Docket No. 50-255

May 5, 1977

DISTRIBUTION
Docket
NRC PDR
Local PDR
ORB#1 Reading
VStello
KRGoller
SMSheppard
ASchwencer
GZech
OELD
OI&E(5)
BJones(4)
BScharf(15)
JMMcGough
JRBuchanan
TBAbernathy
DEisenhut
ACRS(16)
OPA(CMiles)
DRoss
BHarless

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 27 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 January 6, 1977.

This amendment makes revisions to the Palisades Plant Technical Specifications and ensures compliance with the fracture toughness requirements of Appendix G to 10 CFR Part 50 during heatup and cooldown operations, system hydrostatic tests and reactor core criticality.

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

Sincerely,

Original Signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 27 to DPR-20
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

Const. 1
GD

x27433:tsb

OFFICE ▶	ORB#1	OELD	ORB#1	OT:EB		
SURNAME ▶	GZech		ASchwencer	L.SHAO		
DATE ▶	4/25/77	5/3/77	5/5/77	4/25/77		

NRC Form 318 (2-76) NRCM 02040

cc: M. I. Miller, Esquire
Isham, Lincoln & Beale
Suite 4200
One First National Plaza
Chicago, Illinois 60670

J. L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Paul A. Perry, Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Myron M. Cherry, Esquire
Suite 4501
One IBM Plaza
Chicago, Illinois 60611

Kalamazoo Public Library
315 South Rose Street
Kalamazoo, Michigan 49006

Mr. Jerry Sarno
Township Supervisor
Covert Township
Route 1, Box 10
Van Buren County, Michigan 49043

Mr. John D. Beck (2 cys)
Division of Intergovernmental
Relations
Executive Office of the Governor
Lewis Cass Building, 2nd Floor
Lansing, Michigan 48913

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U.S. Environmental Protection Agency
Room 645, East Tower
401 M Street, SW
Washington, D.C. 20460

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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No.27
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 January 6, 1977, 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 Facility License No. DPR-20 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 27, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 5, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 27

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A as follows:

Remove Pages	Insert Revised Pages
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
	3-14
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 60°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. The average heatup and cooldown rates for the pressurizer shall not exceed 200°F/h in any one-hour time period.

d. Allowable combination of pressure and temperature for inservice testing from heatup or cooldown are for reference. Those curves include allowances for the temperature change rates notes above.

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

    (1) The curve in Figure 3-3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens show increases above this curve, a new curve shall be constructed

3.1 PRIMARY COOLANT SYSTEM (Contd)

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 and 3-2 apply, the limit lines on the figures shall be updated for a new integrated power period as follows. The total integrated reactor thermal power from start-up to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV). For this plant, 3.64×10^{19} nvt is the calculated fluence at the reactor vessel beltline for 40 years at 2540 MWt and an 80% load factor. The predicted transition temperature increase for the end of the new period may then be obtained from Figure 3-3.

(3) The limit lines in Figure 3-1 and Figure 3-2 shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature increase during the period since the curves were last constructed. For both figures, the lower vertical, intermediate and upper limit lines shall remain at 80°F, 110°F, and 160°F respectively, as they are set by the NDTT of the reactor vessel flange, steam generator and pressurizer manway covers, respectively, and are not subject to fast neutron flux. These vertical lines bound a nonoperating region defined by other than reactor vessel considerations. This area is that of previous technical specifications and it will not change in the future.

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 60°F and 100°F per hour, respectively, 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, and the material has been tested to verify conformity to specified requirements and to determine the actual initial maximum NDTT value of -30°F. An NDTT value of -30°F is used as the 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.[3]

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

3.1 PRIMARY COOLANT SYSTEM (Contd)

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.

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 MWt and 80% load factor.[4] The predicted NDTT shift for any integrated fast neutron ($E > 1$ MeV) exposure may be determined from Figure 3-3 or the formula noted in that figure. The NDTT shift of Figure 3.3 is a change from that used in previous technical specifications. The change is dictated by Reference (5) and reflects an analysis of surveillance coupon testing results from other nuclear reactor plants. The actual shift in NDTT 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

3.1 PRIMARY COOLANT SYSTEM (Contd)

caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown.

Reference (6) 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[6] is a function of NDTT, operating temperature, and vessel wall temperature gradients.

Pressure-temperature limit calculational procedures for the reactor coolant pressure boundary are defined in Reference (7) based upon Reference (6). These procedures are employed in Reference (8) to generate the curves of Figures 3-1 and 3-2.

The limit lines of Figures 3-1 and 3-2 consider a 54 psi pressure allowance to account for the fact that pressure is measured in the pressurizer rather than at the vessel beltline. In addition, for calculational purposes, 5°F and 30 psi were taken as measurement error allowances for temperature and pressure respectively. These allowances deviate slightly from those of previous technical specifications yet the effect upon the heatup and cooldown curves is negligible. By Reference (6), reactor vessel wall locations at 1/4 and 3/4 thickness are limiting. It is at these locations that the crack propagation

3.1 PRIMARY COOLANT SYSTEM (Contd)

associated with the hypothetical flaw must be arrested. At these locations, fluence attenuation and thermal gradients have been evaluated[8]. During cooldown, the 1/4 thickness location is always more limiting in that the NDTT 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 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 up to 2.6×10^{18} nvt or approximately 2.2×10^{6} MWD of thermal reactor power. The previous revision of the pressure-temperature limits is conservative with respect to the revised procedures of References (5, 6, 7) up to 1.8×10^{18} nvt or approximately 1.5×10^{6} MWD thermal. The application of fluence attenuation factors at the 1/4 and 3/4 thickness location results in section fluences of 1.34×10^{18} nvt and 3.1×10^{17} nvt respectively. It may be determined from Figure 3-3 that the appropriate NDTT shifts are 85 °F at the 1/4 thickness location (heatup or cooldown limiting) and 40°F at the 3/4 thickness location (heatup limiting). This implies that the NDTT temperatures at the 1/4 and 3/4 thickness locations at 2.6 x 10^{18} nvt are 55°F and 10°F respectively. The criticality condition which defines a temperature below which the core cannot be made critical (strictly based upon fracture mechanics considerations) is NDTT + 139°F. Based upon References (7) and (8), the NDTT is taken at the most limiting wall location. The most limiting wall location is at 1/4 thickness. The minimum criticality temperature (224°F) is the minimum permissible temperature for the inservice system hydrostatic pressure test. That temperature is calculated based upon a potential 2250 psig operation pressure.

With respect to the curves of Figures 3-1 and 3-2, the hatched area is nonoperating from other than reactor vessel limitations. That region has not and will not change from previous technical specifications. The heatup and cooldown curves are for situations where the reactor is not critical. Restriction of the heatup rate to 60°F/h and maintenance of a pressure-temperature relationship to the right of the heatup curve of Figure 3-1 insures that the requirements of References (5, 6, 7 and 9) are fulfilled with respect to heatup at any location in the vessel wall. Similarly, restriction of the cooldown rate to 100°F/h and maintenance of a pressure-temperature relationship to the right of the cooldown curve of Figure 3.2 fulfills the referenced requirements with respect to cooldown. The core operational limits apply only when the reactor is critical. The minimum criticality temperature is determined from Reference (7). It may be observed that once the reactor is critical, 3.1 ~~PRIMARY COOLANT~~ SYSTEM (Contd)

there is at least 40°F temperature margin between any pressure at a given heatup or cooldown rate and that same pressure at the same heatup or cooldown rate in a reactor noncritical situation. At higher pressures where a core operational limit for a given heatup or cooldown rate is not vertical, the noncritical curve is shifted exactly 40°F.

The inservice test curve is drawn per Reference (7). The heatup inservice test curve associated with the 1/4 thickness location is drawn in Figure 3-1 in that it is more limiting than the 3/4 thickness location curve. The heatup inservice test curve is calculated upon the basis of no thermal gradient across the vessel wall. The cooldown inservice test curve also relates to the 1/4 thickness location and conservatively assumes a thermal gradient associated with 100°F/h cooldown. Plant procedures provide that normal heatup or cooldown limit lines be used to achieve inservice test pressure. This guideline will continue to be used.

3.1 PRIMARY COOLANT SYSTEM (Contd)

References

(1) FSAR, Section 4.2.2.

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

(3) FSAR, Section 4.2.4.

(4) FSAR, Amendment 15.

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

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

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

(8) Battelle Report, "Development of Pressure-Temperature Operating Curves for Palisades," April 21, 1976.

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

PRESSURE-TEMPERATURE LIMITS FOR HEATUP

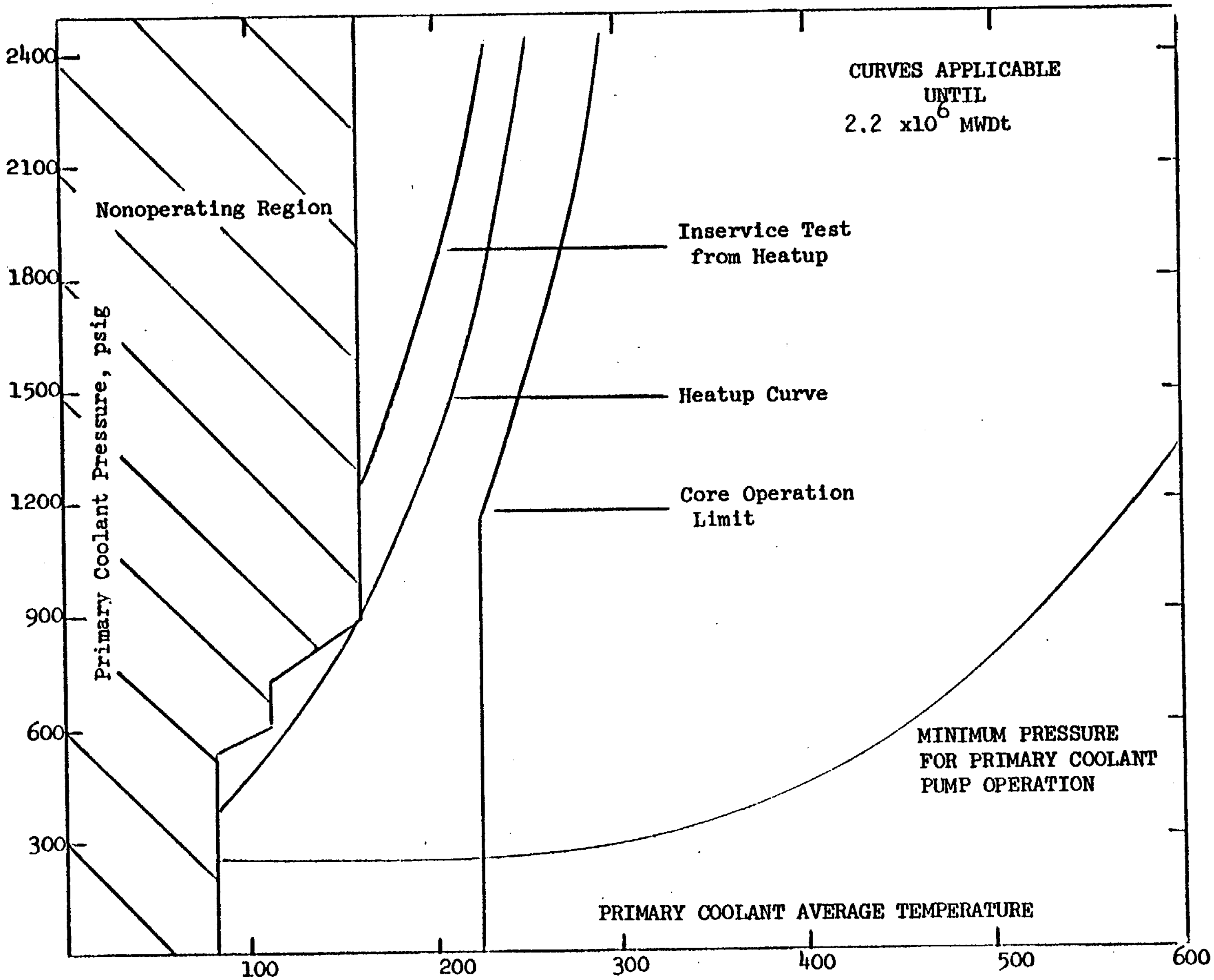
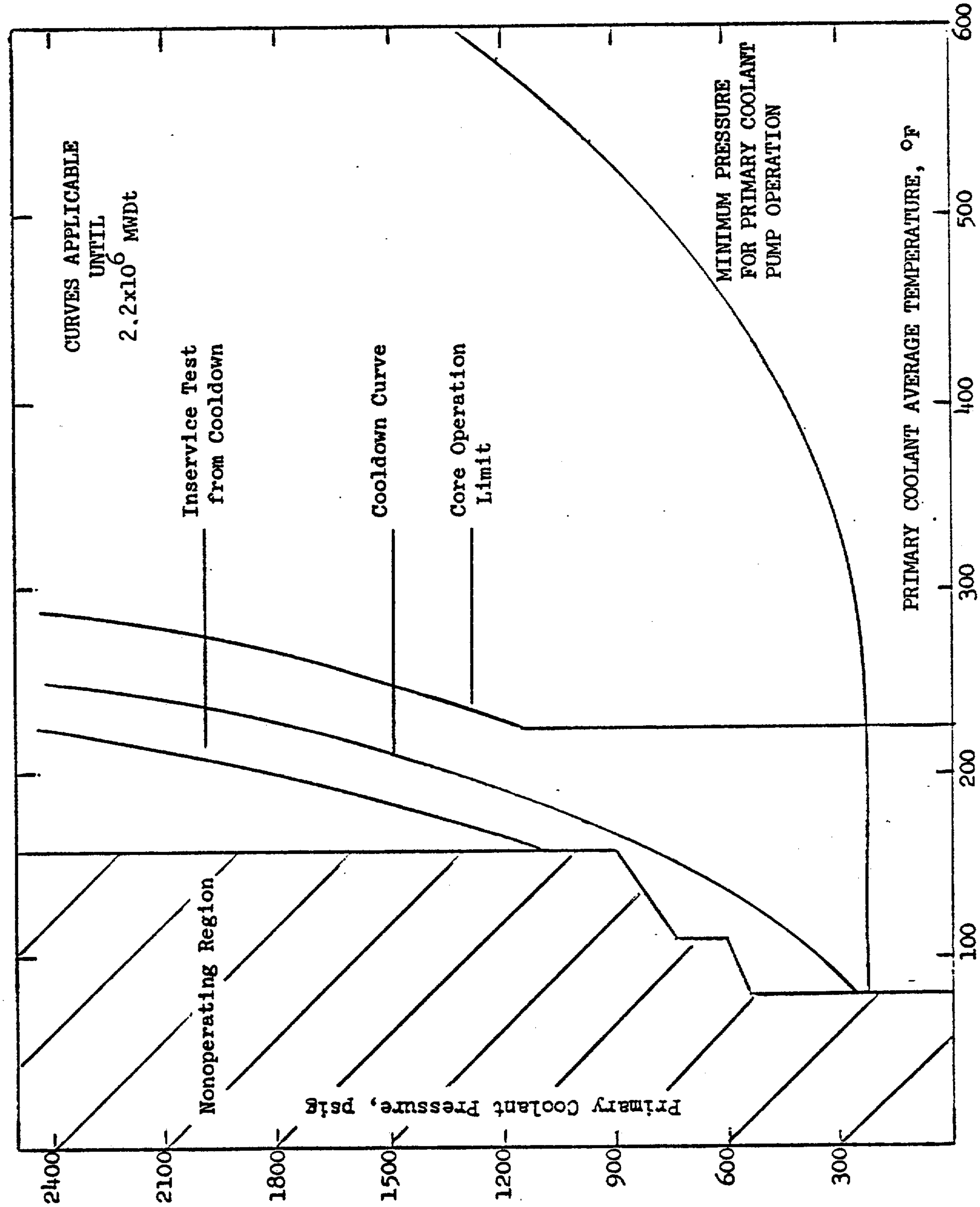


Figure 3-1

Figure 3-2

PRESSURE-TEMPERATURE RELATIONSHIP DURING COOLDOWN



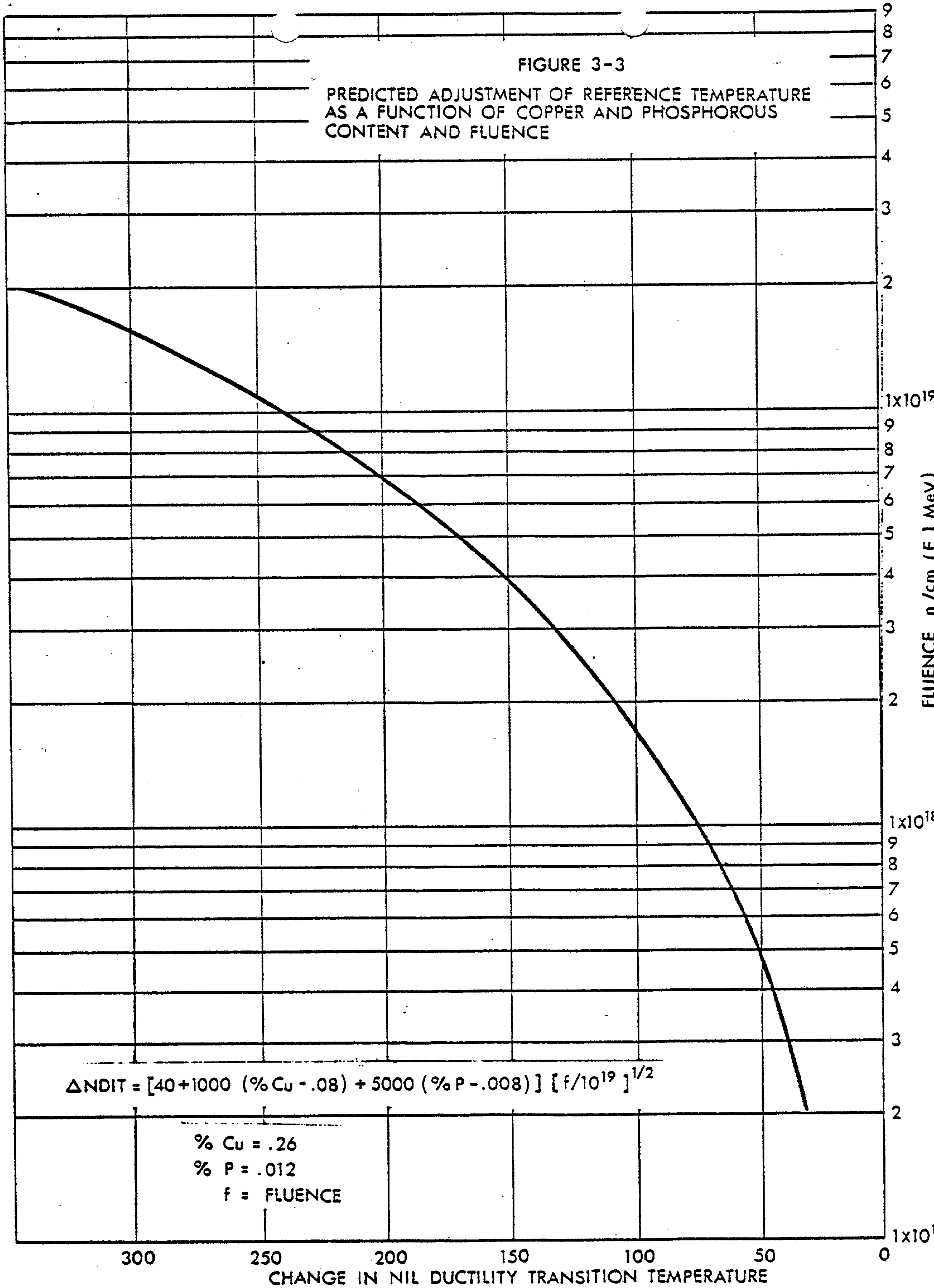


FIGURE 3-3

PREDICTED ADJUSTMENT OF REFERENCE TEMPERATURE AS A FUNCTION OF COPPER AND PHOSPHOROUS CONTENT AND FLUENCE

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.3 Minimum Conditions for Criticality

a. Except during low-power physics tests, 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 NDTT + 139°F.

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

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

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

Basis

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

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

3.1 PRIMARY COOLANT SYSTEM (Contd)

Assuming the most pessimistic rods out moderator coefficient, the maximum potential reactivity insertion that could result from depressurizing the coolant from 2100 psia to saturation pressure at 525°F is 0.1% $\Delta\rho$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated $\Delta\rho$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below NDTT + 139°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 of 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. 27 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

Introduction

By letter dated January 6, 1977, Consumers Power Company (the licensee) requested changes to the Technical Specifications appended to Provisional Operating License No. DPR-20, for operation of the Palisades Plant in Van Buren County, Michigan. The requested changes relate to heatup and cooldown operation, system hydrostatic tests and reactor core criticality.

Discussion

By letter dated March 1, 1976, we requested that the licensee review the reactor coolant system pressure temperature limits in the Palisades Technical Specifications to determine if they are in full compliance with Appendix G to 10 CFR Part 50. The review was to include the pressure-temperature limits for heatup and cooldown operation, system hydrostatic tests and reactor core criticality. By letter dated June 1, 1976, the licensee advised us that the current Palisades Technical Specifications were appropriately conservative for the then estimated reactor vessel fluence, but that revisions would be required in the near future. By letter dated January 6, 1977, the licensee proposed revised pressure-temperature curves and other appropriate Technical Specification changes.

Evaluation

The licensee's proposed pressure-temperature operating limits were calculated in accordance with Appendix G to 10 CFR Part 50 and Appendix G to ASME Code Section III. The limiting material is beltline weld material with a copper content of 0.20% and a phosphorous content of 0.012%. As indicated in the analysis submitted with the licensee's letter of January 6, 1977, this material is assumed to have an initial nil-ductility temperature,

(RT_{NDT}, of -30°F. (RT_{NDT} is defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code.) The shift in RT_{NDT} due to irradiation effects is calculated from Regulatory Guide 1.99 using the calculated fluence value on the vessel wall (for 40 years of operation with an 80% load factor) of 3.64×10^{19} n/cm^2. Pressure-temperature operating curves were proposed for operation through 3×10^6 thermal megawatt days (MWDt).

In the bases section of the licensee's proposed Technical Specifications, it is indicated that the initial maximum nil-ductility transition temperature, NDTT, was -30°F. This was determined from drop weight tests conducted on samples of reactor vessel material. An NDTT value of -30°F is therefore referred to as the unirradiated value to which irradiation effects are added. As indicated above, the analysis conducted to develop the proposed operating limits assumed that the initial RT_{NDT} was also -30°F. While RT_{NDT} and NDTT may be approximately the same, generally this is not a fully valid assumption. To confirm that the RT_{NDT} is the same as the NDTT, Charpy V-notch tests on beltline weld material are necessary. The licensee advised us that such tests are now being conducted by Battelle Columbus Laboratories on unirradiated specimens made from materials from the Palisades Plant reactor vessel. The licensee anticipates that the results of these tests will show that RT_{NDT} is the same as, or only slightly greater than, the NDTT temperature of -30°F. Based on the data collected on vessels with welds similar to those at Palisades Plant, we agree with this prediction. Because this has not yet been confirmed, however, the licensee has agreed to reduce the time period that the proposed pressure-temperature operating limits would be in effect. By reducing this time period, the fluence that the reactor vessel will be exposed to with these limits will be less and a smaller shift in RT_{NDT} will have occurred. This reduction in shift of RT_{NDT} by shortening the operating period will compensate for any potential difference that may exist between the RT_{NDT} and NDTT temperatures. The licensee has proposed to reduce the effective period of the operating limits from 3×10^6 MWDt to 2.2×10^6 MWDt which would reduce the accumulated fluence of the vessel from 3.7×10^{18} n/cm^2 to about 2.7×10^{18} n/cm^2 at the end of the period. This revised period is estimated to end about January of 1978.

We have reviewed the licensee's proposal and have concluded that the reduction in the effective period, for which the proposed operating limits are to apply, conservatively compensates for the minor uncertainty in the initial value of RT_{NDT} for the Palisades Plant pressure vessel material. In addition, the results of the Battelle Columbus Laboratories tests will be available in the summer of 1977, significantly prior to the end of the proposed operating period, so that adjustments can be made to revise the operating limits, if necessary. The licensee has agreed to provide us the results of these tests as soon as they are available. We have further concluded that the proposed pressure-temperature operating limits are in compliance with the requirements of Appendix G to 10 CFR Part 50. Compliance with Appendix G ensures that adequate safety margins exist during testing, operation, maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requirements of General Design Criteria 31 of Appendix A to 10 CFR Part 50.

Environmental Considerations

We have determined that the amendment does not authorize 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 5, 1977

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. 27 to Facility Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised 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.

This amendment makes revisions to the Palisades Plant Technical Specifications and ensures compliance with the fracture toughness requirements of Appendix G to 10 CFR Part 50 during heatup and cooldown operations, system hydrostatic tests and reactor core criticality.

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 amendment 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 January 6, 1977, (2) Amendment No. 27 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 5th day of May 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors