Docket No. 50-255

March 27, 1992

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

Dear Mr. Slade:

SUBJECT: PALISADES PLANT - AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-20 (TAC NO. M82059)

The Commission has issued the enclosed Amendment No. 143 to Facility Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated November 1, 1991.

This amendment revises the Palisades Technical Specifications in support of Cycle 10 operations. Additionally, changes are allowed to the upper limit of the boron concentration for the safety injection tanks and the refueling water storage tank.

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. 143 DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

BC:PRPB
LJ CUNNINGHAM
3/13/92

OFC	:LA:PDIII-1	:PM:PDIII-1	:SRXB	:PRPB	:OGC	:D:PD3-1
NAME	:MShuttleworth	:BHolian	:SBrewer	:KEccleston:	:	:LMarsh
DATE	: 3/18/92	: 3/16/92	: 3/13/92:	3/13/92	: 3/24/92	: 3/27/92

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

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

Enclosures:
1. Amendment No.143DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

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

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. Patrick M. Donnelly
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
Monitor Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 30195

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 Company
c/o Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, Maryland 20877

Roy W. Jones
Manager, Strategic Program
Development
Westinghouse Electric Corporation
4350 Northern Pike
Monroeville, Pennslyvania 15146

DATED: March 27, 1992

AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-20-PALISADES

Docket File
NRC & Local PDRs
PDIII-1 Reading
Palisades Plant File
B. Boger
J. Zwolinski
L. Marsh
M. Shuttleworth
B. Holian OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
K. Eccelston 10/D/4
ACRS (10)
S. Brewer 8/E/23
GPA/PA
OC/LFMB
W. Shafer, R-III

cc: Plant Service list

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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated November 1, 1991, 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;

   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 the license amendment and Paragraph 2.C.2 of Facility 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. 143 , 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

William O. Long

for 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: March 27, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 143

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

REMOVE	INSERT
1-2	1-2
2-4	2-4
2-9	2-9
3-3	3-3
3-3a	3-3a
3-29	3-29
3-31	3-31
3-66b	3-66b
3-67	3-67
3-105	3-105
3-107	3-107
3-111	3-111

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 Radial Peaking Factor - F_r^T

The maximum product of the ratio of individual assembly power to core average assembly power times the highest local peaking factor integrated over the total core height, including tilt. Local peaking is defined as the maximum ratio of an individual fuel rod power to the assembly average rod power.

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.

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

where:

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

$\quad\;\; = Q \quad Q > 1.0$

$ASI = 0$ when $Q < 0.0625$

$QA = -0.720(ASI) + 1.028$ when $-0.628 \leq ASI < -0.100$

$\quad\;\; = -0.333(ASI) + 1.067$ when $-0.100 \leq ASI < +0.200$

$\quad\;\; = +0.375(ASI) + 0.925$ when $+0.200 \leq ASI \leq +0.565$

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 $P_{trip} + \Delta P$.

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 shutdown 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) EMF-91-176, Table 15.0.7-1
(2) deleted
(3) Updated FSAR, Section 7.2.3.3.
(4) EMF-91-176, 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 (Cont'd)

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:

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

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.

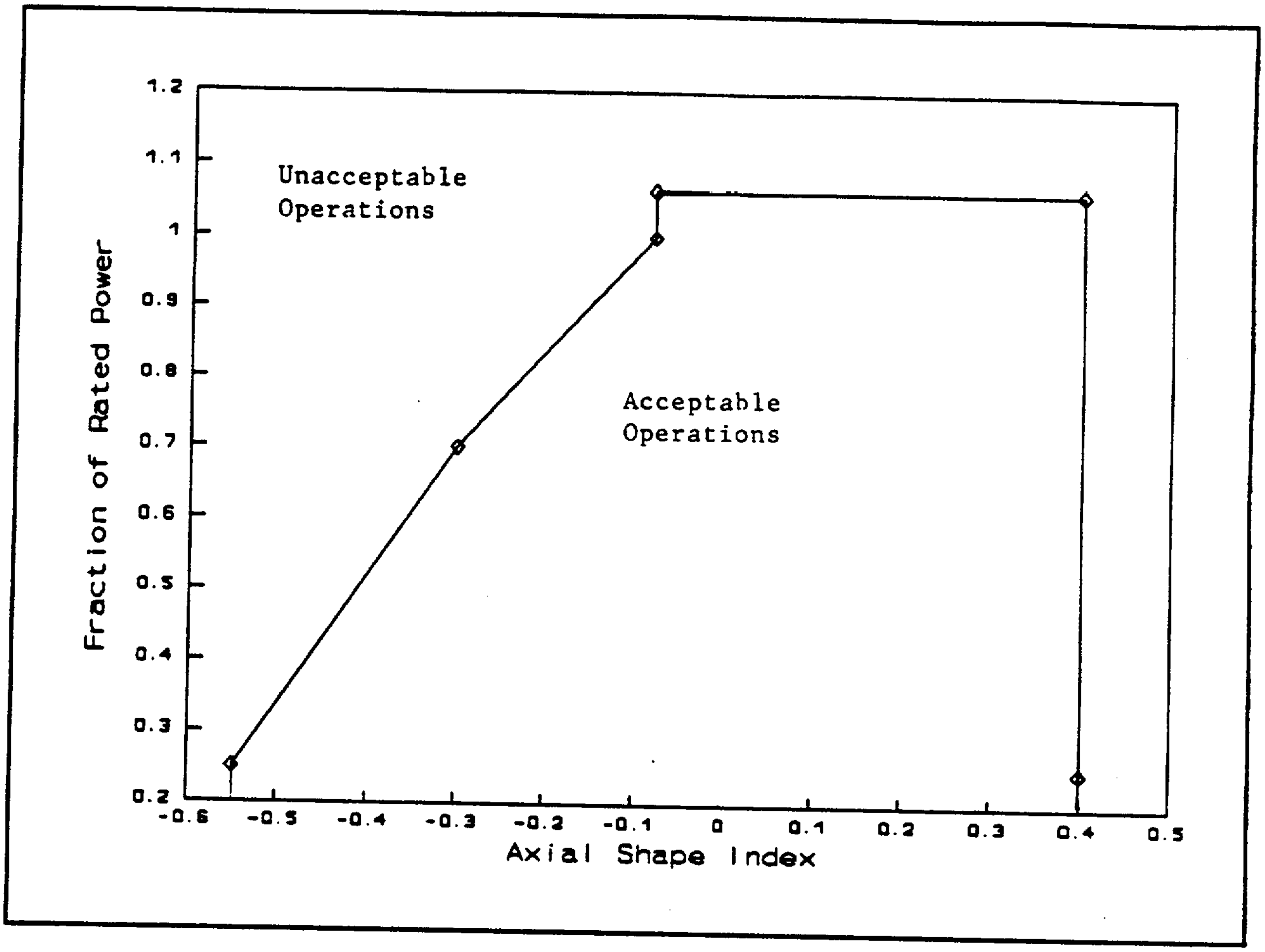
The Axial Shape Index alarm channel is being used to monitor the ASI to ensure that the assumed axial power profiles used in the development of the inlet temperature LCO bound measured axial power profiles. The signal representing core power (Q) is the auctioneered higher of the neutron flux power and 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) Deleted
(4) EMF-91-176 Section 15.0.7.1
(5) ANF-90-078
(6) Consumers Power Company Engineering Analysis EA-A-NL-89-14-1

ASI Limit for T_{inlet} function



Break Points:

-0.550, 0.250
-0.300, 0.700
-0.080, 1.000
-0.080, 1.065
+0.400, 1.065
+0.400, 0.250

FIGURE 3-0

3.3 EMERGENCY CORE COOLING SYSTEM

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core in either emergency or normal shutdown situations.

Specifications

Safety Injection and Shutdown Cooling Systems

3.3.1 The reactor shall not be made critical, except for low-temperature physics tests, unless all of the following conditions are met:

a. The SIRW tank contains not less than 250,000 gallons of water with a boron concentration of at least 1720 ppm but not more than 2500 ppm at a temperature not less than 40°F.

b. All four Safety Injection tanks are operable and pressurized to at least 200 psig with a tank liquid level of at least 174 inches and a maximum level of 200 inches with a boron concentration of at least 1720 ppm but not more than 2500 ppm.

c. One low-pressure Safety Injection pump is operable on each bus.

d. One high-pressure Safety Injection pump is operable on each bus.

e. Both shutdown heat exchangers and both component cooling heat exchangers are operable.

f. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the primary cooling system.

g. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable.

h. The Low-Pressure Safety Injection Flow Control Valve CV-3006 shall be opened and disabled (by isolating the air supply) to prevent spurious closure.

i. The Safety Injection bottle motor-operated isolation valves shall be opened with the electric power supply to the valve motor disconnected.

j. The Safety Injection miniflow valves CV-3027 and 3056 shall be opened with HS-3027 and 3056 positions to maintain them open.

c. If Specification a. and b. cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown condition within 12 hours, and cold shutdown within the next 24 hours.

Basis

The normal procedure for starting the reactor is, first, to heat the primary coolant to near operating temperature by running the primary coolant pumps. The reactor is then made critical by withdrawing control rods and diluting boron in the primary coolant.[1] With this mode of start-up, the energy stored in the primary coolant during the approach to criticality is substantially equal to that during power operation and, therefore, all engineered safety features and auxiliary cooling systems are required to be fully operable. During low-temperature physics tests, there is a negligible amount of stored energy in the primary coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards' systems are not required.

The SIRW tank contains a minimum of 250,000 gallons of water containing a minimum of 1720 ppm boron and a maximum of 2500 ppm. This is sufficient boron concentration to provide a 5% shutdown margin with all control rods withdrawn and a new core at a temperature of 60°F.

Heating steam is provided to maintain the tank above 40°F to prevent freezing. The 1.43% boron (2500 ppm) solution will not precipitate out above 32°F. The source of steam during normal plant operation is extraction steam line in the turbine cycle.

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 174-inch level corresponds to a volume of 1040 ft^3 and the maximum 200-inch level corresponds to a volume of 1176 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the

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:

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

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) EMF-91-177

*Corrected

(next page is 3-66d)

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) EMF-91-176, Section 15.0.5

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) EMF-91-177

(2) (Deleted)

(3) (Deleted)

(4) XN-NF-80-47

(next page is 3-107)

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 (Reload M and earlier)	216
Assembyl F_r^A	1.48	1.57	1.66
Peak Rod F_r^T	1.92	1.92	1.92

POWER DISTRIBUTION LIMITS

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 LOCA analysis supports the radial peaking factor limits in Table 3.23-2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated November 1, 1991, Consumers Power Company submitted a proposal to amend the Technical Specification (TS) to Facility Operating License DRP-20 for Cycle 10 operation of the Palisades Plant. The evaluation for Cycle 10 operation is provided in the Siemens Nuclear Power Corporation (SNP) report EMF-91-176 entitled, "Palisades Cycle 10: Disposition and Analysis of Standard Review Plan Chapter 15 Events."

The report documents the results of the disposition and analysis of the FSAR Chapter 14 events in support of Palisades Cycle 10 operation with up to 15.0% steam generator tube plugging. The events were evaluated in accordance with Chapter 15 of the Standard Review Plan (SRP) and SNP methodology. The changes that are proposed to be implemented for Cycle 10 include (1) the insertion of the second full reload of fuel that uses High Thermal Performance (HTP) grid spacers; (2) an increase in assembly radial power peaking to accommodate a low radial leakage loading pattern; (3) the inclusion of eight partial shielding assemblies (PSA) in low powered peripheral locations to reduce vessel fluence; (4) Reactor Protection System set point modifications (FC-888); and (5) Main Feedwater Control upgrade (FC)-920.

The large break loss-of-coolant accident (LBLOCA) analysis is summarized in the SNP report EMF-91-177, entitled "Palisades Large Break LOCA/ECCS Analysis with Increased Radial Peaking and Reduced ECCS Flow." The analysis supports the following primary changes:

- A reduction in emergency core cooling system (ECCS) flow due to a change in the Low Pressure Safety Injection (LPSI) flow curve and the assumed loss of a High Pressure Safety Injection (HPSI) pump, along with a LPSI pump when the worst single failure is considered (i.e., one emergency diesel generator).

- To bound future cycles an assembly radial peaking limit of 1.76 and a peak rod radial peaking limit of 2.04 were used.

- A 5 mil increase in the pellet diameter (i.e., reduction in the pellet-to-clad gap)

- An increase in pellet density to 94.5% of the theoretical density.
- An increase in minimum Technical Specification safety injection tank (SIT) level.

2.0 EVALUATION

Cycle 10 of the Palisades Plant is designed to operate at 2530 MWt. The plant safety analyses, both LOCA and non-LOCA have been performed to support Palisades Cycle 10 operation with:

- Steam generators with up to 15% tube plugging.
- A fuel rod peaking factor limit of 1.92 for all fuel assembly types.
- An increase in assembly peaking factor limit from 1.57 for reload M to 1.66 for reload N (216 fuel rods per assembly) to accommodate a low radial leakage loading pattern. The peaking factor limit for 208 fuel rods per assembly, reload L, remains the same at 1.48.
- Inclusion of eight partial shielding assemblies (PSA) in low powered peripheral locations to reduce vessel fluence.
- Modifications in the Reactor Protection System set points.
- Upgrades to the Main Feedwater Controller.

The changes specific to Cycle 10 operation are necessary due to the changes to the fuel design (reload N) and the fuel management scheme for a low leakage core. For Cycle 10 only minimum DNBR calculations were performed. The Cycle 9 transient analysis (ANF-90-078, (Ref. 5)) still bounds the thermal hydraulic response for events identified as requiring DNB analysis. The operating parameters as described in sections 15.0.1 through 15.0.8 of EMF-91-176 remain applicable for the Cycle 10 analysis relative to the Cycle 9 analysis.

Several factors offset the loss in DNB margin from the increased assembly radial peaking. They include: (1) use of the Advanced Nuclear Fuels (ANFP), critical heat flux correlation, (2) use of the HTP spacer fuel, (3) improved reload N specific fuel design, and (4) less limiting axial shape characteristic of full power control rod position.

<u>Non-LOCA Transient Analysis</u>

The basis for event selection is documented in the Disposition and Analysis of Events report (Ref. 3). Listed below are the SRP Chapter 15 events that were reanalyzed for the Cycle 10 submittal:

Increase in Heat Removal by the Secondary System

15.1.3 Increase in Steam Flow

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

15.4.2 Uncontrolled Control Rod Bank Withdrawal at Power Operation Conditions
15.4.3 Control Rod Misoperation
(1) Dropped Control Bank/Rod
(5) Single Control Rod Withdrawal

Decreases in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve

The events that were reanalyzed were all found to meet the staff's acceptance criteria of no centerline melt and no DNB in the hottest fuel rod.

LOCA Analysis

The changes for Cycle 10 will not affect the relative severity between the LBLOCA and small break LOCA (SBLOCA). The licensee reviewed the significant parameters for SBLOCA listed in the FSAR for Palisades. The review indicated that the parameters assumed the reference SBLOCA analysis bound the corresponding values for Cycle 10.

The Cycle 10 LBLOCA analysis was performed assuming that the Palisades plant was operating at 2582 MWt (2530 MWt + 2% uncertainty) and incorporates a maximum average steam generator tube plugging level of 29.3% with up to 4.5% asymmetry in the system blowdown, hot channels, and reflood calculations. The changes supported by this analysis do not affect the limiting break size identified by SNP's LOCA methodology since the changes do not effect system blowdown. Therefore, the break limiting size of a 0.6 double ended guillotine break (DECLG) at the pump discharge, as previously identified, for Cycle 8 analysis was used.

The results of the Cycle 10 analysis demonstrate that the 10 CFR 50.46 criteria are met for Palisades plant with the axially dependent power peaking limit curve in Figure 2.1 of EMF-91-177. The analysis also supports a maximum linear heat ratio (LHR) of 15.28 kw/ft up to a relative core height of 0.6 and a LHR of 14.75 kw/ft up to a relative core height of 0.8. A total radial peaking factor of 2.04 and a maximum average steam generator tube plugging of 29.3% with up to 4.5% asymmetry are supported. The peak cladding temperature was calculated to be 1926.5 degrees F for the beginning of cycle (BOC) profile and 2110.6 degrees F for the end-of-cycle (EOC) profile. The analysis supports Cycle 10 operation and the staff finds this acceptable.

Safety Injection Boron Concentration

The licensee completed an analysis of post LOCA long term cooling to determine the effect of raising the boron concentration limit for the SITs and the Safety Injection and Refueling Water (SIRW) tank from 2000 ppm to 2500 ppm.

Since several plant parameters of the previous analysis have been changed or will be changed with the proposed increase in boron concentration. These changes are:

- An increase in the SITs level from 198" to 200," corresponding to a total liquid inventory increase of approximately 2308 lbm.
- An increase in the boron concentration limit from 2000 ppm to 2500 ppm, corresponding to an increase from 1.13 wt % boric acid to 1.43 wt % boric acid.
- The Technical Specification concentration limit of the boric acid storage tank (BAST) is 10 wt % where as 12 wt % was used in the current longterm cooling (LTC) calculation.
- Installation of new power operated relief valves (PORVs) with larger effective throat area that are used for long term cooling following SBLOCA.

The available margin in boric acid concentration from the LTC analysis for both large break and small break LOCAs was evaluated for the effect from increasing the boric acid concentration of the SITs and the SIRW tank by 0.3 wt%. The effect of increasing the boric acid limit of the SITs and SIRW tank from 2000 ppm to 2500 ppm is an increase of 0.2 wt% boric acid in the containment sump. This is insignificant when compared to excess margin available. A large excess margin stems from conservative assumptions in the analysis regarding the period of injection from the high concentration BAST (12%) versus the period of injection from the lower concentration SIRW and SIT (1.43%). The staff has reviewed the licensee's analysis and finds the increase in the boron concentration to 2500 ppm acceptable.

Fuel Handling Design Basis Accident

Since the licensee plans to use extended burnup fuel enriched to greater than 4.0 w/o U_{235}, the staff reanalyzed the fuel handling design basis accident (DBA) for this case. As noted in NUREG/CR-5009 "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988, increased burnup could increase offsite doses from the fuel handling accident by a factor of 1.2 due to the fact that the calculated iodine gap-release fraction for some high power fuel designs is increased by 20%.

Thus, the staff conservatively assumed an increased gap fraction of 0.12 as compared to the previously assumed gap-release fraction of 0.10 for iodine for the spent fuel handling accident.

The spent fuel assembly drop consequences analyzed in the Palisades SER were previously calculated by the staff to be 9 rem (thyroid) at the exclusion area boundary. With the 20% increase in radioiodine gap activity described in NUREG/CR-5009, the calculated radiological consequences at the exclusion area boundary would increase to 10.8 rem thyroid. The resultant calculated thyroid dose of 10.8 rem is well within the guideline values of 10 CFR Part 100 and meets the acceptance criterion of SRP 15.7.4, "Radiological Consequences of Fuel Handling Accidents," that calculated doses should be well within the guideline values of 10 CFR Part 100. The staff finds the TS changes proposed by the licensee, with respect to the radiological aspects of the planned changes, acceptable.

3.0 CONCLUSION

The staff has reviewed the modifications to the Palisades Technical specifications and the reload configurations for Cycle 10 and finds them acceptable.

The staff has also reviewed the licensee's proposal to increase the boron concentration, from 2000 ppm to 2500 ppm, in the SITs and the SIRW tank. We find that the increase is insignificant when considering the available margin and the conservatism incorporated. This change will not compromise the capability for post LOCA long term cooling and is, therefore, acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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. 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 (56 FR 64653). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

Principal Contributor: S. Brewer
K. Eccleston

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7.0 REFERENCES

1. Letter from G.B. Slade, Consumers Power to USNRC, "Palisades Plant - Technical Specifications Change Request - Cycle 10 Fuel Design and Safety Injection Boron Concentration," November 1, 1991.

2. "Palisades Large Break LOCA/ECCS Analysis with Increased Radial Peaking and Reduced ECCS Flow," EMF-91-177, Siemens Nuclear Power Corporation, Richland, WA 99352, October 1991.

3. "Palisades Cycle 10: Disposition and Analysis of Standard Review Plan Chapter 15 Events," EMF-91-176, Siemens Nuclear Power Corporation, Richland, WA 99352, October 1991.

4. "Effect of Increased SIRW Tank Boron Concentration on Long Term Cooling," EA-PAH-91-04, Consumers Power Corporation, November 1, 1991.

5. "Palisades Cycle 9: Analysis of Standard Review Plan Chapter 15 Events," ANF-90-078, Advanced Nuclear Fuels Corporation, September 1990.