February 20, 1991

Docket No. 50-255

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Mr. Gerald B. Slade
Plant General Manager
Palisades Plant
Consumers Power Company
27780 Blue Star Memorial Highway
Covert, Michigan 49043

Dear Mr. Slade:

SUBJECT: AMENDMENT NO. 137 TO PROVISIONAL OPERATING LICENSE NO. DPR-20: (TAC NOS. 77576 AND 77670)

The Commission has issued the enclosed Amendment No. 137 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your applications dated August 31, and September 19, 1990, as amended October 3 and December 28, 1990.

This amendment will allow use of both the ANFP (Advanced Nuclear Fuels) DNB (departure from nucleate boiling) correlation for high thermal performance fuel and the XNB (Exxon Nuclear) DNB correlation for PWR fuel designs for the Cycle 9 fuel reload. This amendment also includes revisions to the reactor protective system set points, limiting conditions for operation (LCO), Bases, and references, which are required for Cycle 9 power operations. These changes are a result of changes to plant equipment, fuel design, and the fuel management scheme for Cycle 9.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Brian Holian, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 137 to License No. DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRP345	PM/PD31:DRP345	D/PD31:DRP345	OGC
PShuttleworth	BHolian	LMarsh	
2/6/91	2/6/91	2/21/91	2/15/91

PALISADES AMEND 77576/77670

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PDR ADOCK 05000255
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270090

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

February 20, 1991

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Mr. Gerald B. Slade
Consumers Power Company

Palisades Plant

cc:

M. I. Miller, Esquire
Sidley & Austin
54th Floor
One First National Plaza
Chicago, Illinois 60603

Mr. Thomas A. McNish, Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

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

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Jerry Sarno
Township Supervisor
Covert Township
36197 M-140 Highway
Covert, Michigan 49043

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Mr. David J. Vandewalle
Director, Safety and Licensing
Palisades Plant
27780 Blue Star Memorial Hwy.
Covert, Michigan 49043

Resident Inspector
c/o U.S. Nuclear Regulatory Commission
Palisades Plant
27782 Blue Star Memorial Hwy.
Covert, Michigan 49043

Nuclear Facilities and
Environmental Monitoring
Section Office
Division of Radiological
Health
P.O. Box 30035
Lansing, Michigan 48909

Gerald Charnoff, P.C.
Shaw, Pittman, Potts &
Trowbridge
2300 N. Street, N.W.
Washington, D.C. 20037

Mr. David L. Brannen
Vice President
Palisades Generating Plant
c/o Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, Maryland 20877

February 20, 1991

Docket No. 50-255

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PALISADES AMEND 77576/77670

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. 137
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 August 31, and September 19, 1990 as amended October 3 and December 28, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B. of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 137 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 20, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 137

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 amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-2	1-2
2-1	2-1
2-2	2-2
2-4	2-4
2-6	2-6
2-9	2-9
3-1b	3-1b
3-1c	3-1c
3-2	3-2
3-3	3-3
3-3a	3-3a
3-61	3-61
3-63	3-63
3-64	3-64
3-67	3-67
3-107	3-107
3-111	3-111
4-84	4-84

1.1 REACTOR OPERATING CONDITIONS (Contd)

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 therefore 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.

Total Interior Rod Radial Peaking Factor - F_r^T

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. Local peaking is defined as the maximum ratio of the power in an individual fuel rod to assembly average rod power.

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 the DNB correlation safety limit.

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 a DNB Correlation. DNB Correlations have 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 DNB correlation safety limit. A DNBR equal to the DNB correlation safety limit 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.

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 the DNB correlation safety limit. The DNB correlations used in the Palsiades safety analysis are listed in the following table.

Name	Safety Limit	References Correlation	Applicability
XNB	1.17	1	2
ANFP	1.154	4	5

The MDNBR analyses are performed in accordance with Reference 6.

References

(1) XN-NF-621(P)(A), Rev 1
(2) XN-NF-709
(3) Updated FSAR, Section 14.1.
(4) ANF-1224 (P)(A), May 1989
(5) ANF-89-192(P), January 1990
(6) XN-NF-82-21(A), Revision 1

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 maximun 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.

$$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9493$$

where:

$$\begin{array}{llll} QR_1 & = 0.412(Q) + 0.588 & Q \leq 1.0 & Q = \dfrac{\text{core power}}{\text{rated power}} \\ & = Q & Q > 1.0 & \end{array}$$

$$\begin{array}{lll} QA & = -0.720(ASI) + 1.028 & -0.628 \leq ASI < -0.100 \\ & = -0.333(ASI) + 1.067 & -0.100 \leq ASI < +0.200 \\ & = +0.375(ASI) + 0.925 & +0.200 \leq ASI \leq +0.565 \\ & = \ \ 1.085 \text{ when } Q < 0.0625 & \end{array}$$

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 $P_{trip} + \Delta P$.

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

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 115%, 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)

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

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 ($\geq$ 260°F) and low pressure ($\geq$ 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-90-078, Table 15.0.7-1
(2) deleted
(3) Updated FSAR, Section 7.2.3.3.
(4) ANF-90-078, Section 15.0.7-1
(5) XN-NF-86-91(P)
(6) deleted
(7) deleted
(8) ANF-90-078, Section 15.1.5
(9) ANF-87-150(NP), Volume 2, Section 15.2.7
(10) Updated FSAR, Section 7.2.3.9.
(11) ANF-90-078, Section 15.2.1

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 shotdown, 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 ont 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 permited and power operation with less than three pumps is not permitted.

c. The measured four primary coolant pumps operating reactor vessel flow shall be 140.7×10^6 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) Deleted

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:

$$T_{inlet} \leq 542.99 + .0580(P-2060) + 0.00001(P-2060)^{**}2 + 1.125(W-138) - 0.0205(W-138)^{**}2$$

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 150 M lbm/hr, the maximum inlet temperature shall be less than or equal to the T_{Inlet} LCO at 150 M lbm/hr.

Basis (Cont'd)

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

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

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

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

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

(3) Acceptable stresses during accident conditions.

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

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

The transient analyses were performed assuming a vessel flow at hot zero power (532°F) of 140.7 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 the DNB correlation safety limit. This analysis includes the following uncertainties and allowances: 2% of rated power for power

Basis (Cont'd)

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

$$T_{inlet} \leq 542.99 + .0580(P-2060) + 0.00001(P-2060)^{**}2 + 1.125(W-138) - .0205(W-138)^{**}2$$

The limits of validity of this equation are:

$1800 \leq$ pressure ≤ 2200 psia
$100.0 \times 10^6 \leq$ Vessel Flow $\leq 150 \times 10^6$ lb/h
ASI as shown in Figure 3.0

With measured primary coolant system flow rates > 150 M lbm/hr, limiting the maximum allowed inlet temperature to the T_{Inlet} LCO at 150 M lbm/hr increases the margin to DNB for higher PCS flow rates.

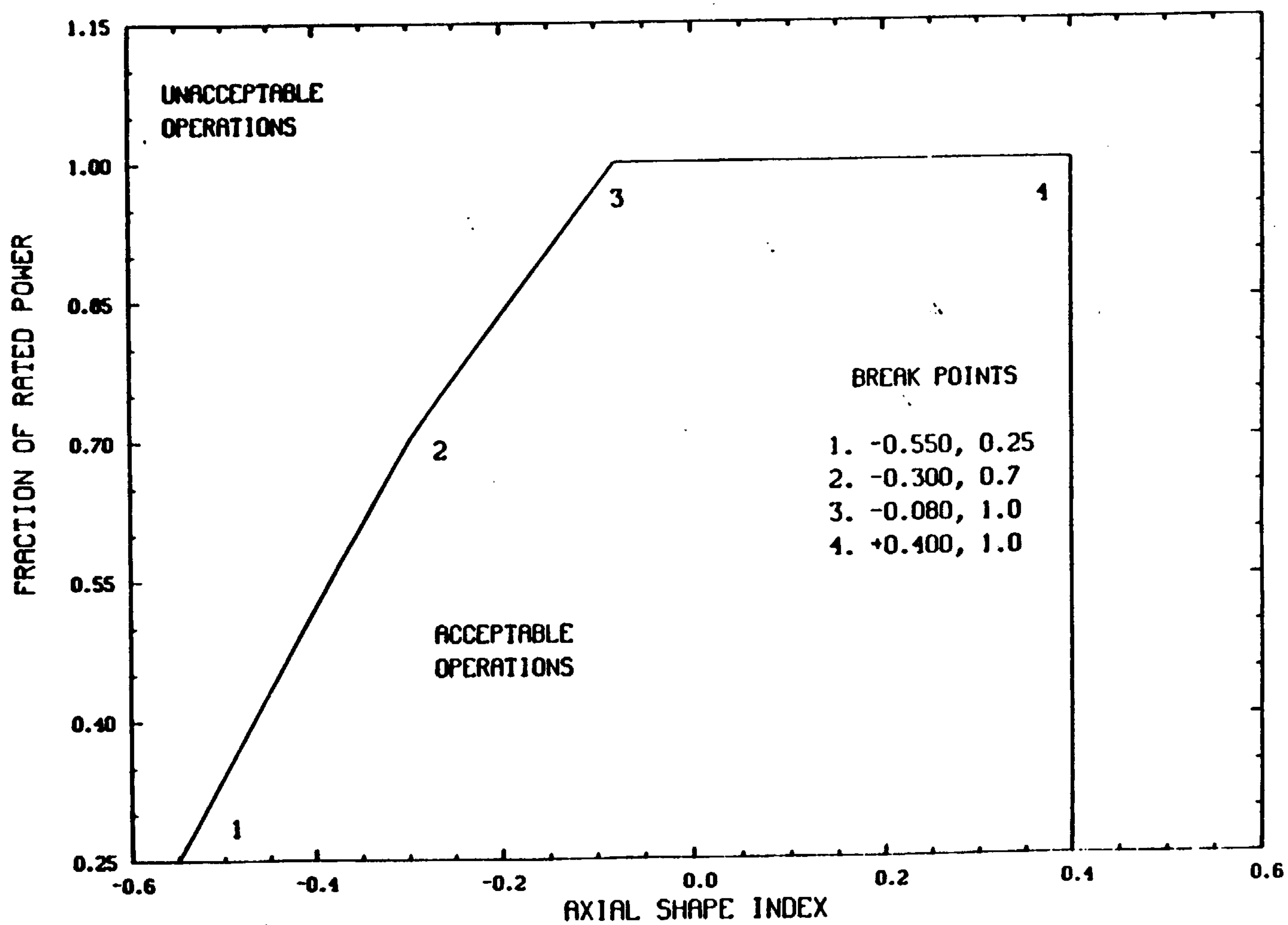
The Axial Shape Index alarm channel is being used to monitor the ASI to ensure that the assumed axial power profiles used in the development of the inlet temperature LCO bound measured axial power profiles. The signal representing core power (Q) is the auctioneered higher of the neutron flux power and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

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

References

(1) Updated FSAR, Section 14.3.2.
(2) Updated FSAR, Section 4.3.7.
(3) Palisades 1983/1984 Steam Generator Evaluation and Repair Program Report, Section 4, April 19, 1984
(4) ANF-90-078 Section 15.0.7.1
(5) ANF-90-078
(6) Consumers Power Company Engineering Analysis EA-A-NL-89-14-1

FIGURE 3-0
ASI LCO FOR Tinlet FUNCTION



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 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.[1] The change in insertion limit with reactor power shown on Figure 3-6 insures that the shutdown

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.[1]

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) ANF-90-078

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 $+0.5 \times 10^{-4}\ \Delta\rho/^{\circ}F$ 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-90-078, Section 15.0.5

(next page is 3-69)

TABLE 3.23-1

LINEAR HEAT RATE LIMITS

	No. of Fuel Rods 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	1.57
Peak Rod F_r^T	1.92	1.92

3.23.2 RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

The radial peaking factors F_r^A, and F_r^T 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 ≥ .5 and the quantity 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 ≥ 50% of rated with any radial peaking factor exceeding its limit, reduce thermal power within 6 hours to less than the lowest value of:

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

Where F_r is the measured value of either F_r^A, or F_r^T and F_L is the corresponding limit from Table 3.23-2.

Basis

The limitations on F_r^A, and F_r^T 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.

The radial peaking is limited to those values used in the LOCA analysis. Since the LOCA analysis limits the magnitude of radial peaking, Table 3.23-2 explicitly contains these limits.

4.19.2 RADIAL PEAKING FACTORS

SURVEILLANCE REQUIREMENTS

4.19.2.1 The measured radial peaking factors (F_r^A, and F_r^T) 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO CYCLE 9 RELOAD

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letters dated August 31, (Ref. 1), and September 19, 1990, as amended October 3, and December 28, 1990 (Ref. 2), Consumers Power Company (CPC) submitted proposed Technical Specification (TS) changes to Facility Operating License DPR-20 for Cycle 9 operation of the Palisades Plant. The October 3 and December 28, 1990, submittals provided additional information and a revised Technical Specification page, and did not change the determination of no significant hazards or alter the proposed action. The evaluation of the changes for Cycle 9 operation is provided by the Advanced Nuclear Fuels Corporation (ANF) safety analysis report and evaluation of Standard Review Plan (SRP), Chapter 15 events (Refs. 3 and 4).

2.0 EVALUATION

2.1 Palisades Cycle 9 Safety Analysis Report

The results of the safety evaluation for Cycle 9 of the Palisades nuclear plant are presented in ANF-90-076 (Ref. 3). The Palisades Cycle 9 reload will consist of 52 Reload M assemblies at an average enrichment of 2.69 w/o U-235. Twenty of the Reload M assemblies will each contain 12 gadolinia-bearing fuel rods containing 6.0 w/o Gd_2O_3, 8 of the M assemblies will each contain 8 rods containing 4.0 w/o Gd_2O_3, 16 of the M assemblies will each contain 4 rods containing 4.0 w/o Gd_2O_3, and 8 of the M assemblies will each contain an asymmetric loading of 3 rods each containing 4.0 w/o Gd_2O_3. The fuel assembly design for Reload M is similar to the four Reload L High Thermal Performance (HTP) assemblies loaded in Cycle 8. Reload M fuel rods differ from those of the Reload L design primarily in enrichment. The fuel pellet density and pellet diameter are slightly increased for Reload M. The Reload L HTP design is similar to the Reload H extended burnup design.

Cycle 9 uses a low radial leakage loading plan similar to the design developed for Cycle 8. This Cycle 9 plan differs in its use of hafnium inserts in selected peripheral assemblies, rather than assemblies reconstituted with stainless steel rods. This use of the hafnium combined with the low radial leakage design reduces the neutron fluence on critical pressure vessel welds.

Cycle 9 of the Palisades nuclear plant is designed to operate at 2,530 MWt.

Plant safety analyses, both LOCA and non-LOCA, have been performed to support Palisades Cycle 9 operation. These analyses support full power operation with:

- The new steam generators with up to 15% tube plugging
- An F^A_r limit of 1.57 (216 rods/assembly) and 1.48 (208 rods/assembly)
- An F^T_r limit of 1.92 for all assemblies
- The F^T_r limit replaces the current $F\Delta H_r$ Technical Specification limit
- 16 assemblies with hafnium inserts

The increase in the radial peaking is required to support the low radial leakage reload pattern.

Palisades Cycle 8 has been chosen as the reference neutronics cycle due to the close resemblance of the overall neutronic characteristics of Cycle 9 and Cycle 8.

The results contained in ANF-90-076 show that operation at a power level of 2,530 MWt with the increased radial peaking can be safely achieved. A comparison summary of the Cycle 9 plant parameters to the core license limits are given in Table 2.1 (Ref. 3). Analyses to support increased burnup levels for ANF fuel have also been performed and a summary of the Cycle 9 maximum pellet exposures is shown in Table 2.2 (Ref. 3) and is compared to the exposure limits of these analyses.

The staff has reviewed these analyses and has determined that they were performed with NRC approved ANF methodology and are acceptable. Also, The use of hafnium inserts in the low power Reload I fuel is acceptable for Cycle 9.

2.2 Transient and Accidents - Analysis of Standard Review Plan Chapter 15 Events

ANF-90-078 (Ref. 4) documents the results of a FSAR Chapter 14 plant transient analysis performed in support of Palisades Cycle 9 operation with up to 15.0% steam generator tube plugging. The events were selected in accordance with Chapter 15 of the Standard Review Plan (SRP) and Advanced Nuclear Fuels (ANF) Corporation methodology (Ref. 5 & 6). The basis for event selection is documented in the Disposition of Events report (Ref. 7) and the events analyzed are shown in Table 1.

ANF-90-078 presents the conditions employed in the event analyses and the results of these event analyses. It includes a tabular list of the disposition of events and analysis of record for Palisades, Chapter 15 events, with a cross reference between SRP event numbers and the Palisades Updated FSAR. The staff has reviewed these analyses and finds them to be acceptable.

A bounding MDNBR analysis was performed for Cycle 9 that is based on the radial peaking factors given in Table 15.0.3-1 (Ref. 4). The bounding analysis consisted of evaluating the MDNBR's for Reload L fuel with standard bi-metallic spacers and radial peaking factors of 1.66 and 2.03 for the assembly and peak rod, respectively. The MDNBR's for the standard bi-metallic spacer fuel (Reload L) were evaluated with the XNB critical heat flux correlation. Confirmatory MDNBR calculations were performed for Reload M fuel with HTP spacers and radial peaking factors of 1.57 and 1.92 for the assembly and peak rod, respectively.

The MDNBR's for the HTP spacer fuel (Reload M) were evaluated with the ANFP critical heat flux correlation. The Cycle 9 Technical Specifications on radial peaking for both Reload L and M fuel will be limited to 1.57 and 1.92 for the assembly and peak rod, respectively. This is acceptable to the staff.

TABLE 1

EVENTS REQUIRING REANALYSIS

Increase in Heat Removal by the Secondary System

15.1.3 Increase in Steam Flow

Decrease in Heat Removal by the Secondary System

15.2.1 Loss of External Load

Decrease in Reactor Coolant System Flow

15.3.1 Loss of Forced Reactor Coolant Flow
15.3.3 Reactor Coolant Pump Rotor Seizure

Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Startup Condition
15.4.2 Uncontrolled Control Rod Bank Withdrawal at Power
15.4.3 Control Rod Misoperation
(1) Dropped Control Rod/Bank
(4) Statically Misaligned Control Rod/Bank
(5) Single Control Rod Withdrawal
(6) Core Barrel Failure
15.4.6 CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant
15.4.8 Spectrum of Control Rod Ejection Accidents

Decreases in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve
15.6.3 Radiological Consequences of Steam Generator Tube Failure

2.3 Technical Specification Changes

The proposed amendment would modify the Palisades Plant Technical Specifications (TS) as follows:

In TS Section 1.1, the definition of "Interior Fuel Rod" is deleted since it is no longer used. Accordingly, all references to "interior" are deleted from the definition of "Total Interior Rod Radial Peaking Factor". Total Rod Radial Peaking Factor "F_r^{Δ}" is renamed "F_r^T" and a "local peaking" definition is added to F_r^T for clarification.

In TS Section 2.1, basis and references, a sentence is added stating that the analysis is performed in accordance with the added references 6. This change is necessary since Cycle 9 has a mixed core arrangement and a new Minimum Departure from Nucleate Boiling Ratio (MDNBR) analysis is required.

The "P_{var}" equation in TS section 2.3 is revised to reflect the update to the constants for the Thermal Margin/Low Pressure (TM/LP) equation for Cycle 9. In the basis, the Variable High Power Trip (VHPT) setpoint is changed from "112% to "115%" as a result of the uncertainty on the VHPT setpoint being increased from 5.5% to 8.5% for Cycle 9.

The references 1, 4, 8 and 11 are replaced with an updated Cycle 8 basis report and the reference 12 is deleted.

TS Section 3.1, and its associated basis and references, is change to incorporate a new primary coolant pump measured flowrate in order to reflect the associated lower assumed tube plugging ratio of the replacement steam generators. The T_{inlet} equation, and the associated basis, is changed to reflect this increased margin. Also Figure 3-0, "ASI LCO for T_{inlet} Function", is replaced with an updated Figure due to an expanded Axial Shape Index (ASI) Limiting Condition for Operation (LCO) operating window between 70% and 25% power. Finally, references 4 and 5 are replaced with one Cycle 9 specific references.

TS Section 3.10, basis and references, and TS Section 3.12, references, have been revised with the appropriate references to reflect the changes in the Cycle 9 analysis.

TS Tables 3.23-2, TS Section 3.32.2 and its associated basis, and TS Section 4.19.2.1, similarly replaced $F_r^{\Delta H}$ with F_r^T for the reasons listed above. Also, in TS Table 3.23-2, the radial peaking factor limits are revised with values consistent with the cycle 9 analysis. Finally, a paragraph is added in the basis of Section 3.23-2 explaining the Loss of Coolant Accident Analysis Limits.

The majority of the Technical Specification Changes proposed for Cycle 9 are either editorial in nature or are made to support Cycle 9 operation according to previously staff approved ANF methodology and, therefore, are acceptable.

The following two changes do not fall into these categories and the staff focused on these changes as part of the review:

1. The licensee proposed to replace the INCA method of analyzing in-core power detector data with their new PIDAL software.

2. The licensee proposed to extend the previously approved ANFP DNB correlation for High Thermal Performance Fuel in Palisades Reactor (Ref. 1 & 8).

The staff is currently performing an in-depth review of the PIDAL Code which will not be completed in time for the start-up of Cycle 9 operation. Therefore, the staff will require that the previously approved INCA methodology be continued during Cycle 9 operation.

ANF-89-192(P) demonstrates that the ANFP DNB correlation may be used to predict the DNB heat flux in the 15x15 High Thermal Performance (HTP) fuel design for Palisades. The Palisades design is similar to those represented in the ANFP correlation data base in most key design characteristics, but differs in having a shorter heated length and no interior guide tubes. The ANFP correlation is shown to predict low DNB heat fluxes for ANF DNB Tests 56 and 57, which together represent these two special features of the Palisades design. Use of the ANFP correlation thus results in a conservative estimation of the DNB heat flux for the Palisades design.

For both tests, the data presented in ANF-89-192(P) indicate that on average the predicted heat flux at which DNB occurs is less than that measured. This is conservative. Also, the tolerance limits lie below the 1.154 safety limit of the ANFP correlation. Hence, the ANFP correlation and its associated safety limit of 1.154 provide a conservative assessment of the DNB performance for Tests 56 and 57 and its use for analyses of the Palisades Cycle 9 reload core is acceptable to the staff.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 SUMMARY

The staff has reviewed the proposed modifications to the Palisades Technical Specifications and reload fuel configuration for Cycle 9 operation and finds them acceptable provided that the INCA method of analyzing in-core power detector data is retained.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change in a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff, therefore, concludes that the proposed changes are acceptable.

Principal Contributors: B. Holian
G. Schwenk

Date: February 20, 1991

REFERENCES

1. Letter, G. B. Slades (CP) to NRC, "Palisades Plant - Technical Specifications Change Request - DNB Correlation," dated August 31, 1990

2. Letters, G. B. Slade (CP) to NRC "Palisades Plant - Cycle 9 Technical Specification Change Requests," dated September 19, October 3 and December 28, 1990.

3. K. C. Segard, "Palisades Cycle 9 Safety Analyses Report," ANF-90-076, dated September 19, 1990.

4. T. R. Lindquist, "Palisades Cycle 9: Analysis of Standard Review Plan Chapter 15 Events," dated September 25, 1990.

5. "Advanced Nuclear Fuels Corporation Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," ANF-84-73(P), Revision 4, Appendix B, Advanced Nuclear Fuels Corporation, August 1989.

6. Letter, Mr. Ashok C. Thadani (USNRC) to Mr. R. A. Copeland (ANF), dated July 13, 1990, "Acceptance for Referencing of Topical Report ANF-84-73(P), Revision 4, Appendix B, 'Advanced Nuclear Fuels Corporation Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events.'"

7. "Disposition of Standard Review Plan Chapter 15 Events for Palisades Cycle 9, "ANF-90-041, Revision 2, Advanced Nuclear Fuels Corporation, September 19, 1990.

8. F. T. Adams, "Justification of the ANFP DNB Correlation for High Thermal Performance Fuel in the Palisades Reactor," dated January 17, 1990.