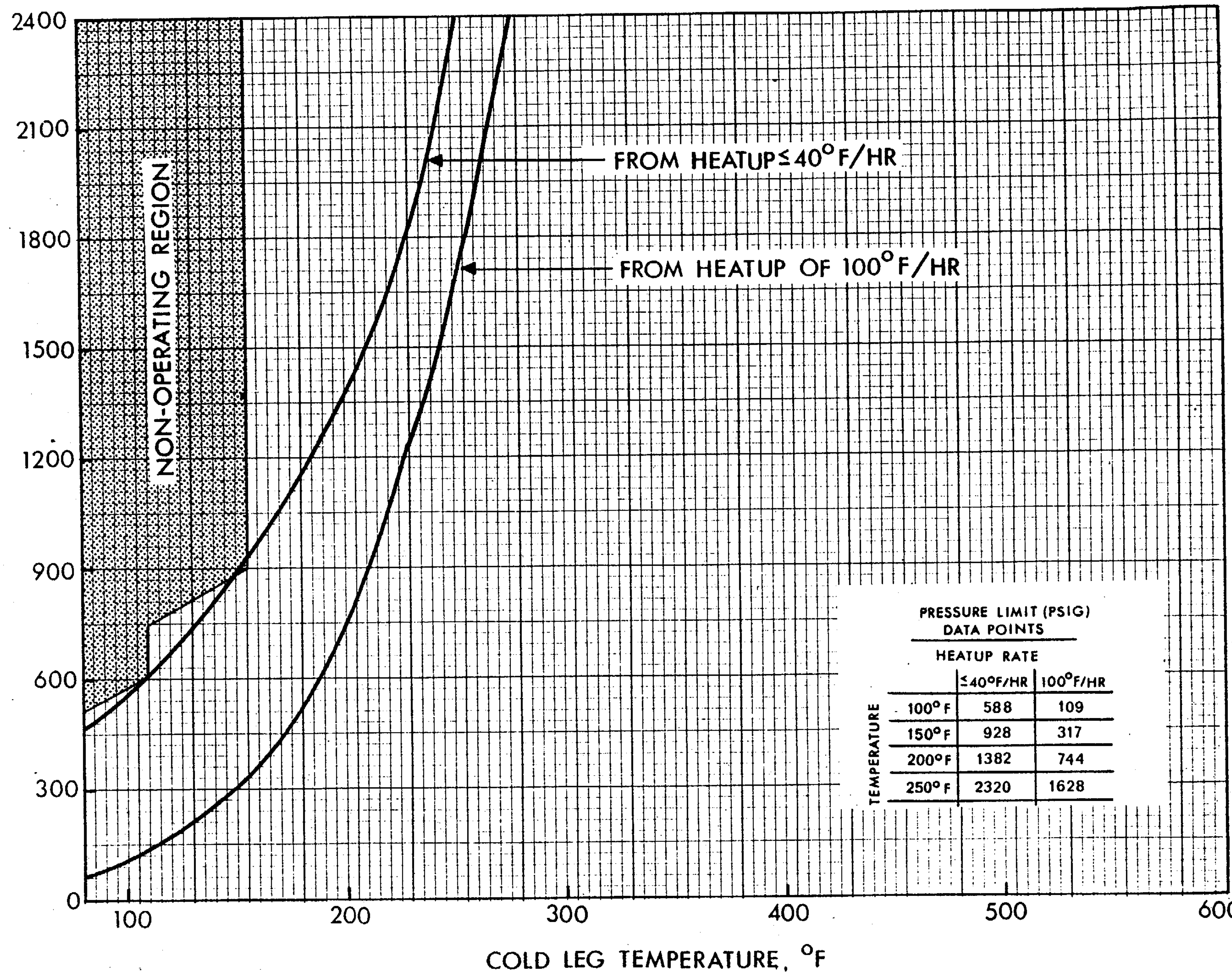
PRESSURE LIMIT (PSIG) DATA POINTS

TEMPERATURE	COOLDOWN RATE 0°F/HR	COOLDOWN RATE 100°F/HR
100° F	510	219
150° F	675	451
200° F	1016	931
250° F	1719	
300° F	3172	

PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN-TO 3.0×10^6 MWDT

FIGURE 3-2

AMEND.NO: 27, 41



PRESSURE LIMIT (PSIG)
DATA POINTS

TEMPERATURE	HEATUP RATE ≤40°F/HR	HEATUP RATE 100°F/HR
100° F	588	109
150° F	928	317
200° F	1382	744
250° F	2320	1628

PRESSURE - TEMPERATURE LIMITS
INSERVICE TEST - TO 3.0 X 10^6 MWDT
FIGURE 3-3

AMEND. NO: 27, 41

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3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.3 Minimum Conditions for Criticality (Contd)

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, 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.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.3 Minimum Conditions for Criticality (Contd)

The requirement that the reactor is not to be made critical below RT_{NDT} + 132°F provides increased assurance that the proper relationship between primary coolant pressure and temperature will be maintained relative to the NDTT of the primary coolant system. 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. 41 TO LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

Introduction

By letter dated February 22, 1978, as supplemented by letter dated March 7, 1978, the Consumers Power Company (the licensee) requested changes to Sections 3.1.2. and 3.1.3. of the Technical Specifications for Palisades Plant. The proposed changes would modify the limits on primary coolant pressure and temperature for normal reactor operation, heatup and cooldown operations and tests.

Discussion

Pressure-temperature limits for the primary coolant system are set to assure that operation conforms to the requirements of Appendix G, 10 CFR Part 50 and to assure adequate protection against non-ductile or rapidly propogating failure of the reactor coolant pressure boundary as required by Criterion 31 of Appendix A, 10 CFR Part 50. The considerations involved in setting the limits include material properties, effects of irradiation on materials properties, residual, steady-state and transient stresses, and size of flaws. The technical specifications for Palisades Plant presently include appropriate limiting conditions of operation including the above considerations for a reactor vessel inner wall irradiation up to 2.6×10^{18} nvt or approximately 2.2×10^6 megawatt days of thermal reactor power. This irradiation level is now being approached and revised pressure-temperature limits are needed to allow continued operation in the current Cycle 3.

In a letter dated January 6, 1977, Consumers Power Company requested a change to Technical Specifications 3.1.2 and 3.1.3 of Palisades regarding pressure-temperature operating limits. The proposed operating limits were calculated for reactor vessel irradiation through 3×10^6 MWd_t. We approved these operating limits for reactor vessel irradiation through 2.2×10^6 MWd_t.

The operating period for these pressure-temperature operating limits was reduced because the initial (unirradiated) value of RT_{ndt} of the reactor vessel materials was not accurately determined. Since this submittal, Battelle Columbus Laboratories have conducted tensile, Charpy V-notch and drop weight tests on unirradiated base, HAZ and weld specimens to establish baseline data for the Palisades material surveillance program. In letter dated February 22, 1978, Consumers Power Company submitted the results of the above test program and submitted a request to change Technical Specifications 3.1.2 and 3.1.3 for Palisades. The heatup/cooldown curves in the February 22 submittal were modified by Consumers Power Company letter dated March 7, 1978.

The proposed operating limits are proposed for operation through 3.0×10^6 MWd_t which is through the completion of the current cycle, Cycle 3. During Cycle 3, tests will be conducted on the first irradiated capsule removed from Palisades. Thus, for the next operating period, Cycle 4, pressure-temperature operating limits will be based on irradiated test data from the Palisades surveillance program. Since at present no irradiated capsules have been tested, the operating limits through 3.0×10^6 MWd_t are based on data from unirradiated material and radiation damage predictions of Regulatory Guide 1.99, Revision 1. Base material with a copper content of 0.25% is the limiting material for this operating period.

We have reviewed the proposed change to Technical Specification 3.1.2 and 3.1.3 and conclude that it is in accordance with Appendix G, 10 CFR Part 50 and is acceptable for operation through 3.0×10^6 MWd_t. Compliance with Appendix G in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

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.

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: May 16, 1978

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. 41 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 Palisades Technical Specifications relating to the limits on primary coolant pressure and temperature for normal reactor operation, heatup and cooldown operations and tests.

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 February 22, 1978 and supplement thereto dated March 7, 1978, (2) Amendment No. 41 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 16th day of May, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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