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

November 15, 1988

Docket No. 50-255

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

Dear Mr. Berry:

SUBJECT: AMENDMENT NO. 118 TO PROVISIONAL OPERATING LICENSE NO. DPR-20: MODIFICATION OF REACTOR PROTECTION SYSTEM, PEAKING FACTORS AND LINEAR HEAT RATE LIMITS (TACS NOS. 66901 and 69281)

The Commission has issued the enclosed Amendment No. 118 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated March 25, 1988, as supplemented by submittals dated June 17 and 27, 1988, and your application dated September 1, 1988, as supplemented September 19, 1988.

This amendment revises the provisions of the Technical Specifications to add limitations to plant operation with less than four reactor coolant pumps operating to conform to analyses for certain postulated accidents. It also modifies the limiting conditions for operation, reactor protection system setpoints, surveillance requirements, radial peaking factors and the linear heat rate limit. The enclosed Safety Evaluation also approves the reactor protection system modification proposed in the application and used in the accident and transient analyses submitted in support of the revised Technical Specifications. That part of the March 25, 1988 application that proposed a change to the differential pressure limit for the steam generators has been superseded by an application dated October 7, 1988, which will be addressed in a future action.

Your letter dated November 9, 1988 provided five commitments to resolve concerns raised in the course of our review. We accept those commitments and the schedule for documenting the results upon completion. The concerns are addressed in the enclosed Safety Evaluation and summarized in sections 2.5.1 and 2.5.2.

Sincerely,

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A copy of the Notice of Issuance is also enclosed.

Sincerely,
orignial signed by

Thomas V. Wambach, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 118 to License No. DPR-20
2. Safety Evaluation
3. Notice of Issuance

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A copy of the Notice of Issuance is also enclosed.

Sincerely,

Thomas V. Wambach

Thomas V. Wambach, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

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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.118
License No. DPR-20

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Consumers Power Company (the licensee) dated March 25 and September 1, 1988, comply 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 applications, 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.

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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. 118 , 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

Theodore Quay

Theodore Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 15, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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.

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Low Power Physics Testing

Testing performed under approved written procedures to determine control rod worths and other core nuclear properties. Reactor power during these tests shall not exceed 2% of rated power, not including decay heat and primary system temperature and pressure shall be in the range of 260°F to 538°F and 415 psia to 2150 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions therefor in these Technical Specifications.

Shutdown Boron Concentrations

Boron concentration sufficient to provide $k_{eff} \leq 0.98$ with all control rods in the core and the highest worth control rod fully withdrawn.

Refueling Boron Concentration

Boron concentration of coolant at least 1720 ppm (corresponding to a shutdown margin of at least 5% $\Delta\rho$ with all control rods withdrawn).

Quadrant Power Tilt

The difference between nuclear power in any core quadrant and the average in all quadrants.

Assembly Radial Peaking Factor - F_r^A

The assembly radial peaking factor is the maximum ratio of individual fuel assembly power to core average assembly power integrated over the total core height, including tilt.

Interior Fuel Rod

Any fuel rod of any assembly that is not on that assembly's periphery.

Total Interior Rod Radial Peaking Factor - $F_r^{\Delta H}$

The maximum product of the ratio of individual assembly power to core average assembly power times the highest interior local peaking factor integrated over the total core height including tilt.

1.1 REACTOR OPERATING CONDITIONS (Continued)

Axial Offset or Axial Shape Index

The power in the lower half of the core minus the power in the upper half of the core divided by the sum of the powers in the lower half and upper half of the core.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

Applicability

This specification applies when the reactor is in hot standby condition and power operation condition. / /

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the primary coolant.

Specifications

The MDNBR of the reactor core shall be maintained greater than or equal to 1.17. / / /

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high-cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of thermal power, primary coolant flow, temperature and pressure, can be related to DNB through the use of the XNB DNB / Correlation."[1] The XNB DNB Correlation has been developed to predict / DNB and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.17. A DNBR of / 1.17 corresponds to a 95% probability at a 95% confidence level that /

2.1 SAFETY LIMITS - REACTOR CORE (Contd)

DNB will not occur which is considered an appropriate margin to DNB for all operating conditions.[1]

The reactor protective system is designed to prevent any anticipated combination of transient conditions for primary coolant system temperature, pressure and thermal power level that would result in a DNBR of less than 1.17[3] The XNB DNB correlation has been shown to be applicable to the Palisades Plant in Reference 2.

References

(1) XN-NF-621(P)(A), Rev 1
(2) XN-NF-709
(3) Updated FSAR, Section 14.1.

2.2 SAFETY LIMITS - PRIMARY COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on primary coolant system pressure.

Objective

To maintain the integrity of the primary coolant system and to prevent the release of significant amounts of fission product activity to the primary coolant.

Specification

The primary coolant system pressure shall not exceed 2750 psia when there are fuel assemblies in the reactor vessel.

Basis

The primary coolant system[1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The maximum transient pressure allowable in the primary coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the primary coolant system piping, valves and fittings under ASA Section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings and capacity of the secondary coolant system safety valves (985-1025 psig)[3], the reactor high-pressure trip (≤2400 psia) and the primary safety valves (2500-2580 psia)[4] have been established to assure never reaching the primary coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psia (125% of design pressure) to verify the integrity of the primary coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the secondary coolant system steam dump and bypass valves at 900 psia. /

References

(1) Updated FSAR, Section 4. /
(2) Updated FSAR, Section 4.3. /
(3) Updated FSAR, Table 4-5 /
(4) Updated FSAR, Table 4-10 /

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM

Applicability

This specification applies to reactor trip settings and bypasses for instrument channels.

Objective

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1.

The TM/LP trip system monitors core power, reactor coolant maximum / inlet temperature, (T_{in}), core coolant system pressure and axial shape / index. The low pressure trip limit (P_{var}) is calculated using the / following equation. /

$$P_{var} = 1563.7(QA)(QR_1) + 12.3(T_{in}) - 6503.4$$ /

where:

$$QR_1 = 0.412(Q) + 0.588 \qquad Q \leq 1.0 \qquad Q = \frac{\text{core power}}{\text{rated power}}$$ /
$$= Q \qquad Q > 1.0$$ /

$$QA = -0.691(ASI) + 1.058 \qquad -0.653 \leq ASI < -0.156$$ /
$$= -0.521(ASI) + 1.085 \qquad -0.156 \leq ASI < +0.162$$ /
$$= 0.226(ASI) + 0.964 \qquad +0.162 \leq ASI \leq +0.544$$ /
$$= 1.085 \text{ when } Q < 0.0625$$ /

The calculated limit (P_{var}) is then compared to a fixed low pressure / trip limit (P_{min}). The auctioneered highest of these signals becomes / the trip limit (P_{trip}). P_{trip} is compared to the measured reactor / coolant pressure (P) and a trip signal is generated when P is less / than or equal to P_{trip}. A pre-trip alarm is also generated when P / is less than or equal to the pre-trip setting $P_{trip} + \Delta P$. /

TABLE 2.3.1

Reactor Protective System Trip Setting Limits

		Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating (4)
1.	Variable High Power (1)	≤10% above core power, with a minimum setpoint of ≤30% of rated power and a maximum of ≤106.5% of rated power	≤10% above core power with a minimum setpoint of ≤15% rated power and a maximum of ≤49% of rated power
2.	Primary Coolant Flow (2)	≥95% of Primary Coolant Flow With Four Pumps Operating	≥60% of Primary Coolant Flow With Four Pumps Operating
3.	High Pressure Pressurizer	≤2255 Psia	≤2255 Psia
4.	Thermal Margin/Low Pressure (2,3)	P_{trip} ≥ Applicable Limits	Replaced by Variable High Power Trip and 1750 Psia Minimum Low-Pressure Setting
5.	Steam Generator Low Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level
6.	Steam Generator Low Pressure (2)	≥500 Psia	≥500 Psia
7.	Containment High Pressure	≤3.70 Psig	≤3.70 Psig

(1) The VHPT can be 30% of rated power for power levels ≤ 20% of rated power.

(2) May be bypassed below 10^{-4}% of rated power provided auto bypass removal circuitry is operable. For low power physics tests, thermal margin/low pressure, primary coolant flow and low steam generator pressure trips may be bypassed until their react points are reached (approximately 1750 psia and 500 psia, respectively), provided automatic bypass removal circuitry at 10^{-1}% rated power is operable.

(3) Minimum trip setting shall be 1750 psia.

(4) Operation with three pumps for a maximum of 12 hours is permitted to provide a limited time for repair/pump restart, to provide for an orderly shutdown or to provide for the conduct of reactor internals noise monitoring test measurements.

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

1. Variable High Power - The variable high power trip (VHPT) is incorporated in the reactor protection system to provide a reactor trip for transients exhibiting a core power increase starting from any initial power level (such as the boron dilution transient). The VHPT system provides a trip setpoint no more than a predetermined amount above the indicated core power. Operator action is required to increase the setpoint as core power is increased; the setpoint is automatically decreased as core power decreases. Provisions have been made to select different set points for three pump and four pump operations.

   During normal plant operation with all primary coolant pumps operating, reactor trip is initiated when the reactor power level reaches 106.5% of indicated rated power. Adding to this the possible variation in trip point due to calibration and instrument errors, the maximum actual steady state power at which a trip would be actuated is 112%, which was used for the purpose of safety analysis.[1]

2. Primary Coolant System Low Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.[3] Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves. The percent of normal core flow is shown in the following table:

4 Pumps	100.0%
3 Pumps	74.7%

   During four-pump operation, the low-flow trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors.[4]

Basis (Contd)

Provisions are made in the reactor protective system to permit operation of the reactor at reduced power if one coolant pump is taken out of service. These low-flow and high-flux settings have been derived in consideration of instrument errors and response times of equipment involved to assure that thermal margin and flow stability will be maintained during normal operation and anticipated transients.[5] For reactor operation with one coolant pump inoperative, the low-flow trip points and the overpower trip points must be manually changed to the specified values for the selected pump condition by means of set point selector switches. The trip points are shown in Table 2.3.1.

3. High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the primary and secondary safety valves to prevent primary system overpressure (Specification 3.1.7). In the event of loss of load without reactor trip, the temperature and pressure of the primary coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. This setting is consistent with the trip point assumed in the accident analysis.[11]

Basis (Continued)

4. Thermal Margin/Low-Pressure Trip

The TM/LP trip set points are derived from the 4-pump operation core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. A pressure allowance of 165 psi is assumed to account for: instrument drift in both power and inlet temperatures; calorimetric power measurement; inlet temperature measurement; and primary system pressure measurement. Uncertainties accounted for that are not a part of the 165 psi term include allowances for: assembly power tilt; fuel pellet manufacturing tolerances; core flow measurement uncertainty and core bypass flow; inlet temperature measurement time delays; and ASI measurement. Each of these allowances and uncertainties are included in the development of the TM/LP trip set point used in the accident analysis.

For three-pump operation, continued power operation is restricted. During this mode of operation, the high power level trip in conjunction with the TM/LP trip (minimum set point = 1750 psia) and the secondary system safety valves (set at approximately 1000 psia) assure that adequate DNB margin is maintained.[5]

5. Low Steam Generator Water Level - The low steam generator water level reactor trip protects against the loss of feed-water flow accidents and assures that the design pressure of the primary coolant system will not be exceeded. The specified set point assures that there will be sufficient water inventory in the steam generator at the time of trip to allow a safe and orderly plant shutdown and to prevent steam generator dryout assuming minimum auxiliary feedwater capacity.[9]

The setting listed in Table 2.3.1 assures that the heat transfer surface (tubes) is covered with water when the reactor is critical.

Basis (Contd)

6. Low Steam Generator Pressure - A reactor trip on low steam generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the primary coolant. The setting of 500 psia is sufficiently below the rated load operating point of 739 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used in the accident analysis.[8]

7. Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down before the initiation of the safety injection system and containment spray.[10]

8. Low Power Physics Testing - For low power physics tests, certain tests will require the reactor to be critical at low temperature (≥ 260°F) and low pressure (≥ 415 psia). For these certain tests only, the thermal margin/low pressure, primary coolant flow and low steam generator pressure trips may be bypassed in order that reactor power can be increased for improved data acquisition. Special operating precautions will be in effect during these tests in accordance with approved written testing procedures. At reactor power levels below 10^{-1}% of rated power, the thermal margin/low-pressure trip and low flow trip are not required to prevent fuel rod thermal limits from being exceeded. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown, should a steam line break occur during these tests.

References

(1) ANF-87-150(P), Volume 2, Table 15.0.7-1
(2) deleted
(3) Updated FSAR, Section 7.2.3.3.
(4) ANF-87-150(P), Volume 2, Section 15.3
(5) XN-NF-86-91(P)
(6) deleted
(7) deleted
(8) XN-NF-77-18, Section 3.8
(9) ANF-87-150(P), Volume 2, Section 15.2.7
(10) Updated FSAR, Section 7.2.3.9.
(11) ANF-87-150(P), Volume 2, Section 15.2.1
(12) ANF-87-150(P), Volume 2, Section 15.0.7.2

3.1 PRIMARY COOLANT SYSTEM

Applicability

Applies to the operable status of the primary coolant system.

Objective

To specify certain conditions of the primary coolant system which must be met to assure safe reactor operation.

Specifications

3.1.1 Operable Components

a. At least one primary coolant pump or one shutdown cooling pump with a flow rate greater than or equal to 2810 gpm shall be in operation whenever a change is being made in the boron concentration of the primary coolant and the plant is operating in cold shutdown or above, except during an emergency loss of coolant flow situation. Under these circumstances, the boron concentration may be increased with no primary coolant pumps or shutdown cooling pumps running.

b. Four primary coolant pumps shall be in operation whenever the reactor is operated above hot shutdown, with the following exception:

   Before removing a pump from service, thermal power shall be reduced as specified in Table 2.3.1 and appropriate corrective action implemented. With one pump out of service, return the pump to service within 12 hours (return to four-pump operation) or be in hot shutdown (or below) with the reactor tripped (from the C-06 panel, opening the 42-01 and 42-02 circuit breakers) within the next 12 hours. Start-up (above hot shutdown) with less than four pumps is not permitted and power operation with less than three pumps is not permitted.

c. The measured four primary coolant pumps operating reactor vessel flow shall be 124.3×10^6 lb/hr or greater, when corrected to 532°F.

d. Both steam generators shall be capable of performing their heat transfer function whenever the average temperature of the primary coolant is above 325°F.

e. Maximum primary system pressure differentials shall not exceed the following:

   (1) Maximum steam generator operating differential of 1380 psi.

3.1.1 Operable Components (Continued)

(2) Hydrostatic tests shall be conducted in accordance with applicable paragraphs of Section XI ASME Boiler & Pressure Vessel Code (1974). Such tests shall be conducted with sufficient pressure on the secondary side of the steam generators to restrict primary to secondary pressure differential to a maximum of 1380 psi. Maximum hydrostatic test pressure shall not exceed 1.1 Po plus 50 psi where Po is nominal operating pressure.

(3) Primary side leak tests shall be conducted at normal operating pressure. The temperature shall be consistent with applicable fracture toughness criteria for ferritic materials and shall be selected such that the differential pressure across the steam generator tubes is not greater than 1380 psi.

(4) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum temperature of 100°F is required. Only ten cycles are permitted.

(5) Maximum secondary leak test pressure shall not exceed 1000 psia. A minimum temperature of 100°F is required.

(6) In performing the tests identified in 3.1.1.e(4) and 3.1.1.e(5), above, the secondary pressure shall not exceed the primary pressure by more than 350 psi.

f. Nominal primary system operation pressure shall not exceed 2100 psia.

g. The reactor inlet temperature (indicated) shall not exceed the value given by the following equation at steady state power operation: /

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

Where: T_{inlet} = reactor inlet temperature in F°
P = nominal operating pressure in psia
W = total recirculating mass flow in 10^6 lb/h corrected to the operating temperature conditions.

When the ASI exceeds the limits specified in Figure 3.0, within 15 minutes, initiate corrective actions to restore the ASI to the acceptable region. Restore the ASI to acceptable values within one hour or be at less than 70% of rated power within the following two hours. /

If the measured primary coolant system flow rate is greater than 130 M lbm/hr, the maximum inlet temperature shall be less than or equal to the T_{Inlet} LCO at 130 M lbm/hr. /

3.1.1 Operable Components (Cont'd)

h. During initial primary coolant pump starts (i.e., initiation of forced circulation), secondary system temperature in the steam generators shall be ≤ the PCS cold leg temperature unless the PCS cold leg temperature is ≥ 450°F.

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 or 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.(6) 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)

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.

Basis (Contd)

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 mayway cover of + 40°F.

The transient analyses were performed assuming a vessel flow at hot zero power (532°F) of 124.3 x 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 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.

Basis (Cont'd)

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. /

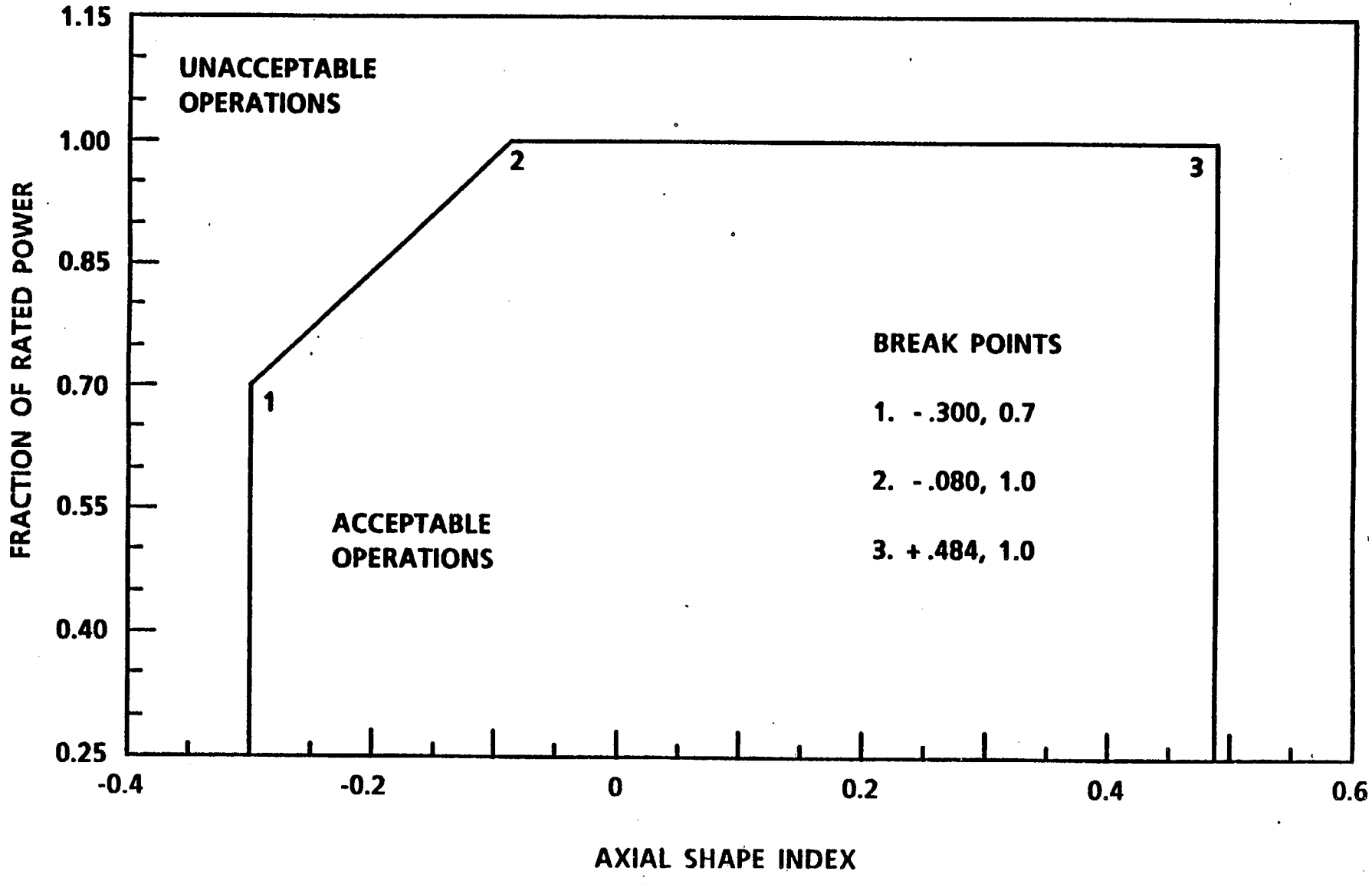
The requirement that the steam generator temperature be $\leq$ the PCS temperature when forced circulation is initiated in the PCS ensures that an energy addition caused by heat transferred from the secondary system to the PCS will not occur. This requirement applies only to the initiation of forced circulation (the start of the first primary coolant pump) when the PCS cold leg temperature is < 450°F. At or above 450°F, the PCS safety valves prevent the PCS pressure from exceeding 10CFR50 Appendix G limits.

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(P), Volume 2, Section 15.0.7.1 /
(5) (Deleted)

(6) ANF-88-108 /

FIGURE 3-0
ASI LCO FOR Tinlet FUNCTION



3.1.7 Primary and Secondary Safety Valves

Specifications

a. The reactor shall not be made critical unless all three pressurizer safety valves are operable with their lift settings maintained between 2500 psia and 2580 psia (± 1%).

b. A minimum of one operable safety valve shall be installed on the pressurizer whenever the reactor head is on the vessel.

c. Whenever the reactor is in power operation, a minimum of 23 secondary system safety valves shall be operable with their lift settings between 985 psig (± 30 psig) and 1025 (± 3%) psig.

Basis

The primary and secondary safety valves pass sufficient steam to limit the primary system pressure to 110 percent of design (2750 psia) following a complete loss of turbine generator load without simultaneous reactor trip while operating at 2650 MW_t.[1]

The reactor is assumed to trip on a "High Primary Coolant System Pressure" signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the secondary system safety valves. Conservative values for all system parameters, delay times and core moderator coefficient are assumed. Overpressure protection is provided to the portions of the primary coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of one valve's capacity. One valve, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

The total relief capacity of the 24 secondary system safety valves is 11.7×10^6 lb/h. This is based on a steam flow equivalent to an NSSS power level of 2650 MW_t at the nominal 1000 psia valve lift pressure.

At the power rating of 2530 MW_t, a relief capacity of less than 11.2×10^6 lb/h is required to prevent overpressurization of the secondary system of loss of load conditions, and 23 valves provide relieving capability of 11.2×10^6 lb/h.[1,2]

The overpressurization analysis for the loss of load event[2] supports the specified secondary safety valve lift pressure tolerance. ASME B&PV Code, 1986 edition, Section XI, subsection IWV-3500, specifies ANSI/ASME OM-1-1981 requirements which allow the specified tolerances in the lift pressures of the safety valves.

References

(1) Updated FSAR, Section 4.3.4 and 4.3.9.4
(2) ANF-87-150(NP), Volume 2, Section 15.2.1

BASIS

The Steam and Power Conversion System is designed to receive steam from the NSSS and convert the steam thermal energy into electrical energy. A closed regenerative cycle condenses the steam from the main turbine and returns the condensate as heated feedwater to the steam generators. Normally, the capability to supply feedwater to the steam generators is provided by operation of the turbine-driven main feedwater pumps.

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser, or by steam discharge to the atmosphere via the main steam safety valves or power operated relief valves.[1,2] If the main feedwater pumps are not operating, any one auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the Plant. The Plant is provided with two motor driven auxiliary feedwater pumps (P-8A, P-8C) and one turbine driven auxiliary feedwater pump (P-8B). The Auxiliary Feedwater System is designed so that an automatic start signal is generated to the auxiliary feedwater pumps upon low secondary side steam generator level. Upon low secondary side steam generator level, auxiliary feedwater pump P-8A would be the first auxiliary feedwater pump to receive an automatic start signal. If pump P-8A failed to start or establish flow within a specified period of time, auxiliary feedwater pump P-8C would receive an automatic start signal. If both pump P-8A and pump P-8C failed to start or establish flow within each pump's specified period of time, auxiliary feedwater pump P-8B would receive an automatic start signal. All three auxiliary feedwater pumps normally take suction from the condensate storage tank. The minimum amount of water in the condensate storage tank and primary coolant system makeup tanks combined is the amount needed for 8 hours of auxiliary feedwater pump operation. If the outage is more than 8 hours, Lake Michigan water can be used, by utilizing a fire pump to supply water to the auxiliary feedwater pumps P-8A and P-8B, or by utilizing a service water pump to supply water to auxiliary feedwater pump P-8C.

Three fire pumps are provided, one motor driven and two diesel driven, each capable of delivering 1500 gpm at 125 psig. Three service water pumps are provided, all of which are motor driven, each capable of delivering 8000 gpm at 60 psig.

REFERENCES

(1) Updated FSAR, Section 10.2.1
(2) ANF-87-150(P), Volume 2, Section 15.2.7

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to operation of control rods and hot channel factors during operation.

Objective

To specify limits of control rod movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.

Specifications

3.10.1 Shutdown Margin Requirements

a. With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 2%.

b. With less than four primary coolant pumps in operation at hot shutdown and above, boration shall be immediately initiated to increase and maintain the shutdown margin at $\geq$ 3.75%.

c. At less than the hot shutdown condition, with at least one primary coolant pump in operation or at least one shutdown cooling pump in operation, with a flow rate $\geq$ 2810 gpm, the boron concentration shall be greater than the cold shutdown boron concentration for normal cooldowns and heatups, ie, nonemergency conditions. /

During nonemergency conditions, at less than the hot shutdown condition with no operating primary coolant pumps and a primary system recirculating flow rate < 2810 gpm but $\geq$ 650 gpm, then within one hour either: /

1. (a) Establish a shutdown margin of $\geq$ 3.5% and /
   (b) assure two of the three charging pumps are electrically disabled. /

OR /

2. At least every 15 minutes verify that no charging pumps are operating. If one or more charging pumps are determined to be operating in any 15 minute surveillance period, terminate charging pump operation and insure that the shutdown margin requirements are met and maintained. /

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Continued) /

3.10.1 Shutdown Margin Requirements (Continued) /

During nonemergency conditions, at less than the hot shutdown condition with no operating primary coolant pumps and a primary system recirculating flow rate less than 650 gpm, within one hour: / / / /

(a) Initiate surveillance at least every 15 minutes to verify that no charging pumps are operating. If one or more charging pumps are determined to be operating in any 15-minute surveillance period, terminate charging pump operation and insure that the shutdown margin requirements are met and maintained. / / / / / /

d. If a control rod cannot be tripped, shutdown margin shall be increased by boration as necessary to compensate for the worth of the withdrawn inoperable rod.

e. The drop time of each control rod shall be no greater than 2.5 seconds from the beginning of rod motion to 90% insertion.

3.10.2 (Deleted) /

3.10.3 Part-Length Control Rods

The part-length control rods will be completely withdrawn from the core (except for control rod exercises and physics tests).

3.10.6 Shutdown Rod Limits

a. All shutdown rods shall be withdrawn before any regulating rods are withdrawn.

b. The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer.

c. The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted.

3.10.7 Low Power Physics Testing

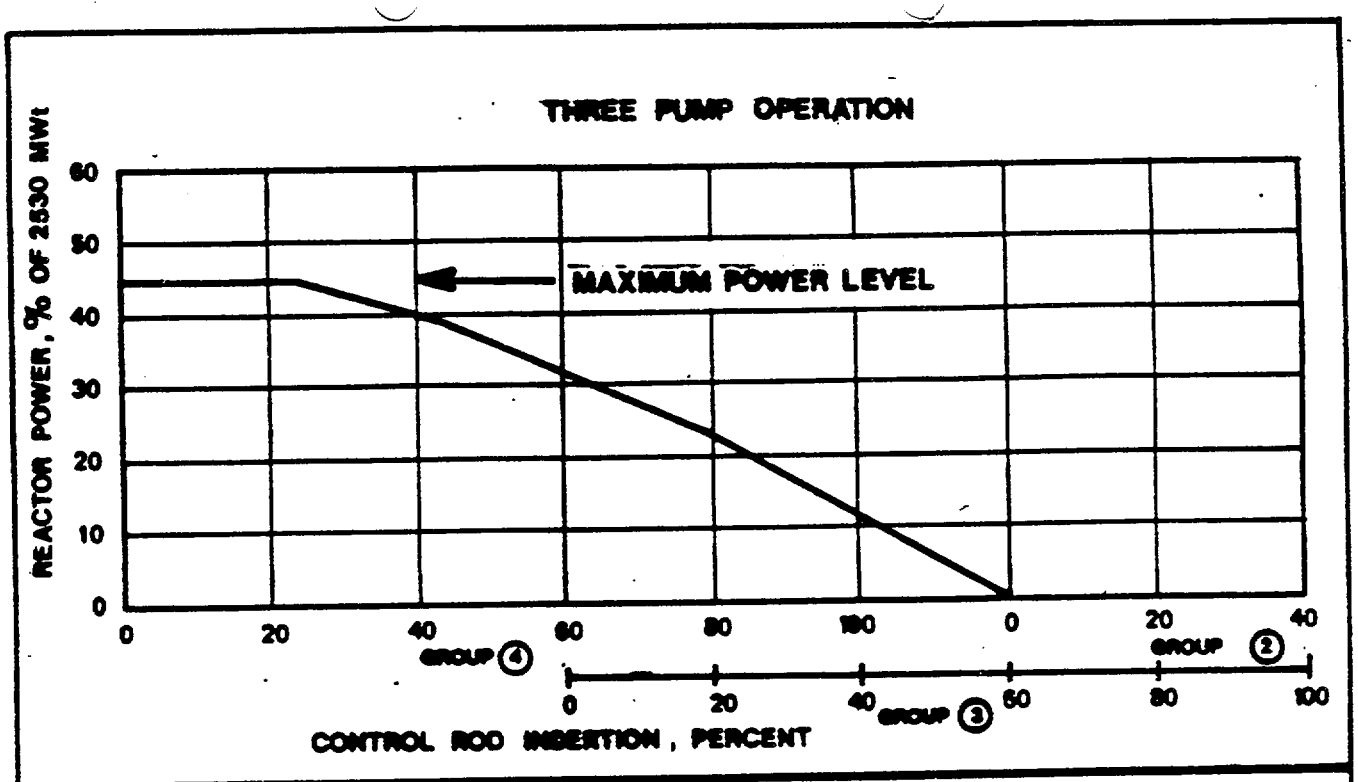
Sections 3.10.1.a, 3.10.1.b, 3.10.3, 3.10.4.b, 3.10.5 and 3.10.6 may be deviated from during low power physics testing and CRDM exercises if necessary to perform a test but only for the time necessary to perform the test. /

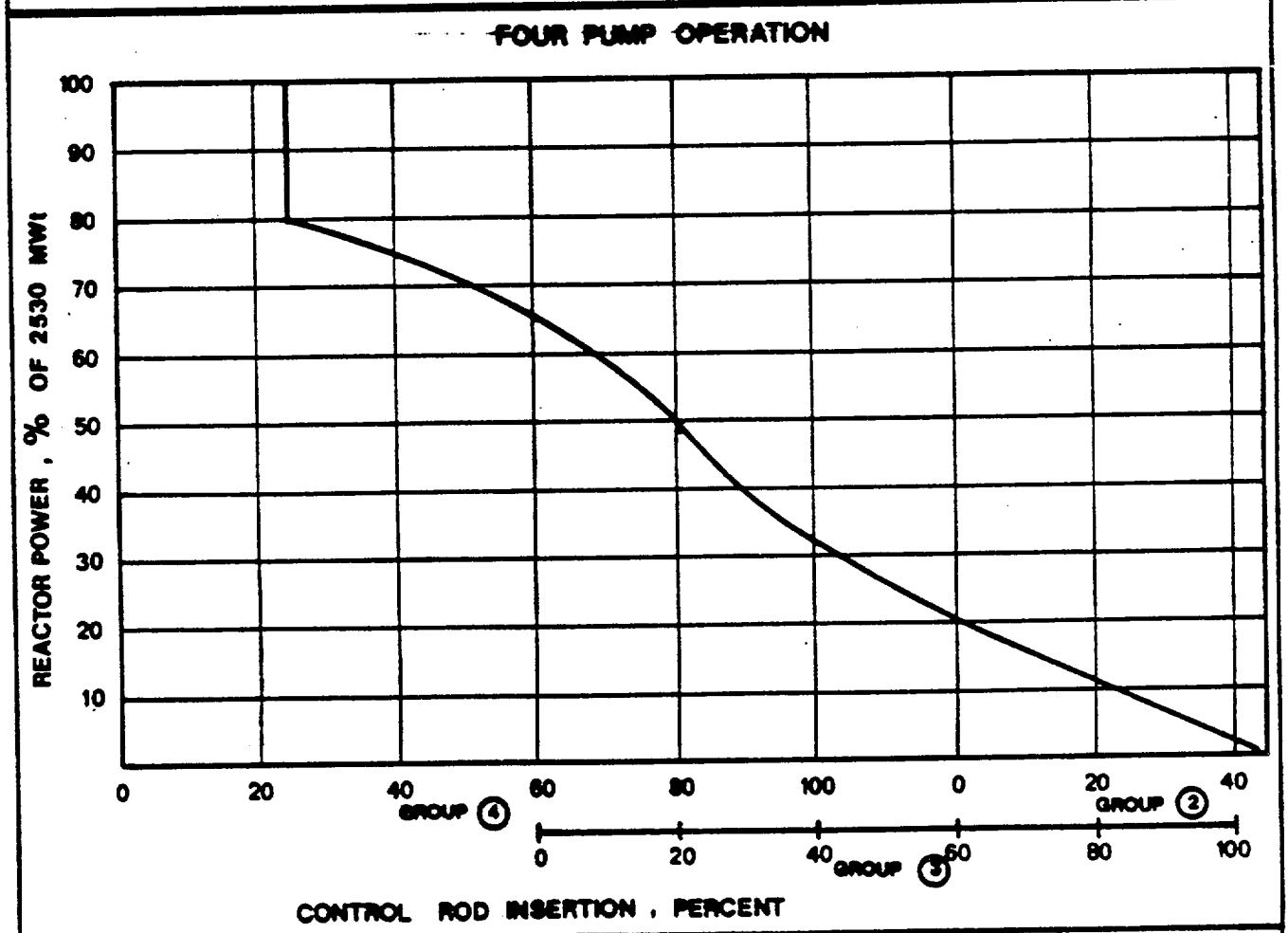
3.10.8 Center Control Rod Misalignment

The requirements of Specifications 3.10.4.1, 3.10.4.a, and 3.10.5 may be suspended during the performance of physics tests to determine the isothermal temperature coefficient and power coefficient provided that only the center control rod is misaligned and the limits of Specification 3.23 are maintained.

Basis

Sufficient control rods shall be withdrawn at all times to assure that the reactivity decrease from a reactor trip provides adequate shutdown margin. The available worth of withdrawn rods must include the reactivity defect of power and the failure of the withdrawn rod of highest worth to insert. The requirement for a shutdown margin of 2.0% in reactivity with 4-pump operation, and of 3.75% in reactivity with less than 4-pump operation, is consistent with the assumptions used in the analysis of accident conditions (including steam line break) as reported in Reference 1 and 2 and additional analysis. Requiring the boron concentration to be at cold shutdown boron concentration at less than hot shutdown assures adequate shutdown margin exists to ensure a return to power does not occur if an unanticipated cooldown accident occurs. This requirement applies to normal operating situations and not during emergency conditions where it is necessary to perform operations to mitigate the consequences of an accident. By imposing a minimum shutdown cooling pump flow rate of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under asymmetric conditions. For operation with no primary coolant pumps operating and a recirculating flow rate less than 2810 gpm the increased shutdown margin and controls on charging pump operability or alternately the surveillance of the charging pumps will ensure that the acceptance criteria for an inadvertent boron dilution event will not be violated.[3] The change in insertion limit with reactor power shown on Figure 3-6 insures that the shutdown





CONTROL ROD INSERTION LIMITS	PALISADES TECHNICAL SPECIFICATION	FIGURE 3-6

Amendment No ~~37~~, 118

Basis (Continued)

margin requirements for 4-pump operation is met at all power levels. The 2.5-second drop time specified for the control rods is the drop time used in the transient analysis.[2]

The insertion of part-length rods into the core, except for rod exercises or physics tests, is not permitted since it has been demonstrated on other CE plants that design power distribution envelopes can, under some circumstances, be violated by using part-length rods. Further information may justify their use. Part-length rod insertion is permitted for physics tests, since resulting power distributions are closely monitored under test conditions. Part-length rod insertion for rod exercises (approximately 6 inches) is permitted since this amount of insertion has an insignificant effect on power distribution.

For a control rod misaligned up to 8 inches from the remainder of the banks, hot channel factors will be well within design limits. If a control rod is misaligned by more than 8 inches, the maximum reactor power will be reduced so that hot channel factors, shutdown margin and ejected rod worth limits are met. If in-core detectors are not available to measure power distribution and rod misalignments >8 inches exist, then reactor power must not exceed 75% of rated power to insure that hot channel conditions are met.

Continued operation with that rod fully inserted will only be permitted if the hot channel factors, shutdown margin and ejected rod worth limits are satisfied.

In the event a withdrawn control rod cannot be tripped, shutdown margin requirements will be maintained by increasing the boron concentration by an amount equivalent in reactivity to that control rod. The deviations permitted by Specification 3.10.7 are required in order that the control rod worth values used in the reactor physics calculations, the plant safety analysis, and the Technical Specifications can be verified. These deviations will only be in effect for the time period required for the test being performed. The testing interval during which these deviations will be in effect will be kept to a minimum and special operating precautions will be in effect during these deviations in accordance with approved written testing procedures.

Basis (Continued) /

Violation of the power dependent insertion limits, when it is necessary to rapidly reduce power to avoid or minimize a situation harmful to plant personnel or equipment, is acceptable due to the brief period of time that such a violation would be expected to exist, and due to the fact that it is unlikely that core operating limits such as thermal margin and shutdown margin would be violated as a result of the rapid rod insertion. Core thermal margin will actually increase as a result of the rapid rod insertion. In addition, the required shutdown margin will most likely not be violated as a result of the rapid rod insertion because present power dependent insertion limits result in shutdown margin in excess of that required by the safety analysis. /

References

(1) XN-NF-77-18 /
(2) ANF-87-150(NP), Volume 2 /
(3) ANF-88-108 /

POWER DISTRIBUTION INSTRUMENTATION

3.11.2 EXCORE POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

The excore monitoring system shall be operable with:

a. The target Axial Offset (AO) and the Excore Monitoring Allowable Power Level (APL) determined within the previous 31 days using the incore detectors, and the measured AO not deviated from the target AO by more than 0.05 in the previous 24 hours.

b. The AO measured by the excore detectors calibrated with the AO measured by the incore detectors.

c. The quadrant tilt measured by the excore detectors calibrated with the quadrant tilt measured by the incore detectors.

APPLICABILITY:

(1) Items a., b. and c. above are applicable when the excore detectors are used for monitoring LHR.

(2) Item c. above is applicable when the excore detectors are used for monitoring quadrant tilt.

(3) Item b., above is applicable for each channel of the TM/LP trip and the Axial Shape Index (ASI) alarm. /

ACTION 1:

With the excore monitoring system inoperable, do not use the system for monitoring LHR.

ACTION 2:

If the measured quadrant tilt has not been calibrated with the incores, do not use the system for monitoring quadrant tilt.

ACTION 3: /

When the measured AO uncertainty is greater than specified in Specification 4.18.2, the TM/LP trip function and the ASI alarm setpoints shall be conservatively adjusted within twelve (12) hours or that channel shall be declared inoperable. The operability requirements for TM/LP and ASI are given in Table 3.17.1 and 3.17.4, respectively. / / / / /

Basis

The excore power distribution monitoring system consists of Power Range Detector Channels 5 through 8.

The operability of the excore monitoring system ensures that the assumptions employed in the PDC-II analysis[1] for determining AO limits that ensure operation within allowable LHR limits are valid.

3.11.2 EXCORE POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

Basis (Contd)

Surveillance requirements ensure that the instruments are calibrated to agree with the incore measurements and that the target AO is based on the current operating conditions. Updating the Excore Monitoring APL ensures that the core LHR limits are protected within the ±0.05 band on AO. The APL considers LOCA based LHR limits, and factors are included to account for changes in radial power shape and LHR limits over the calibration interval. /

The APL is determined from the following:

$$APL = \left[\frac{LHR(Z)_{TS}}{LHR(Z)_{Max} \times V(Z) \times 1.02} \right]_{Min} \times \text{Rated Power} \quad (2)$$ /

Where:

(1) $LHR(Z)_{TS}$ is the limiting LHR vs Core Height (from Section 3.23.1),

(2) $LHR(Z)_{Max}$ is the measured peak LHR including uncertainties vs Core Height,

(3) V(Z) is the function (shown in Figure 3.11-1),

(4) The factor of 1.02 is an allowance for the effects of upburn, /

(5) The quantity in brackets is the minimum value for the entire core at any elevation (excluding the top and bottom 10% of core) considering limits for peak rods. If the quantity in brackets is greater than one, the APL shall be the rated power level. / /

References /

(1) XN-NF-80-47
(2) ANF-88-107 /

3.12 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to the moderator temperature coefficient of reactivity for the core.

Objective

To specify a limit for the positive moderator coefficient.

Specifications

The moderator temperature coefficient (MTC) shall be less positive than $+0.5 \times 10^{-4}$ $\Delta\rho/°F$ at $\leq$ 2% of rated power. /

Bases

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the safety analysis [1] remain valid. /

Reference

(1)ANF-87-150(P), Volume 2, Section 15.0.5 /

If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change power, variable high power and high pressurizer pressure),[1] which results in a one-out-of-three channel logic. If, in the 2 of 4 logic system of either the reactor protective system or the engineered safeguards system, one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1 of 2. At rated power, the minimum operable variable high power level channels is 3 in order to provide adequate flux tilt detection. If only 2 channels are operable, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts.

The engineered safeguards system provides a 2 out of 4 logic on the signal used to actuate the equipment connected to each of the 2 emergency diesel generator units.

Two start-up channels are available any time reactivity changes are deliberately being introduced into the reactor and the neutron power is not visible on the log-range nuclear instrumentation or above 10^{-4}% of rated power. This ensures that redundant start-up instrumentation is available to operators to monitor effects of reactivity changes when neutron power levels are only visible on the start-up channels. In the event only one start-up range channel is available and the neutron power level is sufficiently high that it is being monitored by both channels of log-range instrumentation, a startup can be performed in accordance with footnote (d) of Table 3.17.4.

The Zero Power Mode Bypass can be used to bypass the low flow, steam generator low pressure, and TM/LP trips[2] for all four Reactor Protective system channels to perform control rod testing or to perform low power physics testing below normal operating temperatures. The requirement to maintain cold shutdown boron concentration when in the bypass condition provides additional assurance that an accidental criticality will not occur. To allow low power physics testing at reduced temperature and pressure, the requirement for cold shutdown boron concentration is not required and the allowed power is increased to 10^{-1}%.

References

(1) Updated FSAR, Section 7.2.7.
(2) Updated FSAR, Section 7.2.5.2

Table 3.17.1
Instrumentation Operating Requirements for Reactor Protective System

No	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
1	Manual (Trip Buttons)	1	None	None
2	Variable High Power Level	2[b,d]	1[d]	None
3	Log Range Channels	2	1	Below 10^{-4}%[e] or Above 15% Rated Power[a] Except as Noted in (c)
4	Thermal Margin/ Low-Pressurizer Pressure	2[b,f]	1	Below 10^{-4}%[e] of Rated Power[a] and greater than cold shutdown boron concentration.
5	High-Pressurizer Pressure	2[b]	1	None
6	Low Flow Loop	2[b]	1	Below 10^{-4}%[e] of Rated Power[a] and greater than cold shutdown boron concentration.
7	Loss of Load	1	None	None
8	Low Steam Generator Water Level	2/Steam Gen[b]	1/Steam Generator	None
9	Low Steam Generator Pressure	2/Steam Gen[b]	1/Steam Generator	Below 10^{-4}%[e] of Rated Power[a] and greater than cold shutdown boron concentration.
10	High Containment Pressure	2[b]	1	None

(a) Bypass automatically removed.
(b) One of the inoperable channels must be in the tripped condition.
(c) Two channels required if TM/LP, low steam generator or low-flow channels are bypassed.
(d) If only two channels are operable, load shall be reduced to 70% or less of rated power.
(e) For low power physics testing, 10^{-4}% may be increased to 10^{-1}% and cold shutdown boron concentration is not required.
(f) AO operability requirements are given in Specification 3.11.2.

Table 3.17.4 (Cont'd)

No	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
8.	Pressurizer Wide Range Water Level Indication	2 (l, p, q)	None	Not required in Cold or Refueling Shutdown
9.	Pressurizer Code Safety Relief Valves Position Indication (Acoustic Monitor or Temperature Indication)	1 per Valve	None	Not Required below 325°F
10.	Power Operated Relief Valves (Acoustic Monitor or Temperature Indication)	1 per Valve	None	Not required when PORV isolation valve is closed and its indication system is operable
11.	PORV Isolation Valves Position Indication	1 per Valve	None	Not required when reactor is depressurized and vented through a vent ≥1.3 sq.in.
12.	Subcooling Margin Monitor	1	None	Not required below 515°F
13.	Auxiliary Feed Flow Rate Indication	1 per flow Control Valve[h]	None	Not required below 325°F
14.	Auxiliary Feedwater Actuation System Sensor Channels	2 per steam generator[e]	1	Not required below 325°F
15.	Auxiliary Feedwater Actuation System Actuation Channels	2[f]	1	Not required below 325°F
16.	Excore Detector Deviation Alarms	1[g]	None	Not Required Below 25% of Rated Power / /
17.	Axial Shape Index Alarm	2[i]	1	Not Required Below 25% of Rated Power / /

(e) Auxiliary Feedwater System Actuation System Sensor Channels contain pump auto initiation circuitry. If two sensor channels for one steam generator are inoperable, one of the steam generator low level bistable modules in one of the inoperable channels must be in the tripped condition.

(f) With one Auxiliary Feedwater Actuation System Actuation Channel inoperable, in lieu of the requirement of 3.17.2, provide a second licensed operator in the control room within 2 hours. With both inoperable, in lieu of following the requirements of 3.17.2, start and maintain in operation the turbine driven auxiliary feed pump.

(g) Calculate the Quadrant Power Tilt using the excore readings at least once per 8 hours when the excore detectors deviation alarms are inoperable, or at least once per 8 hours using symmetric incore detectors when the difference between the excore and the incore measured Quadrant Power Tilt exceeds 2%. /

(h) With two flow rate indicators inoperable for a given control valve, the control valve shall be considered inoperable and the requirements of 3.5.2(e) apply.

(i) AO operability requirements are given in Specification 3.11.2. /

(j, k) Blank /

(l) The provisions of Specification 3.0.4 are not applicable.

(m, n, o) Blank

(p) With one OPERABLE Pressurizer Wide Range Water Level Channel in lieu of the requirement of 3.17.2, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

(q) With no OPERABLE Pressurizer Wide Range Water Level Channels in lieu of the requirements of 3.17.2, either restore at least one of the inoperable channels to OPERABLE status within 48 hours, or be in at least HOT SHUTDOWN within the next 12 hours.

(next page is 3-82)

3.23 POWER DISTRIBUTION LIMITS

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

The LHR in the peak power fuel rod at the peak power elevation Z shall not exceed the value in Table 3.23-1 times $F_A(Z)$ [the function $F_A(Z)$ is shown in Figure 3.23-1].

APPLICABILITY: Power operation above 50% of rated power.

ACTION 1:

When using the incore alarm system to monitor LHR, and with four or more coincident incore alarms, initiate within 15 minutes corrective action to reduce the LHR to within the limits and restore the incore readings to less than the alarm setpoints within 1 hour or failing this, be at less than 50% of rated power within the following 2 hours.

ACTION 2:

When using the excore monitoring system to monitor LHR and with the AO deviating from the target AO by more than 0.05, discontinue using the excore monitoring system for monitoring LHR. If the incore alarm system is inoperable, within 2 hours be at 85% (or less) of rated thermal power and follow the procedure in ACTION 3 below.

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

ACTION 3:

If the incore alarm system is inoperable and the excore monitoring system is not being used to monitor LHR, operation at less than or equal to 85% of rated power may continue provided that incore readings are recorded manually. Readings shall be taken on a minimum of 10 individual detectors per quadrant (to include 50% of the total number of detectors in a 10-hour period) within 4 hours and at least every 2 hours thereafter. If readings indicate a local power level equal to or greater than the alarm setpoints, the action specified in ACTION 1 above shall be taken.

Basis

The limitation of LHR ensures that, in the event of a LOCA, the peak temperature of the cladding will not exceed 2200°F.[1]

Either of the two core power distribution monitoring systems (the incore alarm system or the excore monitoring system) provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its limits. The incore alarm system performs this function by continuously monitoring the local power at many points throughout the core and comparing the measurements to predetermined setpoints above which the limit on LHR could be exceeded. The excore monitoring system performs this function by providing comparison of the measured core AO with predetermined AO limits based on incore measurements. An Excore Monitoring Allowable Power Level (APL), which may be less than rated power, is applied when using the excore monitoring system to ensure that the AO limits adequately restrict the LHR to less than the limiting values.[4]

If the incore alarm system and the excore monitoring system are both inoperable, power will be reduced to provide margin between the actual peak LHR and the LHR limits and the incore readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the manpower available, the reactor power will be reduced to a point below which it is improbable that the LHR limits could be exceeded.

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

Basis (Contd)

The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the monitoring systems are returned to service.

To ensure that the design margin of safety is maintained, the determination of both the incore alarm setpoints and the APL takes into account a measurement uncertainty factor of 1.10, an engineering uncertainty factor of 1.03, a thermal power measurement uncertainty factor of 1.02 and allowance for quadrant tilt.

References

(1) ANF-88-107 /

(2) (Deleted) /

(3) (Deleted) /

(4) XN-NF-80-47 /

TABLE 3.23-1

LINEAR HEAT RATE LIMITS

	No. of Fuel Rods in Assembly		
	208	216	
Peak Rod	15.28 kW/ft	15.28 kW/ft	/

TABLE 3.23-2

RADIAL PEAKING FACTOR LIMITS, F_L

Peaking Factor	No. of Fuel Rods in Assembly		
	208	216	
Assembly F_r^A	1.48	1.50	/
Interior Rod $F_r^{\Delta H}$	1.70	1.73	/

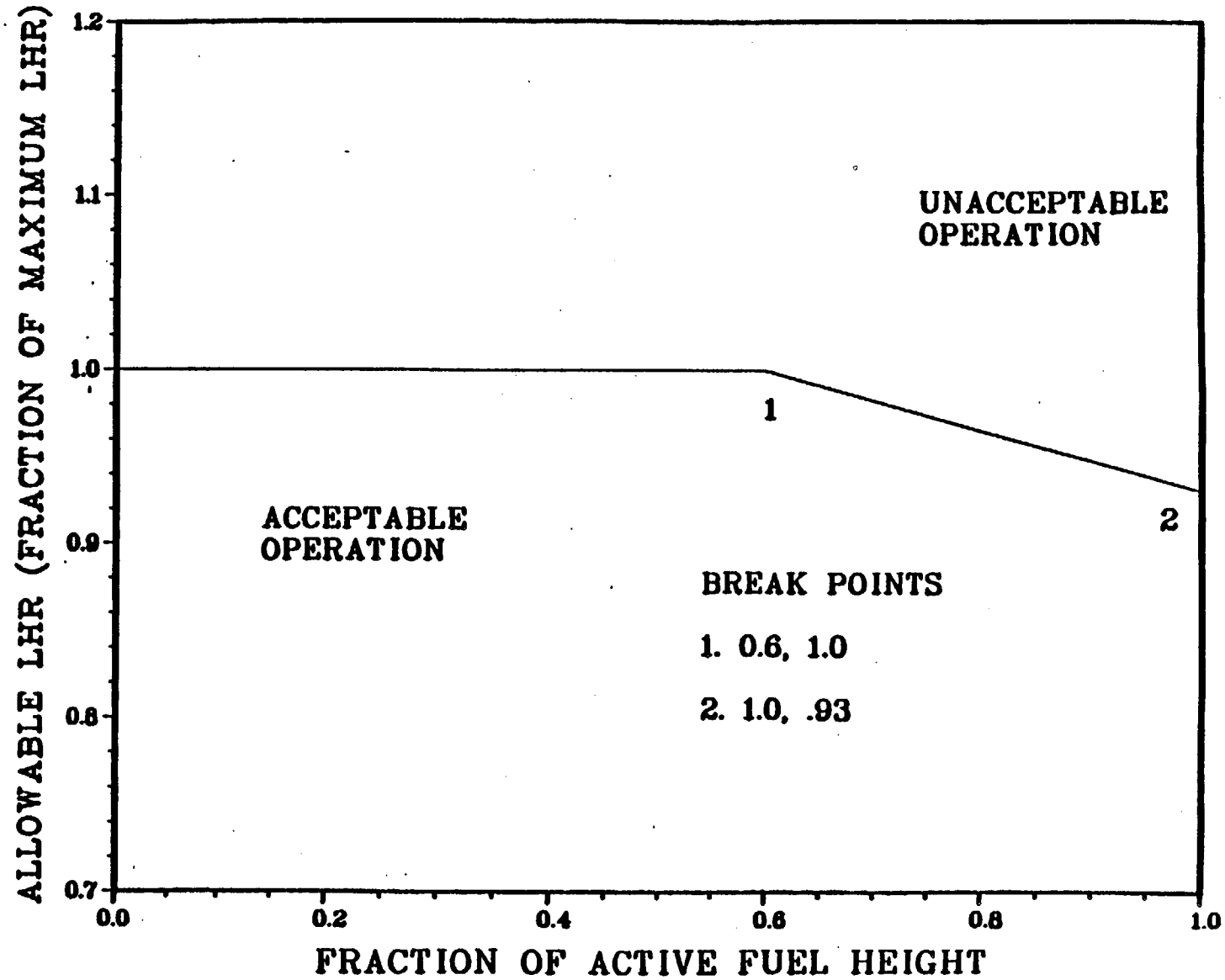


FIGURE 3.23-1 ALLOWABLE LHR AS A FUNCTION OF PEAK POWER LOCATION

FIGURE 3.23-2

(Deleted)

FIGURE 3.23-3

(Deleted)

3.23.2 RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

The radial peaking factors F_r^A, and $F_r^{\Delta H}$ shall be less than or equal to the value in Table 3.23-2 times the following quantity. The quantity is [1.0 + 0.3 (1 - P)] for P $\geq$ .5 and the quantitiy is 1.15 for P < .5. P is the core thermal power in fraction of rated power.

APPLICABILITY: Power operation above 25% of rated power.

ACTION:

1. For P < 50% of rated with any radial peaking factor exceeding its limit, be in at least hot shutdown within 6 hours.

2. For P $\geq$ 50% of rated with any radial peaking factor exceeding its limit, reduce thermal power within 6 hours to less than the lowest value of:

$$[1 - 3.33 \left(\frac{F_r}{F_L} - 1\right)] \times \text{Rated Power}$$

Where F_r is the measured value of either F_r^A, or $F_r^{\Delta H}$ and F_L is the corresponding limit from Table 3.23-2.

Basis

The limitations on F_r^A, and $F_r^{\Delta H}$ are provided to ensure that assumptions used in the analysis for establishing DNB margin, LHR and the thermal margin/low-pressure and variable high-power trip set points remain valid during operation. Data from the incore detectors are used for determining the measured radial peaking factors. The periodic surveillance requirements for determining the measured radial peaking factors provide assurance that they remain within prescribed limits. Determining the measured radial peaking factors after each fuel loading prior to exceeding 50% of rated power provides additional assurance that the core is properly loaded.

3.23.3 QUADRANT POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

The quadrant power tilt (T_q) shall not exceed 5%.

APPLICABILITY: Power operation above 25% of rated power. /

ACTION:

1. With quadrant power tilt determined to exceed 5% but less than or equal to 10%. /

   a. Correct the quadrant power tilt within 2 hours after exceeding the limit, or /

   b. Determine within the next 2 hours and, at least once every 8 hours thereafter, that the radial peaking factors are within the limits of Section 3.23.2, or /

   c. Reduce power, at the normal shutdown rate, to less than 85% of rated power and determine that the radial peaking factors are within the limits of Section 3.23.2. At reduced power, determine at least once every 8 hours that the radial peaking factors are within the limits of Section 3.23.2. /

2. With quadrant power tilt determined to exceed 10%: /

   a. Correct the quadrant power tilt within 2 hours after exceeding the limit, or /

   b. Reduce power to less than 50% of rated power within the next 2 hours and determine that the radial peaking factors are within the limits of Section 3.23.2. At reduced power, determine at least once every 8 hours that the radial peaking factors are within the limits of Section 3.23.2. /

3. With the quadrant power tilt determined to exceed 15%, be in at least hot standby within 12 hours.

Basis

Limitations on quadrant power tilt are provided to ensure that design safety margins are maintained. Quadrant power tilt is determined from excore detector readings which are calibrated using incore detector measurements.[1] Calibration factors are determined from incore measurements by performing a two-dimensional, full-core surface fit of deviations between measured and theoretical incore readings and integrating the fitting function over each core quadrant. Values of LHR and radial peaking factors are increased by the value of quadrant tilt.

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 pilot valves are used as an alternative to venting the PCS, then verify both PORV pilot 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 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 bona fide signal.

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System(5)

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Power Range Safety Channels	a. Check	S	a. Comparison of four-power channel readings.
	b. Check(3)	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparator alarms.
	c. Test	M(2)	c. Internal test signal. /
	d. Calibrate (6)	R	d. Channel alignment through measurement/adjustment of internal test points. /
2. Wide-Range Logarithmic Neutron Monitors	a. Check	S	a. Comparison of both wide-range readings.
	b. Test	P	b. Internal test signal.
3. Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.
	b. Calibrate	R	b. Known differential pressure applied to sensors. /
	c. Test	M(2)	c. Bistable trip tester.(1)(4)
4. Thermal Margin/Low Pressurizer Pressure	a. Check:	S	a. Check:
	(1) Temperature Input		(1) Comparison of four separate calculated trip pressure set point indications.
	(2) Pressure Input		(2) Comparison of four pressurizer pressure indications. (Same as 5(a) below.)
	b. Calibrate	R	b. Calibrate:
	(1) Temperature Input		(1) Known resistance substituted for RTD coincident with known pressure and power input. /
	(2) Pressure Input		(2) Part of 5(b) below.
	c. Test	M(2)	c. Bistable trip tester.(1) /
5. High-Pressurizer Pressure	a. Check	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors. /
	c. Test	M(2)	c. Bistable trip tester.(1)

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System(5) (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
6. Steam Generator Level	a. Check	S	a. Comparison of four level indications per generator.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)
7. Steam Generator Pressure	a. Check	S	a. Comparisons of four pressure indications per generator.
	b. Calibrate	R	b. Known pressure applied to sensors. /
	c. Test	M(2)	c. Bistable trip tester.(1)
8. Containment Pressure	a. Calibrate	R	a. Known pressure applied to sensors.
	b. Test	M(2)	b. Simulate pressure switch action.
9. Loss of Load	a. Test	P	a. Manually trip turbine auto stop oil relays.
10. Manual Trips	a. Test	P	a. Manually test both circuits.
11. Reactor Protection System Logic Units	a. Test	M(2)	a. Internal test circuits.
12. Axial Shape Index (ASI)	a. Test	R	a. Known power inputs applied to Thermal Margin Calculator. / /
13. ΔT Power	a. Check	S	a. Same as 1(a). /
	b. Check (3)	D	b. Same as 1(b). /
	c. Test	R	c. Known temperature imputs applied to Thermal Margin Calculator. / /

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System(5) (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
14. Thermal Margin Calculator	a. Check	Q	a. Verify constants.

NOTES: (1)The bistable trip tester injects a signal into the bistable and provides a precision readout of the trip set point.

(2)All monthly tests will be done on only one of four channels at a time to prevent reactor trip.

(3)Adjust the nuclear power or ΔT power until readout agrees with heat balance calculations when above 15% of rated power.

(4)Trip setting for operating pump combination only. Settings for other than operating pump combinations must be tested during routine monthly testing performed when shut down and within four hours after resuming operation with a different pump combination if the setting for that combination has not been tested within the previous month.

(5)It is not necessary to perform the specified testing during prolonged periods in the refueling shutdown condition. If this occurs, omitted testing will be performed prior to returning the plant to service.

(6)Also includes testing variable high power function in the Thermal Margin Calculator.

FREQUENCY NOTATION

Notation	Frequency
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
P	Prior to each start-up if not done previous week.
NA	Not applicable.

TABLE 1.3

Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Start-Up Range Neutron Monitors	a. Check	S	a. Comparison of both channel count rate indications when in service.
	b. Test	P	b. Internal test signals.
2. Primary Rod Position Indication System	a. Check	S	a. Comparison of output data with secondary RPIS.
	b. Check	M	b. Check of power dependent insertion limits monitoring system.
	c. Calibrate	R	c. Physically measured rod drive position used to verify system accuracy. Check rod position interlocks.
3. Secondary Rod Position Indication System	a. Check	S	a. Comparison of output data with primary RPIS.
	b. Check	M	b. Same as 2(b) above.
	c. Calibrate	R	c. Same as 2(c) above, including out-of-sequence alarm function.
4. Area Monitors Note: Process Monitor Surveillance Requirements are located in Tables 4.24-1 and 4.24-2	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	M	c. Detector exposed to remote operated radiation check source or integral electronic check source.
5. Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
	b. Test	M	b. Battery check.
6. Environmental Monitors	a. Check	M	a. Operational check.
	b. Calibrate	A	b. Verify airflow indicator.
7. Pressurizer Level Instruments	a. Check	S	a. Comparison of two wide and two narrow range independent level readings.
	b. Calibrate	R	b. Known differential pressure applied to sensor.
	c. Test	M	c. Signal to meter relay adjusted with test device.

TABLE 4.1.3

Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls (Continued)

Channel Description	Surveillance Function	Frequency	Surveillance Method
8. Control Rod Drive System Interlocks	a. Test	R	a. Verify proper operation of all manual rod drive control system interlocks, using simulated signals where necessary. /
	b. Test	P	b. Same as 8(a) above, if not done within three months.
9. Flux-ΔT Power Comparator	a. Calibrate	R	a. Use simulated signals. /
	b. Test	M	b. Use simulated signals. /
10. Calorimetric Instrumentation	a. Calibrate	R	a. Known differential pressure applied to feedwater flow sensors. /
11. Containment Building Humidity Detectors	a. Test	R	a. Expose sensor to high humidity atmosphere.
12. Interlocks - Isolation Valves on shutdown Cooling Line	a. Calibrate	R	a. Known pressure applied to sensor.
13. Service Water Break Detector in Containment	a. Test	R	a. Known differential pressure applied to Sensors.

/
/
/
/

4.15 Primary System Flow Measurement

Applicability

Applies to the measurement of primary system flow rate with four primary coolant pumps in operation.

Objective

To provide assurance that the primary system flow rate is equal to or above the flow rate required in 3.1.1.c.

Specification

After each refueling outage, or after plugging 10 or more steam generator tubes, a primary system flow measurement shall be made with four primary coolant pumps in operation. This measurement shall be made within the first 31 days of rated power operation. / /

Basis

This surveillance program assures that the reactor coolant flow is consistent with that assumed as the basis for Specification 3.1.1c.

4.18.2 EXCORE MONITORING SYSTEM

SURVEILLANCE REQUIREMENTS

4.18.2.1 At least every 31 days of power operation:

a. A target AO and excore monitoring allowable power level shall be determined using excore and incore detector readings at steady state near equilibrium conditions.

b. Individual excore channel measured AO shall be compared to the total core AO measured by the incores. If the difference is greater than 0.02, the excore monitoring system shall be recalibrated. / /

c. The excore measured Quadrant Power Tilt shall be compared to the incore measured Quadrant Power Tilt. If the difference is greater than 2%, the excore monitoring system shall be recalibrated.

4.19 POWER DISTRIBUTION LIMITS

4.19.1 LINEAR HEAT RATES

SURVEILLANCE REQUIREMENTS

4.19.1.1 When using the incore alarm system to monitor LHR, prior to operation above 50% of rated power and every 7 days of power operation thereafter, incore alarms shall be set based on a measured power distribution.

4.19.1.2 When using the excore monitoring system to monitor LHR:

a. Prior to use, verify that the measured AO has not deviated from the target AO by more than 0.05 in the previous 24 hours for each operable channel using the previous 24 hourly recorded values. / / /

b. Once per day, verify that the measured Quadrant Power Tilt is less than or equal to 3%.

c. Once per hour, verify that the power is less than or equal to the APL and not more than 10% of rated power greater than the power level used in determining the APL.

d. Continuously verify that the measured AO is within 0.05 of the established target AO for at least 3 of the 4, 2 of the 3 or 2 of the 2 operable channels, whichever is the applicable case. / / /

4.19 POWER DISTRIBUTION LIMITS

4.19.2 RADIAL PEAKING FACTORS

SURVEILLANCE REQUIREMENTS

4.19.2.1 The measured radial peaking factors (F_r^A, and $F_r^{\Delta H}$) obtained by using the incore detection system, shall be determined to be less than or equal to the values stated in the LCO at the following intervals: /

a After each fuel loading prior to operation above 50% of rated power, and

b. At least once per week of power operation.

4.20 MODERATOR TEMPERATURE COEFFICIENT (MTC) /

SURVEILLANCE REQUIREMENTS /

4.20.1 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 2% of rated thermal power, after each refueling." / / /

(next page is 4-90)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 118 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated December 23, 1987, (Ref. 1) Consumers Power Company (CPCo) (the licensee) submitted a draft of proposed Technical Specification changes to Provisional Operating License DPR-20 for the Palisades Plant to modify the Reactor Protection System (RPS). A subsequent meeting was held with the NRC staff on February 17, 1988, and a final version of the proposed changes was submitted to the NRC on March 25, 1988 (Ref. 2). The staff requested additional information (Ref 4.) which CPCo responded to in June, 1988 (Refs. 5 & 6). The staff also performed audits at the Palisades site and at the primary instrumentation vendor (Gamma-Metrics). The proposed change involved extensive changes to the RPS.

The purpose of the change is to improve the capabilities of the RPS by reducing the uncertainty associated with certain process variables while maintaining required thermal margins. To effect this modification, CPCo is replacing the existing analog trip calculators with microprocessor based, programmable calculators referred to as the Thermal Margin Monitor (TMM).

Included as part of the TMM modifications are: the addition of a Variable High Power Trip; the addition of an Axial Index Alarm; the implementation of an improved Thermal Margin/Low Pressure Trip; the addition of an alarm that monitors the Maximum Cold Leg Temperature; and the modification of the High Rate Trip Bypass hardware. Key issues to be addressed in this SE are the differences in the design bases for the new microprocessor based system and the original analog system; potential hardware vulnerabilities and susceptibilities; software development/verification and validation/configuration management; system failure modes and effects; and the development history of the system.

The licensee submitted three Advanced Nuclear Fuels Corporation (ANF) proprietary reports in support of the proposed changes describing the disposition of Standard Review Plan (Ref. 27) Chapter 15 events (Ref. 28) and the event analyses performed with the modified RPS (Ref. 29) as well as a low flow trip setpoint and thermal margin analysis for three primary coolant pump (PCP) operations (Ref. 30).

By letters dated September 1, 1988 (Ref. 32) and September 19, 1988 (Ref. 36), the licensee submitted additional proposed Technical Specification changes

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to Provisional Operating License DPR-20 for Cycle 8 operation of the Palisades Plant. These proposed changes include an increase of 3.5% in the radial peaking factors resulting from the implementation of a low radial leakage core to reduce reactor vessel fluence. The accommodation of these increased peaking factors, as well as the other proposed modifications for Cycle 8 operation, is based on the ANF evaluation of Standard Review Plan (SRP) Chapter 15 events (Refs. 33 and 37) and on the ANF large break loss of coolant accident (LOCA) analysis (Ref. 34) for Palisades Cycle 8. These analyses incorporate the modified RPS discussed above.

2.0 DISCUSSION AND EVALUATION OF RPS MODIFICATIONS

2.1 System Description

2.1.1 Reactor Protection System

The RPS will be modified to add new Thermal Margin Monitors (TMM), which will contain Thermal Margin/Low Pressure (TM/LP) and Variable High Power Trip (VHPT) circuits. An Axial Shape Index Alarm (ASIA) circuit will be installed in the control room in the locations presently occupied by the delta-T Power Reference Calculators which will no longer be used. An alarm that monitors the maximum cold leg temperature (T-inlet) and a modification to the High Rate Trip Bypass hardware will also be made.

The existing High Power Trip (HPT) is being replaced by the VHPT. The VHPT will provide a trip signal into the existing RPS trip logic which remains unchanged. The FSAR section 7.2 description of the RPS is still applicable except for the substitution of the VHPT for the HPT. The VHPT will provide a pretrip signal and a trip signal at 5% and 10% increases above the current power level. During power ascensions, the trip setpoint can be increased manually by the operator using the TMM keyboard or by an external pushbutton. During power descents, the trip and pretrip setpoints automatically decrease. The trip setpoint can be set by the operator from 30% to 106.5% of rated power for four pump operations. For three pump operation, the operator can vary the trip setpoint from 15% to 49% of rated power. The VHPT is intended to increase the thermal margin allowing Palisades to maintain the current 100% (2530 MWT) rated power level while providing early detection and termination (via RPS logic trip input) of reactivity insertion transients.

The Axial Shape Index (ASI) function is derived from the power range safety excore detectors for upper and lower neutron flux power. The TMM calculates the ASI after correction for detector geometry. Positive and negative setpoints are generated as a function of measured core power and an alarm is actuated if the corrected ASI is not bounded by the setpoints. This enhanced core protection is provided for additional protection for anticipated fuel assembly design changes. The existing TM/LP trip analog calculators are being replaced by the TMM programmable digital calculators. The TM/LP trip is enhanced by using the maximum of the neutron flux power and the delta-T power as direct input parameters to the TMM which calculates core power independent of core temperature. Also, the new TM/LP trip function will be corrected by the measured ASI. The purpose of the changes is to reduce the uncertainties associated with the TM/LP trip thereby gaining the additional operating margin desired by CPCo while maintaining the required thermal margins.

A T-inlet maximum alarm will be generated if the T-inlet is greater or less than the preset setpoints. The purpose of this modification is to alert the operator of an impending limiting operating condition in order that appropriate action may be taken.

Except for the ASI circuit and the substitution of the VHPT for the HPT, the RPS is not changed by the installation of the TMM. The TMM replaces two (HPT and TM/LP trip) of the 11 existing RPS inputs which can cause reactor trips.

2.1.2 Hardware

The primary hardware change of this modification entails the addition of four digital calculator trip/alarm TMM units (one per channel) which will physically and functionally replace the delta-T Power Reference Calculators. The delta-T calculators did not provide input to the RPS but did provide input to the deviation (thermal vs. nuclear) meter and alarm. The TMM replaces the existing TM/LP calculators, high power trips and power ratio calculators. The TMM is intended to be a virtual one for one functional replacement of the previous analog delta-T calculator system. The X1/X10 push buttons (miniature scale multiplier lights) will be removed since they are no longer required due to the TMM CRT displays.

Existing cable, raceway and conduit will be retained as much as possible. New annunciator alarms and related cables will be required as well as some new signal input cabling. New dual indicating meters will be installed in the control console to replace the previous single indicating Nuclear Percent Power Meters. Physical separation and electrical isolation will conform to IEEE STD-279, IEEE STD-384 and GDC-24. Previous separation criteria external to the TMM will be retained.

2.1.3 Software

The TMM portion of the RPS is dependent upon the quality of the software used for the calculator algorithms. The TMM hardware supplier, Gamma-Metrics, is also the supplier of the safety related software. CPCo has specified and Gamma-Metrics has implemented ANSI/IEEE-ANS-7-4.3.2-1982, "American National Standard Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations", for the TMM software. The staff recognizes this standard and endorsed it with Regulatory Guide 1.152 as an acceptable method of verification and validation to be used in development of safety related software.

The staff and consultant (review team) performed an audit review of the software development and the verification and validation (V&V) methodology used by Gamma-Metrics. The objective of the review team was to verify conformance to ANSI/IEEE-ANS-7-4.3.2.

2.2 System Assessment

2.2.1 Design Basis

The principle criteria of interest in this evaluation were ANSI/IEEE-ANS-7-4.3.2, IEEE-279, IEEE-603, IEEE-384, IEEE-323, and IEEE-344.

During the visit to the Gamma-Metrics facility, the staff confirmed that the methodology used for software development demonstrated conformance to ANSI/IEEE-ANS-7-4.3.2-1982.

2.2.1.1 IEEE STD-279-1971 & 603-1980

Regarding IEEE-603, during the visit to the site, the staff discussed a limited sample of design elements regarding conformance to the single failure criterion, completion of protective action, independence, capability for test and calibration, and indication of bypasses. Based on the licensee's statement of commitment to IEEE 603-1980 and onsite discussion of these attendant design attributes for the TMM, the staff concludes that the degree of conformance to IEEE-279-1971 and IEEE-603-1980 has not been diminished as a result of the addition of the TMM. This determination generally reflects CPCo's conclusion that the TMM is transparent to the RPS with respect to single failure, completion of protective action, capability for test and calibration, and indication of bypass.

2.2.1.2 IEEE STD-384-1977

Regarding conformance to IEEE-384, the staff noted that these criteria for circuit independence (particularly for isolation requirements) do not appear to have been invoked in any of the procurement documentation or vendor design documentation. The modification description submitted in support of the Palisades Amendment Request (Ref. 25) provided a fairly detailed narrative description of the isolation and separation features and appeared to claim conformance to IEEE 384-1977. The staff discussed with CPCo the extent of conformance, and examined a sample of the design. The TMM trip channel outputs to the RPS trip matrices and the safety/non-safety-related output isolation and separation features were reviewed. Inspection of the TMM cabinet (containing chassis for four independent channels, vertically stacked with separating barriers) indicated that TMM safety-related input/output and isolated non-safety-related output wiring for a given channel is bundled together within the TMM cabinet. No supporting analysis was provided to justify this configuration. This appears to the staff to be a nonconformance to IEEE 384-1977 in that these non-safety-related outputs from four independent TMM channels ultimately are terminated at a common annunciator. The staff also notes that this is an unanalyzed EMI condition (discussed further in Section 2.2.3 of this SE).

The licensee committed to the following corrective action prior to operation of the TMM:

1. Provide appropriate separation of 120 VAC power and instrumentation wiring to address EMI concerns.

2. Provide appropriate separation of Class 1E/Non-Class 1E wiring to meet IEEE-384-1977.

Apart from potential EMI conditions separately discussed, based on a cursory inspection of the TMM cabinet and discussions with CPCo, the remaining areas of TMM input and output signal independence appear to conform to IEEE 384-1977 and the Palisades design basis.

2.2.1.3 IEEE STD-323-1974

Regarding conformance to IEEE-323, the staff noted FSAR commitments stating that electronic equipment used in plant safety-related components can operate at 120°F continuously and 140°F intermittently, and at 135°F for portions of the RPS in the control room. CPCo Spec. J-54 (Ref. 7) stipulates a 120°F environment; Gamma-Metrics Spec. (Ref. 9) indicates 131°F temperature with no degradation in performance. These specified values are not sufficient to meet the FSAR requirements cited. Additionally, the staff noted during its cursory inspection of the TMM rack that CPCo had not assessed the effects of the site specific configuration of the four TMM chassis in a single enclosure having internal barriers and limited clearances for free convection. Specifically, the horizontal separation barriers use one inch thick marinite board which is necessarily a thermal insulator (since it is intended as a fire barrier), and communicating pathways between channel compartments are sealed to maintain independence. The effects of this configuration on the thermal qualification of the TMM has not been analyzed.

Further, the staff determined that the installed system has apparently been operated to date with the rear (louvered) door removed, so there is no operating history with the system in its final installed configuration.

Although the staff believes that temperature testing has not been demonstrated to bound the installed condition, we believe that a recent commitment to NRC via standing order 54 provides reasonable assurance that the TMM can operate properly. Standing order 54 provides an operating restriction that the maximum control room operating temperature is limited to 90°F. Temperature is verified as an existing surveillance item every shift. While 90°F operating temperature is a significant reduction from the 140°F FSAR design bases and the previous 120°F Technical Specification limit, the staff requires that CPCo demonstrate that the TMM is qualified to its design bases requirement. Based on these observations, the staff concludes that although the licensee and vendor stipulate IEEE-323, the basis for thermal qualification of the TMM in its operating configuration is unresolved. The staff requires that CPCo provide documentation which shows that the TMM design basis operating temperature is bounded by the qualification testing.

2.2.1.4 IEEE STD-344-1975

Regarding IEEE-344 conformance, the staff notes that this requirement was properly stipulated in the licensee and vendor specifications, and that CPCo has stiffened the TMM panel after evaluation of seismic test reports for the panel in which the TMM is now located. These tests included onsite seismic testing. On that basis, the staff concludes the installation appears to conform to IEEE-344.

2.2.1.5 Functional Requirements

Functional requirements for the TMM are provided in considerable detail in CPCo Spec. J-54, CPCo drawing 8-JL-130, and the Gamma-Metrics Software Requirements Specification (Ref. 8). The staff noted some discrepancies on 8-JL-130, but they were considered minor since CPCo uses the Technical Specification and Technical Specification Basis document for identifying the TMM

parameters of record. The Technical Specification Basis document properly references the ANF Analysis ANF-87-150 dated June 1988, which has been reviewed and is found acceptable as reported in later sections of this report. The CPCo Plant Review Committee reviewed the Technical Specification as the basis for the design, and the drawing 8-JL-130 was used only for procurement. While this is acceptable to the staff, we also believe that CPCo should maintain their engineering documentation to reflect as-built conditions.

Apart from the accident, transient and thermal hydraulic analyses discussed later in this report, the staff discussed with CPCo additional analyses provided in support of the modification. Two major analyses not provided by the licensee were the failure modes and effects analysis (FMEA) and a system reliability analysis. Also, no task analysis had been performed for the TMM man/machine interface. The staff concludes these are shortcomings in the licensee's safety analysis. In the absence of such analyses, the staff assessed these areas by discussing them with the licensee.

2.2.2 Failure Modes and Effects

The Palisades design basis includes an FMEA for the RPS. During the site visit, the staff requested the licensee to demonstrate acceptable outcomes for the upgraded RPS by walking through a limited sample of postulated design basis failure modes. The licensee was able to demonstrate successful outcomes for this limited sample, and provided reasonable assurance as to the functional equivalency of the upgraded RPS with respect to failure modes and effects. However, the licensee had not evaluated the effects of failure modes internal to the TMM. The staff concludes that a more exhaustive FMEA is required.

During the onsite visit, the licensee walked through the effects of postulated design basis failure modes for a sample of TMM inputs. Included in this sample were the following failure modes:

1. Loss of signal or open circuit of signal leads
2. Loss of signal power source
3. Loss of TMM power supply
4. Loss of two high pressurizer pressure trip signals

As part of the original design basis for the analog RPS, diversity had to be demonstrated for single failure of two high pressurizer pressure trip signals. The licensee successfully demonstrated acceptable outcomes for these scenarios that were consistent with the analog RPS design basis. In addition, the licensee discussed informal testing directed by CPCo that was intended to develop a "feel" for failure modes and effects internal to the TMM. However, no formal FMEA or formal test program was implemented. Noting that a digital system may have different failure modes than an analog system (such as system stall or timing errors), the staff believes a comprehensive FMEA is required to confirm that the original RPS design basis is maintained. In light of the existing FMEA for the RPS, the staff concludes that CPCo should perform an FMEA for the TMM. It is also the staff's opinion that the TMM as installed in the RPS satisfies single failure criterion. Therefore, the staff believes that TMM is acceptable for restart, however, a more thorough analysis should be performed to assure that all potential failure modes and effects have been considered.

2.2.3 Effects of Retrofit of New Equipment to Existing Systems

The TMM is intended, with few exceptions, to be a one for one replacement of the existing analog equipment. Most of the previous connecting cables and equipment remain unchanged. The output from the TM/LP (Ptrip) calculation in the TMM is a 10-50 ma (representing 1500-2500 PSIA) signal to the RPS trip logic. A complete failure of the TMM would produce a zero output signal (0-ma) which is consistent with the original design. Similarly, the VHPT input to the trip unit is unchanged.

In the Palisades Plant Reactor Protection System Common Mode Failure Analysis (March 1975), the RPS was reviewed for the TM/LP and High Power Trip. The analysis showed that a functional element failure could lead to an untripped output. This situation still applies to the new TMM with potential internal failures which may result in an untripped output upon receipt of a valid trip input signal. For a loss of power or similar total failure of the TMM, the TM/LP low setpoint alarm sounds which requires operator action to clear. The VHPT output will go to a trip condition. The consequences of a totally failed TMM are similar to the previously analyzed and accepted analog system and are therefore acceptable to the staff.

Electromagnetic interference (EMI) which doesn't affect an analog system may affect one that is based on digital microprocessors. The staff reviewed the actions taken for this modification by the licensee and has concluded that further action is warranted. Gamma-Metrics has successfully performed a limited test of the TMM's susceptibility to EMI. The licensee has noted that Non-Class 1E computer equipment powered by the same quality inverter power supplies has not demonstrated EMI related problems. The staff finds that CPCo has not shown that the installed TMM is immune to EMI or that EMI generated by one TMM does not affect one of the other channels. Further discussion and evaluation of EMI is presented in Section 2.3.1.3.

2.2.4 Maintenance of the Design Basis

The TMM is designed so that the operator may change trip setpoints and operating parameters by manipulation of a self contained key pad. The staff has reviewed the adequacy of the administrative controls which provide assurance that only properly controlled changes will be made during operations.

The primary control to assure proper maintenance of the design basis is control of the operators themselves. The only parameter which routinely changed and the only parameter that the control room operators are allowed to change on their own authorization is the "Bias" term, which calibrates the thermal power signal, "B" to a plant heat balance. The plant operations procedures GOP-12 and SOP-35 require independent technical review prior to changing the TMM.

The operators have been specifically excluded from revising any of the other parameters. The staff reviewed this situation due to the TMM operations manual (Ref. 14) description which noted that operator adjustable values within allowable ranges can cause "strange and erratic" system performance as well as crashes if the operator inputs unusual combinations of extreme values. This was an apparent contradiction of IEEE 603-1980 (specified for the TMM in J-54) which

requires, in part, that "safety systems shall be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis."

In order to change the value of any parameters, other than "Bias," the surveillance procedure QI-25 "TMM Constant Checks," which lists all of the parameter values will have to be administratively changed. Any changes to QI-25 would require a 10 CFR 50.59 safety evaluation and Plant Safety Review Committee concurrence. CPCo does not anticipate many revisions based on the previous operating history which had virtually no changes in 17 years other than the expected bias.

CPCo has demonstrated that their administrative controls will prevent any unusual combinations of inputs which could lead to an inadvertent system crash. The change to the "bias" parameter and the testing of the TMM is controlled by a keylock switch which has administrative controls to assure controlled use. Before testing or modification, the channel is placed in either bypass or trip as required by the Technical Specifications. Independent verification of any changes to the bias and verification that no inadvertent change to the other parameters has occurred must be completed prior to returning the affected TMM to service.

The staff considers the limitations imposed upon the parameter modifications to be a prudent action by CPCo due to the possible erratic behavior of the system. The staff finds the administrative controls to be acceptable.

The staff reviewed the drawing configuration control as shown by revision 5 of the 8-JL-130 drawing. The staff finds that the changes made in this case provide more complete information of the operating parameters, were well controlled, accurate, documented, and therefore acceptable.

2.3 Hardware Assessment

2.3.1 Hardware Specifications

The TMM is a standard 19" rack mounted panel with a self contained CRT graphics display and a removable keypad. CPCo specified the TMM with document J-54 (Ref. 7) which included as an attachment drawing 8-JL-130. J-54 describes the functional requirements, operating conditions, required testing, and standards with the intent of assuming a fully qualified Class-1E system. The 8-JL-130 drawing provides a logic diagram of the desired functions of the TMM and its interfaces.

2.3.1.1 Isolation Devices

Isolation devices are provided to prevent faults in Non-Class 1E equipment from propagating to the TMM. IEEE 279-1971 requires isolation to keep the Non-Class 1E fault from propagating into the safety function of the TMM. Palisades uses two types of isolators for the TMM outputs. The first isolator is a Hewlett Packard (HP) 6N137 optical isolator which must withstand a maximum credible fault on the Non-Class 1E side of 125 VDC @ 10 amps. This isolator has the same internal components and has been manufactured by the same assembly process as

the HP-2630 which was previously reviewed and approved by the staff. Therefore, the use of the HP-6N137 isolator is acceptable for this application.

The other isolator is the OPTO22 ODC5AQ optical isolator which must withstand a maximum credible fault of 125 VDC @ 3 amps. This isolator is similar to the OPTO22 DC200P isolator with the only difference being mounting method and control level voltage. The isolation manufacturer, OPTO22, has confirmed that these isolators are electrically the same, with the internal components providing the isolation being identical. The DC200P has been previously tested at higher voltage and current levels than exist at Palisades for this application. The staff reviewed the DC200P and found it acceptable, therefore, the ODC5AQ is also acceptable.

All input and output lines are specified as isolated from the chassis, from other input/output lines and from the TMM power supplies. Each of the four TMM modules is mounted independently and is physically and electrically separated from each other. The qualification of the TMM system is addressed in Section 2.2.1 of this report.

2.3.1.2 Reliability

The reliability of the TMM was not specifically specified by CPCo in the functional requirement. Several items of J-54 (Ref. 7) were, however, intended to produce a reliable component such as the seismic, temperature and EMI testing. The staff requires that the replacement equipment used in a design change be as good as or more reliable than the equipment that is being replaced. The original analog equipment had deteriorated over the years and maintenance parts have become scarce such that the analog system was no longer as reliable as when new. No specific component reliability numbers are available for the existing analog equipment.

This is the first application of a TMM. There is no operating history other than testing at Gamma-Metrics and at Palisades. To date, only two items have been discovered by Gamma-Metrics related to hardware. The first was the inadvertent failure to remove a jumper when the TMM units were put into storage which allowed the internal batteries to drain resulting in dead batteries. The second item involves a partial failure of a video driver chip which the staff witnessed as having a very minor effect on the CRT display. Gamma-Metrics has stated that they intend to replace this chip with one which they believe will be more reliable and will provide better picture clarity. Gamma-Metrics stated that they have not had any repeated failures in other equipment of any components used in the TMM. However, CPCo has identified three hardware failures since testing of the TMM at Palisades started.

As part of the TMM qualification test report, Gamma-Metrics did review the expected sub-component estimated mean time to failure. These data, which were derived primarily from military specifications, show an average life of 1.4 years for the CRT and 3 years for a Ni-Cad battery to 10,000 years for some connectors and resistors. Gamma-Metrics has not generated any reliability figures for the TMM.

Palisades has added TMM internal battery replacement to their scheduled preventative maintenance program. Other components such as the CRT and internal power supplies which have an estimated average life of less than 40 years have not been added to the preventative maintenance program.

A few failures have been experienced during installation and startup of the TMM. CPCo has made an estimate assuming one failure per year and taking credit for the operator verification that the TMM is operating properly every 12 hours and the unavailability is 2.5×10^{-9}. Due to the lack of operating data or detailed sub-component level failure analysis, the staff does not accept the estimate as a demonstration of reliability.

The staff also recognizes the operator training on the TMM which should enable them to quickly identify an inoperable TMM. In addition, the microprocessor hardware self diagnostics and the annunciator alarms for TMM trouble will provide continual surveillance to augment the required Technical Specification surveillances. The staff finds that the maintenance work requests are a proper means of developing an operating reliability history. Due to the lack of a specific reliability study and the hardware failures at Palisades, the staff requested the licensee to collect data on TMM failures and document the root cause, extent of failure, and corrective action in an auditable report. The licensee agrees with the staff and made a commitment to do so.

2.3.1.3 Electromagnetic Interference (EMI)

The staff concluded that minimal consideration had been given to the effects of electromagnetic interference (EMI). While we noted Gamma-Metrics had performed some nonrigorous testing in the factory, not enough detail of the test configuration and acceptance criteria was provided to determine whether the test was conclusive. The staff also was unable to find any requirements addressing EMI susceptibility in any of the design documentation.

Observations during our site visit further confirmed our belief that more attention was required in the design. These observations include:

1. The licensee was unable to retrieve any records of measurements that would establish the power line quality or EMI environment at the plant. It is believed no such measurements have been made.

2. The licensee did not have an overall description or program that would clearly establish the design basis shielding and grounding configuration as installed in the plant.

3. The design basis for surge withstand capability for the TMM was not clear.

4. The potential effects of radiated or conducted emission of one TMM channel's computer to another channel in the installed configuration had not been considered. No qualification data were available.

5. The staff noted that the TMM inputs are single-ended and ungrounded; CPCo had not evaluated the suitability of the existing input cable with respect to common mode rejection capability. A cabling configuration adequate for the old analog system is not necessarily adequate for a digital system.

6. The staff observed that the panel adjacent to the TMM contains ammeters and voltmeters connected to current and potential transformers on the 345 kV switchyard buses. The potential for EMI coupled from these sources was not recognized or considered in the design.

Although the staff has raised these concerns, the following compensating considerations are noted:

1. Even if catastrophic failure of the TMM due to EMI were to occur during a challenge to the RPS (an event judged unlikely by the staff), the remaining RPS functions external to the TMM would be expected to remain available because of isolation features provided in the TMM; moreover, these diverse analog channels have not been susceptible to these types of failures in 17 years of operation.

2. The existing configuration of the RPS provides some design features such as twisted/shielded pair analog input cabling with shields grounded at a single point, physical separation of channel signal inputs and channel power inputs, and the use of conduit.

3. For many events challenging the digital TMM, functional diversity is provided by analog signal processing of diverse parameters.

4. Limited EMI testing of the TMM has been performed by the vendor, with no reported failures.

5. Some design features are provided in the TMM to address EMI, such as hardware and software filters.

Given these considerations, the staff has reasonable assurance that the effects of EMI will not adversely affect the safety function of the RPS. However, the staff requires that confirmatory analyses and tests, as necessary, be performed to assure that there are no unanticipated adverse effects. The staff concludes that there is no undue risk to the health and safety of the public by operation in the interim (approximately one year) until the confirmatory results are documented.

2.3.2 Operating Considerations

For the operations at Palisades, the TMM is required for several functions. The primary function, which is designed to be operator independent, is to provide an accurate trip signal on TM/LP or VHPT. The visual CRT display is required to meet the Technical Specification requirement that the T-inlet and ASI trip values are being met. A trip signal displayed on the TMM would indicate that the Technical Specifications have been exceeded. The staff finds this an acceptable use of the TMM.

The operators have been thoroughly trained in the use of the TMM. The staff notes that the operators suggested a change to the annunciators as a result of the hands-on training with the spare unit and at the simulator. The operators also trained with a PC input simulator similar to the Gamma-Metrics test configuration. The staff finds the operation's staff at Palisades has been adequately trained on the TMM.

2.4 Software Assessment

The assessment of the TMM software built by Gamma-Metrics is an assessment of the methodology and procedures used to develop the software. The TMM software began with a functional requirements document presented to the Gamma-Metrics by CPCo. The Gamma-Metrics development of software is to be evaluated by reviewing the verification and validation trail through the development process.

Verification and validation (V&V) are two separate but related activities that follow the development of software. Verification determines whether the requirements of one phase of the development cycle have been consistently, correctly, and completely transformed to the subsequent phase of the cycle. Validation is the testing of the final product to ensure that performance of the end product conforms to the requirements of the initial specification. The need for V&V arose because software is very complex and prone to human errors of omission, commission and interpretation. V&V provides for an independent verifier to work in parallel with, but independent of, the development team to ensure that human errors do not hinder the production of safety software that is reliable and testable.

In executing V&V, certain principles have proven over time to be very effective in software development programs. These V&V principles, can serve as a comprehensive reference base for applying the applicable criteria for software evaluations of Class 1E safety systems.

- o Well defined systems requirements expressed in a well written document.
- o Development methodology to guide the production of software.
- o Comprehensive testing procedures.
- o Independence of the V&V team from the development organization.

2.4.1 Criteria

The applicable criteria for the development of safety related software products are set forth in ANSI/IEEE-ANS-7-4.3.2. This standard defines the documentation of the computer system requirements, the software development phases, the verification testing of the integrated system, and the validation of the entire development process. The staff review focuses on whether a comprehensive V&V program is applied to the software development of safety related systems, and that the V&V program was carried out according to the applicable criteria. In particular, the following elements are characteristic of an acceptable V&V process.

2.4.1.1 Independence

The verifier must be independent of the developing organization. Although the verifier must work closely with the development team throughout the life-cycle, he will report to someone not connected with the development project. A verifier from outside the development company is commonly used. However in large companies, the staff requires for minimum acceptable independence, that the verifier is not part of the developing organization and reports to a different first-line supervisor.

2.4.1.2 Validation Testing

The validation testing must be done by a team that did not participate in the design or implementation of the software product. Validation testing can be done by someone external to the development company or it can be done by the

licensee as part of acceptance testing of the final product.

2.4.1.3 Discrepancy Resolution

A key element in any V&V effort is the process by which discrepancies uncovered during verification are recorded, identified, resolved and corrected. The resolution of a discrepancy must be reflected in all applicable documents, whether source code, the software design specification, the software requirements, or the original systems specification.

2.4.1.4 Design Approach

The primary specification for the software provides the foundation for not only sound development but also of effective V&V activities. The individual requirements in the specification for any software system describe how the software is to behave in any circumstance. The specification must be reliable and testable. A reliable specification exhibits the following characteristics:

a. Correct - Each requirement of the safety function has been stated correctly.

b. Complete - All of the requirements for the safety function are included.

c. Consistent - The requirements are complementary and do not contradict each other.

d. Feasible - The requirements can be satisfied with available technology.

2.4.2 Software Evaluation

The software development plan for the TMM was contained in the Software Quality Assurance and Development Plan document (Ref. 12). This document was first published in September 1985, soon after the initiation of the project, and revised in October 1985. The review team concluded the document was well written with regard to the above criteria for the development of the software. However, the Plan did not address the topics of requirements traceability, verifier independence, and personal responsibility as expressed by signoffs. These topics are addressed in later sections of this report.

It is recommended to the vendor, for future applications, that a corporate software development plan be written that can be applied to all Class 1E software development. The plan should include a description of the development phases in sufficient detail so that the V&V efforts can be initiated at the beginning of any design effort. The plan should contain a taxonomy of documentation and reviews which demark the injection points for V&V activities. The existence of a corporate software development plan for Class 1E systems is evidence of a corporate commitment to satisfying ANSI/IEEE-ANS-7-4.3.2 and to the development of reliable software.

2.4.2.1 Task Analysis

In reviewing the documents, it became apparent that there was no formal task analysis to support the design of the operator interface. The initial specification provided by the licensee did not require an interactive operator

interface. This interface was suggested by the vendor to make the TMM more useable. From discussions with the vendor's staff, the review team concluded that the interface was developed in an iterative manner by the vendor's programming team. There was also some support and input into the design by operations people from CPCo. A task analysis would have minimized the software development (iterative process); however, the review team concluded that the operator interface functioned as defined in the J-54 requirements document and is, therefore, acceptable.

2.4.2.2 Design Deviations

The design of the TMM was in two phases. The first phase developed the Software Requirements Specifications (SRS) (Ref. 8) from the functional specification (Ref. 7) provided by the licensee. The development of the SRS (Ref. 8) underwent four (4) iterations during the beginning of the project in third quarter 1985. Each iteration represents a further refinement of the software requirements for the TMM, with the fourth iteration approved by the licensee. There was no evidence of the specific methodology or analysis used to develop and refine the broadly stated requirements of J-54 (Ref. 7) into software requirements. It was not evident how the individual requirements of J-54 (Ref. 7) were decomposed into the more detailed requirements in a traceable manner, but the verification of these iterations showed few discrepancies, leading the review team to conclude that all individual requirements were included.

The second phase developed the Software Design Description (SDD) (Ref. 19) from the SRS (Ref. 8). The SDD went through five (5) iterations during the coding phase indicating that the design process was iterative and that the SDD was kept synchronized with the code actually being developed. The review team concluded that this tightly integrated design process was an asset to producing reliable software. A cursory review of the SDD (Ref. 19) indicated that the design appeared to be structured. The verification reports for the SDD were global and listed only discrepancies found, leading the review team to conclude that all the individual requirements of the SRS were included in the SDD.

2.4.2.3 Formal Design Reviews

The formal requirements and design reviews required by the Software Quality Assurance and Develpment Plan (SQADP) (Ref. 12) were conducted in the early stages of the development. Although the documentation of the meetings were sparse, it appeared through discussions that the Design Reviews focused on the major issues. The iterations of the requirements and design documents early in the project indicates the vendor's commitment to developing a coherent design prior to extensive implementation, a positive factor for reliable software. There was no evidence of design reviews during the code development phase of the project. Overall, the review team concluded that the formal reviews ensured that the resulting design would be correct.

2.4.3 Development Methodology

The process of software development for safety related systems has to be methodical and controlled so that the status of the software is known at any time. A well defined and controlled development process provides a good

environment for effective verification as well as management oversight. The development methodology for the TMM is described in SQADP (Ref. 12), and it provides an overall view of the major development phases and quality assurance activities.

2.4.3.1 Development Environment

The software development environment was comprehensive and included the use of simulators and emulators to debug the source code. The code generation and compilation and integration was done on IBM personal computers using Intel Corporation's development system and software tools. For software integration into the target environment of TMM hardware, an emulator was used to provide access to the internal elements of the software during integrated testing. A rudimentary model of a nuclear reactor was developed for an Apple computer which provided "inputs" to the test bed TMM. The review team concluded that the development environment used was conducive to rapidly developing and testing reliable software for the TMM.

The development approach was an iterative top down process with the emphasis on quickly prototyping the software elements of the design. The major elements of the software were defined early and integrated as "stubs" or placeholder programs. This enabled the design team to determine early whether their software architecture and implementation would work. The design documents were continually updated to reflect the changes in the architecture and provided to the verifier. It was through such early tests that the development team concluded that the original timing parameters could not be met. They initiated a dialog with the licensee, which resulted in a change to the specification that was technically feasible. The review team concluded that the development approach was sound and consistent with ANSI/IEEE-ANS-7-4.3.2.

2.4.3.2 Source Code

The program code consisted of 59 modules, encompassing over 120 procedures. Many of these modules are data declarations for the hard coded parameters that are required by the TMM. Although seemingly contradictory to flexibility, the hard coded parameters are not as easy to change, and therefore resistant to spurious manipulation. In a safety system this is a positive attribute. The review team spot checked randomly selected modules for structure, comments, and traceability. The program source code appeared to be well structured and annotated with regard to function. In particular, the preamble to each module contained a list of all revisions made to the code since its initial coding. However, the annotations for revisions were not as complete as one would have liked. While the date and programmer identification were almost always present, the description of the change was often too terse, in particular with regard to the reasons the code was changed. The review team concluded that the source code was well structured and provided good traceability for code verification.

2.4.4 Verification

Verification is the process that determines whether or not the requirements detailed in one phase have been correctly transformed into the requirements of the next phase. The transformation must be sufficiently clear so that a person not involved with the software development can understand the steps

that the design and implementation have taken during the life-cycle. As the software is developed, each phase takes the requirements of the previous phase and adds another level of detail and expansion to those requirements. For the design verification, many software oriented techniques and tools are applicable and should be used when practicable. At the source code level, the details of the design become very important because the computer execution of the code is absolute.

The Verification Plan (Ref. 11) for the TMM was developed early in the project, at the same time that the Software Requirements (Ref. 8) were defined. It was well organized and included specific procedures and tasks for implementing the verification. For each phase and document that was to be reviewed, the Verification Plan listed the documents that the verification process would be based on. The review team examined the Verification Plan and concluded that it was a complete document that contained the major ingredients required by ANSI/IEEE-ANS-7-4.3.2.

2.4.4.1 Verification Process

The verification process of the TMM can be divided into two distinct variants, that which was done prior to 1988, and that which was done during 1988. For the verification execution prior to 1988, there was a general lack of documentation that the verification effort was done with the thoroughness described in the Verification Plan. The traceability of verification efforts during the major code development period of November 1985 to June 1986 is also weak.

The requirements from the J-54 Functional Specification (Ref. 7) provided by the licensee were not traced on an individual basis to the SRS (Ref. 8) and the SDD (Ref. 19). The verification reports treated the documents globally, bringing to light only issues that were not in conformance with the verification baseline. The omission of the other requirements was interpreted as being in conformance and accepted by the verifier. The tracking of individual requirements provides stronger assurances that nothing was omitted, and that the requirements were not misinterpreted by either the design team or the verifier. The review team concluded that the verification was technically in compliance with ANSI/IEEE-ANS-7-4.3.2, however, global document level verification weakens the traceability of individual requirements.

The major tool in the verification of software is the walkthrough of algorithms and the desk checking of code. This is especially true of the TMM with its extensive algorithms and data vectors, and commission logic. Although required by the Verification Plan, there was no evidence that algorithm walkthroughs and code desk checking were done.

One of the criteria in ANSI/IEEE-ANS-7-4.3.2 is the verifier must interact with the design group in a written form. This is interpreted to mean that transmittal letters are used to convey documents to the verifier and verification reports from the verifier to interested parties. Such transmittal letters provide traceability to the verification process, and another level of assurance that proper procedures have been observed. During the meeting with Gamma-Metrics, the review team was presented with no evidence of the written interaction between the verifier and the design team. No transmittal letters for reports, code reviews, and verification testing were available. There were no transmittal

letters conveying to the verifier the documents or code to be reviewed.

The verification process in 1988 was initiated when the licensee published Revision D of its J-54 specification (Ref. 7). In contrast to earlier efforts, this verification effort was precise and overcame the weaknesses exhibited in the previous verification process. This verification effort included a requirements matrix that tracked the requirements to changes in the documents and the code, an algorithm review, and a code review. The verification test was identical to the validation test and executed with the same rigor and formality. The review team concluded that the rigorous and exemplary manner in which this latest verification was carried out more than compensates for the earlier weaknesses.

2.4.4.2 Verification Testing

The Verification Plan had specific procedures for verification testing of the source code including desk checking and testing. From discussions with the vendor, it was indicated to the review team that the verifier had complete access to the program development activity. This included access to the source files for reviewing code, and the use of the development test bed for carrying out the verification tests. Much of the testing was done informally with no directly applicable test procedures and no recording of test data. Although no written evidence exists, the review team concluded that the verifier's access to source code and the development test bed resulted in effective verification testing of the TMM.

2.4.4.3 Independence

A key ingredient in an effective verification process is the independence of the verifier. Although the organizational charts showed that the verifier was reporting to the Director of Engineering, formal reports on verification activities were to be conveyed to the Manager for Quality Assurance. Furthermore, both the verifiers were independent contractors and the review team concluded that this allowed the verifiers to act with the required degree of independence.

2.4.5 Validation

The Validation Plan (Ref. 18) was written by the vendor's Director of Engineering based on the Validation Criteria provided by the licensee. The verifier confirmed that the Validation Plan was in agreement with the Validation Criteria, although there is no evidence that the Validation Criteria included all of the requirements of J-54. Although organizationally in consonance with ANSI/IEEE-ANS-7-4.3.2, the Validation Plan was not considered completed because it omitted the 3-pump test. The explanation given by the vendor was that the algorithms are identical in both the 3- and 4-pump tests and the 4-pump test operational locus includes the 3-pump operations as well. The review team concluded that the Validation Plan was sufficiently detailed to demonstrate via testing, that the functional requirements of J-54 have been satisfied by the TMM for both 3- and 4-pump operation.

2.4.5.1 Validation Test Procedures

Although the TMM consisted of seven functions, the permutations of data and inputs made the execution of the test rather complex. To ensure that all of the individual requirements had been satisfied through validation testing, formal test procedures should have been developed. These test procedures, in addition to helping the validation personnel keep track of test progress and test results, would also have served as written evidence of the extent or coverage of the validation test. The completed test procedures can then be used as the basis for a comprehensive validation report.

Portions of the validation test were demonstrated by the vendor to the review team. A clean copy of the Validation Plan was used as the test procedure document, with the text annotating test results in the margin. The review team witnessed selected portions of the validation test and found the test results in agreement with the Validation Plan.

2.4.5.2 Validation Test Results

The validation testing was carried out according to the Validation Plan, and the plan document was used to record the results of the test. The review team concluded that the complexity and coverage of the Validation Plan should sufficiently demonstrate that the TMM software is performing in accordance with the functional specifications provided by the licensee.

2.5 Conclusions on RPS Installation

2.5.1 Systems

The staff has concluded that this upgrade is an acceptable modification of the RPS and is allowable for restart of Palisades. We find that two confirmatory actions are required (Ref. 26). First, we find that the EMI effects on the TMM have not been fully analyzed and therefore, we require CPCo to provide an analysis (with testing if needed) to show that EMI will not impact RPS function. Due to the reasons stated in Section 2.3.1.3, the staff believes that no undue risk is presented to the operation of the RPS and, therefore, the required EMI analysis does not have to be performed prior to restart.

Second, it is the staff's opinion that the TMM as installed in the RPS satisfies the single failure criteria and therefore is unlikely to prevent the RPS from performing its safety function. However, to assure all possible failure modes have been addressed, we require that the Failure Modes and Effects Analyses be updated to include consideration of the TMM.

2.5.2 Hardware

The TMM is seismically and environmentally qualified. Electrical isolation has been shown to be acceptable. Based on the staff audit, the TMM conforms with the requirements of IEEE STD-279-1971. CPCo has committed (Ref. 26) to correct the separation problem of Class 1E and Non-Class 1E wiring bundled together within the TMM cabinet prior to restart. This commitment also included the separation of 120 volt AC power and instrumentation wiring. The separation criteria for the TMM shall meet the requirements of IEEE STD-384-1977. The

staff finds this acceptable. Due to the lack of a specific reliability study, the staff requires the licensee to collect data on TMM failures and document the root cause, extent of failure and corrective action in an auditable report. The licensee has committed to do this (Ref. 26). Though the staff believes that the 90°F control room temperature provides a substantial improvement in the limiting operating environment, CPCo is still required to demonstrate qualification to the design bases (Ref. 26). Therefore, the staff requires as a confirmatory action following restart, documentation which shows that the operating temperature of the TMM is bounded by the qualification testing.

2.5.3 Software

In summary, we conclude that the software for the TMM is the result of a well structured development methodology supported by comprehensive testing at the integration, verification, and validation level. The close interaction of the independent verifiers with the development team provided assurances that the functional requirements of the J-54 Specification have been successfully translated into software. Furthermore the V&V process used to monitor the development of the software conforms to ANSI/IEEE-ANS 7.4.3.2 and Regulatory Guide 1.152. Therefore, the staff concludes that the software can perform its safety function as part of the RPS at the Palisades Plant.

2.6 Accident and Transient Reanalyses

2.6.1 Variable High Power Trip (VHPT)

The existing high power trip will be replaced by the VHPT. The VHPT is designed to trip the reactor when the core power increases less than or equal to 10% above the current power level. During power ascensions, the trip setpoint can be increased by operator action. During power descents, the setpoint automatically decreases. The minimum trip setpoint is less than or equal to 30% of rated power and the maximum setpoint is less than or equal to 106.5% of rated power for four PCP operation. For three-pump operation, the minimum and maximum setpoints are less than or equal to 15% of rated power and less than or equal to 49% of rated power, respectively.

The Standard Review Plan Chapter 15 events for Palisades were reviewed by the licensee to determine if the event initiator or controlling parameters have been changed from the analysis of record so that the event needs to be reanalyzed for the current licensing action. Those events which required a reanalysis also incorporated the appropriate proposed RPS modification. The reanalyzed events which rely in part on the VHPT for mitigation are the: (1) rod bank withdrawal, (2) loss of external load, (3) uncontrolled control rod bank withdrawal, (4) boron dilution from power, hot standby, hot shutdown, and (5) control rod ejection.

Reanalyses of these events have demonstrated that appropriate acceptance criteria are met. The staff concludes, therefore, that the VHPT provides additional plant protection during rapid power transients as well as during slow reactivity transients such as the boron dilution event initiated at any power level.

2.6.2 Axial Shape Index (ASI) Alarm

The ASI is defined as the power in the lower half of the core minus the power in the upper half of the core divided by the sum of the power in the lower half and upper half of the core. The proposed thermal margin monitors use the excore detectors to determine the ASI after correcting for detector geometry. If the corrected ASI is not bounded by the ASI setpoints generated as a function of core power, a control room panel alarm is actuated. This RPS modification to monitor the axial power distribution will alert the operator in the event of operations outside the power distributions assumed in the licensing basis analysis and is, therefore, acceptable.

2.6.3 Inlet Temperature (T-Inlet) Maximum Alarm

An additional operation will be performed on the calculated T-inlet value. If at any time, T-inlet is greater than the present maximum T-inlet value or less than present minimum T-inlet value, an alarm signal would be generated. Since a new inlet temperature limiting condition of operation (LCO) has been developed for the Palisades Technical Specifications to provide protection against penetrating DNB during limiting anticipated operational occurrences (AOOs), this T-inlet alarm would alert the operator of an impending LCO in order that appropriate action may be taken. This is acceptable.

2.6.4 Thermal Margin/Low Pressure Trip

The replacement of the current TM/LP trip analog calculators by programmable digital calculators has been proposed. This new trip will use the maximum of the neutron flux power and the delta-T power as an input parameter instead of using the hot leg temperatures to implicitly measure core power. Since the trip actuation would no longer be strongly dependent on the response times of the primary coolant system resistance temperature detectors (RTDs) and the loop transport time between the coldleg and hotleg RTDs, the uncertainty in the TM/LP trip associated with core power measurement would be significantly reduced. The staff finds this acceptable.

The proposed TM/LP trip function would also be corrected by the measured ASI. The axial shape used in the development of the current Palisades TM/LP function was conservative in order to ensure that all probable axial shapes were bounded. For the proposed trip, the TM/LP can be developed for the optimum axial shape and the axial function would adjust the trip as the ASI varies from the optimum shape. Therefore, the fuel would be protected if adverse axial power distributions should develop during plant power operations. Although this modification would also reduce the uncertainty associated with the TM/LP trip and, therefore, provide additional operating margin, the required thermal margins would still be maintained. The staff finds this acceptable.

The TM/LP trip discussed is designed to protect against slow heatup and depressurization transients. Those events which rely in part on the TM/LP trip and which were reanalyzed for Palisades are the: (1) increase in steam and feedwater flow, (2) loss of external load, (3) loss of feedwater flow, (4) loss of reactor coolant flow, (5) dropped control rod bank, (6) boron dilution from power, hot standby, and hot shutdown, (7) uncontrolled control rod bank withdrawal, and (8) control rod ejection. The staff has reviewed the reanalyzed

events and finds the the appropriate acceptance criteria are met for each of these events.

Based on the above, the staff concludes that the proposed digital thermal margin monitor is acceptable.

2.6.5 Technical Specification Changes Associated with RPS Modifications

The proposed Palisades Technical Specification changes submitted by the licensee as an attachment to Reference 2 have been reviewed by the staff. In addition to the changes relating to the RPS modifications, changes were also proposed relating to Generic Letter 86-13 which are discussed in Section 4.0 of this report. The changes include the following:

1) changes bases and references to be consistent with the new analyses approved by the staff in this Safety Evaluation;

2) incorporates the setpoints for VHPT and TM/LP to be consistent with the modified RPS;

3) changes the inlet temperature limiting condition for operation and adds ASI restraints to operation to be consistent with the new approved transient analyses;

4) adds ASI to the Technical Specifications as a required functional unit; and

5) adds surveillance requirements for new hardware and functions of the approved modifications to the RPS, which are consistent with the type and frequency of testing for the other RPS equipment.

The staff finds these proposed Technical Specification changes acceptable. In addition, the following modifications were proposed in response to the staff's request (Ref. 8).

1) The CPC definition will treat "axial offset" and axial shape index" synonymously. The definition has, therefore, been clarified to state that the numerator is the power in the lower half of the core minus the power in the upper half of the core. This is acceptable.

2) The Basis statement of 3.1 has been clarified to state 3 primary coolant pumps. Since 2-pump operation has been deleted, reference to operation with 3 pumps is more appropriate than reference to operation with less than 4 pumps. The change is, therefore, acceptable.

3) Technical Specification 3.1.1.b has been changed to state "be in hot shutdown (or below) with the reactor tripped (from the C-06 panel opening the 42-01 and 42-02 circuit breakers) within the next 12 hours." This clarifies the allowed operation for 24 hours (12 + 12) with one pump out of service and also provides an acceptable means of preventing control bank withdrawal from this

mode of operation.

4) The Basis statement of 2.3.4 has been change to state: "For three pump operation, continued power operation is restricted." This is acceptable.

3.0 DISCUSSION AND EVALUATION OF RADIAL PEAKING FACTORS AND LINEAR HEAT RATE LIMIT

3.1 Thermal-Hydraulic Analyses

Two changes in ANF reload fuel are being made which require evaluation to assure that the thermal-hydraulic design criteria for fuel rod integrity are maintained during Cycle 8 operation. These changes are:

1) Insertion of four ANF lead assemblies with high thermal performance (HTP) spacers.

2) Insertion of 16 reconstituted shielding fuel assemblies at locations along the core periphery. The outer four rows of fuel rods along one side of each of these assemblies are replaced with stainless steel rods.

The thermal-hydraulic calculations for Cycle 8 have shown that the XNB 95/95 Departure from Nucleate Boiling Ratio (DNBR) limit of 1.17 is not violated in any standard HTP spacer lead assembly for any limiting AOO. The XNB correlation has been accepted for application to the ANF standard fuel assemblies. Since flow mixing data for assemblies similar to the HTP spacer design have demonstrated significantly improved mixing relative to the ANF standard spacer, the XNB correlation may be conservatively applied to the HTP spacer lead assemblies as well.

The impact on minimum DNBR of the 16 reconstituted shielding assemblies, which are loaded along the core periphery to reduce the neutron fluence on critical vessel welds, has also been evaluated for Cycle 8. Since these previously irradiated assemblies will operate at substantially lower power levels than the adjacent assemblies, the adjacent higher power assemblies may potentially experience an increase in cross flow which could adversely impact minimum DNBR. The thermal-hydraulic analyses have shown that the minimum DNBR for an assembly located adjacent to a shielding assembly is well above the XNB 95/95 correlation limit of 1.17. In addition, the shielding assemblies themselves will not penetrate minimum DNBR limits because of their relatively low assembly power levels.

These results demonstrate the thermal-hydraulic compatibility of the Cycle 8 fuel at Palisades. Since the analyses were performed with the NRC approved ANF thermal-hydraulic methodology (Ref. 35), the staff finds the proposed Cycle 8 fuel design acceptable.

3.2 Transient and Accident Analyses

The proposed increase in radial peaking limits will impact the minimum DNBR for several Cycle 8 transients. Therefore, the licensee has reanalyzed

those events which are not bounded by previous analyses to assure that with 95% probability and 95% confidence, DNB will not occur and that the fuel centerline melt threshold of 21 kw/ft will not be exceeded. Other Cycle 8 parameter changes such as initial and critical boron concentrations were also included in the reanalyses. Those events reanalyzed were the increase in steam flow, loss of external load, loss of forced reactor coolant flow, reactor coolant pump rotor seizure, uncontrolled control rod bank withdrawal, control rod misoperation, startup of an inactive loop, and boron dilution. With the exception of the uncontrolled control bank withdrawal from hot shutdown with only three primary coolant pumps in operation, all of these anticipated operational occurrences (AOOs) resulted in minimum DNBRs greater than the XNB critical heat flux correlation safety limit of 1.17 and maximum peak linear heat rates below the fuel centerline melt criterion of 21 kw/ft. Therefore, no fuel failures are predicted to occur.

Since the control bank withdrawal from hot shutdown with three pump operation resulted in a small amount of rods going into DNB, the staff requested the licensee to provide a positive means of preventing control bank withdrawal from these conditions. The licensee has, accordingly, modified Technical Specification 3.1.1.b to require the reactor to be tripped from the C-06 panel by opening the 42-01 and 42-02 circuit breakers if an out-of-service pump can not be returned to service and the reactor is in hot shutdown (or below). This provides an acceptable means of assuring that a control bank withdrawal would not occur from these conditions.

The rod ejection accident and the LOCA were also reanalyzed for Cycle 8. The radiological doses resulting from the rod ejection accident were found to meet the appropriate acceptance criteria.

The LOCA analysis for Palisades Cycle 8 (Ref. 34) was performed with the ANF EXEM/PWR evaluation model approved by the NRC (Ref. 35). Because of the new methodology as compared to the previous licensing LOCA calculations performed for Palisades, a mini break-spectrum analysis was required to verify the limiting break size. The results verified the 0.6 double ended cold leg guillotine (DECLG) break as the limiting break size. The analysis included calculations at the limiting break size for both a beginning of cycle (BOC) axial power shape peaked at a relative core height of 0.6 (15.28 kw/ft) and an end of cycle (EOC) axial power shape peaked at a relative core height of 0.8 (14.75 kw/ft). A total radial peaking factor of 1.92 was assumed compared to the 1.83 maximum value expected during Cycle 8 in order to bound potential future increases in radial peaking. A maximum average steam generator tube plugging level of 29.3% with up to 4.5% asymmetry was also assumed. The results meet the 10 CFR 50.46 acceptance criteria for peak clad temperature (2200 °F), peak local clad oxidation percentage (17%) and core wide clad oxidation percentage (1%) and are, therefore, acceptable.

3.3 Technical Specification Changes For Radial Peaking Factors, LHR, and Boron Dilution

Separate peaking factor limits for narrow gap ($F^{n}r$) and peak ($F^{T}r$) fuel rods are being removed from Technical Specifications 3.23.1, 3.23.2, and 4.19.2. In addition, the definitions of F^{T}_{r}, F^{N}_{r}, and narrow water gap fuel rod are being deleted from Section 1.1. This is acceptable since the Cycle 8 LOCA analysis

results show that these limits are always bounded by the more limiting interior rods.

The allowed radial peaking factors are being increased by 3.5% in Technical Specification 3.23.2. The limitations on the radial peaking factors have been used in the Cycle 8 analyses for establishing DNB margin, LHR and the thermal margin/low pressure and variable high power trip setpoints using approved methodology. The results of the analyses have shown that the increased peaking factors can be accommodated and since periodic surveillance requirements are in the Technical Specifications to assure that the measured radial peaking factors remain within the prescribed limits, the proposed increase is acceptable.

Separate LOCA kw/ft limits for narrow gap and interior rods are being removed from Specification 3.23.1. This is acceptable since the Cycle 8 LOCA analysis, which was performed for a maximum pellet LHR of 15.28 kw/ft using approved methodology, bounds peak pellet LHR limits for both narrow gap and interior fuel rods.

The burnup penalty previously applied to the LOCA kw/ft limits is being eliminated by removal of Figure 3.23-2. This is acceptable since the Cycle 8 LOCA analysis was performed at a peak assembly discharge burnup of 52.5 GWD/MT and bounds all assembly exposures less than this value. Therefore, the allowable LHR as a function of burnup is not required for exposures below 52.5 GWD/MT. The previous Palisades LOCA analysis required a reduction in the LHR at high exposures due to the different methodology used.

The shutdown cooling flow rate given in Technical Specifications 3.1.1a and 3.10.1c has been increased to 2810 gpm from 1500 gpm. By imposing a minimum shutdown cooling pump flow rate of 2810 gpm, analyses have shown that sufficient time is provided for the operator to terminate a boron dilution event with imperfect mixing during startup with a 2% delta k/k shutdown margin. This value was calculated by evaluating the minimum shutdown cooling pump flow rate necessary to bring the plant to a critical state in no less than 15 minutes. This meets the staff's acceptance criterion and the change is, therefore, acceptable.

The remaining proposed Technical Specification changes are revision to the Bases required for consistency with the above changes or are merely editorial in nature and are acceptable.

4.0 CHANGES RESULTING FROM GENERIC LETTER 86-13

Potential inconsistencies between Technical Specifications and FSAR analysis related to primary coolant pump (PCP) operation and primary coolant system boron concentration were identified in Generic Letter 86-13 (Ref. 31). These inconsistencies primarily involve safety analyses for events initiated from hot standby conditions (or below) which assumed operation of all PCPs while the plant Technical Specifications allowed less than full pump operation. As a result of their review of Generic Letter 86-13, changes have been proposed by the licensee in order to decrease the consequences of events initiated from low power which could be adversely affected by three-pump operation such as the main steam line break and the control rod withdrawal event.

The licensee has proposed a change to Technical Specification 3.10.1.c which would require that at less than the hot shutdown condition, the boron concentration must be greater than the cold shutdown boron concentration for normal cooldowns and heatups. This changes the required shutdown margin for less than four PCPs at less than hot shutdown from 2% to the cold shutdown boron concentration. This requirement with less than four-pump operation ensures that a steam line break occurring from this condition would not result in a reactor return to power. The acceptable results of a postulated steam line break event initiated with three PCPs operational, therefore, assures adequate shutdown margin exists with less than four-pump operation.

The reference analyses of the uncontrolled control bank withdrawal from a subcritical or low power startup condition considered withdrawals from source range to 10% of full power. However, these analyses did not consider operation with less than four PCPs in operation. A proposed revision to Technical Specification 3.1.1.b would require that four pumps be in operation whenever the reactor is operated continually above hot shutdown. In addition, startup above shutdown with less than four pumps would not be permitted. However, the specification does allow limited operation for up to 12 hours with one pump taken out of service with power limited to a maximum of 39% of rated power and total coolant flow limited to a minimum of 74.7% of rated flow. This would provide time for repair and restartup, or for an orderly shutdown.

An analysis of this event for three-pump operation with initial conditions which bound reactor critical, hot standby and hot shutdown modes resulted in about 2.3% of the core experiencing fuel failure. Since operation above hot shutdown with less than four PCPs requires an action within a specified time period (i.e., return the pumps to service within 12 hours or be in hot shutdown or below within 24 hours), the assumption of an additional failure such as an uncontrolled control bank withdrawal above hot shutdown with less than four operational pumps is not necessary. However, hot shutdown or below is an allowable unrestricted operating condition with only three pumps in operation and, therefore, an uncontrolled control bank withdrawal from this mode must meet the requirements of General Design Criteria 10 and 25 which specify that fuel design limits not be exceeded. Therefore, the staff requested the licensee to provide either a positive means of preventing control bank withdrawal or sufficient boration of the reactor coolant system to ensure adequate shutdown margin is maintained even if an inadvertent sequential control bank withdrawal event were to occur at these conditions. The licensee has accordingly modified Technical Specification 3.1.1.b to require the reactor to be tripped from the C-06 panel by opening the 42-01 and 42-02 circuit breakers if an out-of-service pump cannot be returned to service and the reactor is in hot shutdown (or below). This tripped condition can only be reset from outside of the control room (one floor below). The staff, therefore, considers this to be an acceptable means of providing additional assurance that a control bank withdrawal would not occur from these conditions.

Based on the above evaluations, the staff finds the changes proposed by the licensee adequately address the concerns of Generic Letter 86-13 and are acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment and Finding of No Significant Impact has been issued for this amendment (53 FR 45633).

6.0 CONCLUSION

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

Date: November 15, 1988

Principal Contributors:
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7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

NOTICE OF ISSUANCE OF AMENDMENT TO

PROVISIONAL OPERATING LICENSE

The United States Nuclear Regulatory Commission (the Commission) has issued Amendment No. 118 to Provisional Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised the Technical Specifications (TSs) for operation of the Palisades Plant, located in Van Buren County, Michigan. The amendment is effective as of the date of issuance.

The amendment revised the provisions in the Technical Specifications to add limitations to plant operation with less than four reactor coolant pumps operating to conform to analyses for certain postulated accidents. It also modified the limiting conditions for operation, reactor protection system setpoints, surveillance requirements, radial peaking factors and the linear heat rate limit. The currently issued Safety Evaluation also approved the reactor protection system modification proposed in the licensee's March 25, 1988 application and used in the accident and transient analyses submitted in support of the revised Technical Specifications. That part of the March 25, 1988 application that proposed a change to the differential pressure limit for the steam generators has been superseded by an application dated October 7, 1988, which will be addressed in a future action.

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The applications for the amendment comply 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.

Notices of Consideration of Issuance of Amendment to Provisional Operating License and Opportunity for Hearing in connection with this action were published in the FEDERAL REGISTER on July 6, 1988 (53 FR 25397) and September 27, 1988 (53 FR 37663). No request for hearing or petition to intervene was filed following these notices.

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on November 10,1988, at 53 FR 45633 .

For further details with respect to this action, see (1) the application for amendment dated March 25, 1988, as supplemented June 17 and 27, 1988, and November 9, 1988, and application dated September 1, 1988, as supplemented September 19, 1988, (2) Amendment No. 118 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, Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the Van Zoeren Library, Hope College, Holland, Michigan 49423. A copy of items (2) and (3) may be obtained upon request

addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects - III, IV, V, and Special Projects.

Dated at Rockville, Maryland, this 15th day of November 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

Thomas V. Wambach, Project Manager
Project Directorate III-I
Division of Reactor Projects - III, IV, V
& Special Projects

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UNITED STATES NUCLEAR REGULATORY COMMISSION

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

NOTICE OF ISSUANCE OF AMENDMENT TO

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