June 16, 1993

Docket No. 50-255

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. 156 TO FACILITY OPERATING LICENSE NO. DPR-20 (TAC NO. M84793)

The Commission has issued the enclosed Amendment No. 156 to Facility Operating License No. DPR-20 for the Palisades Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated January 29, 1993, as supplemented April 20, 1993.

The amendment revises the Palisades TS Table 3.23-2, Radial Peaking Factor Limits, to add limits for those new fuel bundles to be installed during the 1993 Cycle 11 refueling outage. In addition, the bases for several Specifications (2.1, 2.3, 3.1, 3.12, and 3.23.2) have been updated to reflect the revision of analytical reports for Cycle 11.

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,

ORIGINAL SIGNED BY

Anthony H. Hsia, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 156 to DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PM:PD31	SRXB	OGC	D:PD31
NAME	CMoore	AHsia:sw	RJones	S. Hom	LMarsh
DATE	6/1/93	6/1/93	6/2/93	6/9/93	6/16/93

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Mr. Gerald B. Slade
Consumers Power Company

Palisades Plant

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AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. DPR-20-PALISADES

Docket File
NRC & Local PDRs
PDIII-1 plant file
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OGC-WF
D. Hagan, 3302 MNBB
G. Hill (2), P-137
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 156
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated January 29, 1993, as supplemented April 20, 1993, 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 Appendix A, as revised through Amendment No. 156 , and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall
for

Ledyard 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: June 16, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 156

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 amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
B 2-1	B 2-1
B 2-5	B 2-5
3-3	3-3
3-67	3-67
3-107	3-107
3-111	3-111

2.0 BASIS - Safety Limits and Limiting Safety System Settings

2.1 Basis - Reactor Core Safety limit

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 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 Palisades safety analyses uses two DNB correlations. The XNB correlation discussed in References 1 and 2 determines the safety limit for those fuel assemblies initially loaded prior to Cycle 9. The ANFP correlation discussed in References 4 and 5 determines the safety limit for those fuel assemblies initially loaded in Cycle 9 and later. Fuel assemblies initially loaded prior to Cycle 9 are of a different construction than later assemblies which utilize a High Thermal Performance design.

The minimum DNBR analyses are 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

Amendment No. ~~31~~, ~~43~~, ~~118~~, ~~137~~, ~~150~~,
156

2.0 BASIS - Safety Limits and Limiting Safety System Settings

2.3 Basis - Limiting Safety System Settings (continued)

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.[8]

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

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.[5]

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.[7]

References

(1) EMF-92-178, Table 15.0.7-1
(2) Updated FSAR, Section 7.2.3.3.
(3) EMF-92-178, Section 15.0.7-1
(4) XN-NF-86-91(P)
(5) ANF-90-078, Section 15.1.5
(6) ANF-87-150(NP), Volume 2, Section 15.2.7
(7) Updated FSAR, Section 7.2.3.9.
(8) ANF-90-078, Section 15.2.1

Amendment No ~~31~~, ~~82~~, ~~118~~, ~~137~~, ~~150~~,
156

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[4].

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 < 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-92-178 Section 15.0.7.1
(5) ANF-90-078
(6) Consumers Power Company Engineering Analysis EA-A-NL-89-14-1

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-92-178, Section 15.0.5

(next page is 3-69)

TABLE 3.23-1

LINEAR HEAT RATE LIMITS

	No. of Fuel Rods Assembly	
Peak Rod	208	216
	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	216 Reload N	216 Reload O
Assembly F_r^A	1.48	1.57	1.66	1.76
Peak Rod F_r^T	1.92	1.92	1.92	2.04

TABLE 3.23-3

POWER DISTRIBUTION MEASUREMENT UNCERTAINTY FACTORS

LHR/Peaking Factor Parameter	Measurement Uncertainty[a]	Measurement Uncertainty[b]	Measurement Uncertainty[c]
LHR	0.0623	0.0664	0.0795
F_r^A	0.0401	0.0490	0.0695
F_r^T	0.0455	0.0526	0.0722

(a) Measurement uncertainty for reload cores using all fresh incore detectors.
(b) Measurement uncertainty for reload cores using a mixture of fresh and once-burned incore detectors.
(c) Measurement uncertainty when quadrant power tilt, as determined using incore measurements and an incore analysis computer program[6], exceeds 2.8% but is less than or equal to 5%.

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.

To ensure that the design margin of safety is maintained, the determination of radial peaking factors takes into account the appropriate measurement uncertainty factors[1] given in Table 3.23-3

References

(1) FSAR Section 3.3.2.5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated January 29, 1993, as supplemented on April 20, 1993, the Consumers Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-20 for the Palisades Plant. The proposed amendment would revise the Palisades TS Table 3.23-2, Radial Peaking Factor Limits, to add limits for those new fuel bundles to be installed during the 1993 Cycle 11 refueling outage. In addition, the bases for several Specifications (2.1, 2.3, 3.1, 3.12, and 3.23.2) have been updated to reflect the revision of analytical reports for Cycle 11. The April 20, 1993, submittal provided a correction to the original submittal and did not change the initial proposed no significant hazards consideration determination.

The evaluation for Cycle 11 operation is provided in the Siemens Power Corporation (SPC) report EMF-92-178 entitled, "Palisades Cycle 11: Disposition and Analysis of Standard Review Plan Chapter 15 Events." This report documents the results of a disposition and analysis of the FSAR Chapter 14 events in support of Palisades Cycle 11 operation with up to 15% steam generator tube plugging. The events were evaluated in accordance with Chapter 15 of the Standard Review Plan (SRP) and SPC methodology. The proposed changes for Cycle 11 include (1) the insertion of the third full reload of fuel that uses High Thermal Performance (HTP) grid spacers; (2) increase in assembly and rod radial power peaking limits to accommodate a low radial leakage loading pattern; and (3) the reinsertion of eight Reload N partial shielding assemblies (PSA) and sixteen Reload I hafnium assemblies in low powered peripheral locations to reduce vessel fluence.

2.0 EVALUATION

The system transients for non-LOCA events were previously analyzed for Cycle 9. The licensee identified that the Cycle 11 changes (core loading and increase in radial peaking limits) affect only the event minimum departure from nucleate boiling ratio (MDNBR). Therefore, the licensee concluded that the system thermal hydraulic response for the Cycle 9 non-LOCA transient analysis remains valid for Cycle 11. The large break loss of coolant accident (LBLOCA) was analyzed previously with radial peaking limits consistent with those for Cycle 11 and it remains bounding (Reference 2).

The licensee reviewed the Chapter 15 analyses and selected those events that required reanalysis. Their basis for event selection is documented in the *Disposition and Analysis of Events* report (Reference 1). Listed below are the SRP Chapter 15 events affecting the nuclear steam supply system that were reanalyzed for the Cycle 11 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
(2) Single Control Rod Withdrawal

Decreases in Reactor Coolant Inventory
15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve

Of the events listed above, two are not bounded by the Cycle 10 analysis. They include the reactor coolant pump (RCP) rotor seizure and single control rod withdrawal events. The evaluations of these events are discussed below.

2.1 Reactor Coolant Pump Rotor Seizure

The RCP rotor seizure accident causes the pump to stop, reducing core flow, resulting in a reactor scram on low flow. With the reduction in flow, the primary coolant temperature rises causing the power to rise. The subsequent temperature and power rise challenge thermal limits; therefore, reanalysis of the MDNBR and maximum linear heat rate (LHR) is required.

The licensee calculated the MDNBR for RCP rotor seizure as 1.14 and the peak pellet LHR as 15.9 kW/ft. The calculated MDNBR is below the ANFP correlation limit of 1.15. The licensee predicts 0.1% fuel failure due to the violation of the DNBR limit. The licensee indicated that the radiological consequences of this amount of fuel failure are a very small fraction of the 10 CFR 100 limits.

2.2 Single Control Rod Withdrawal

The rod withdrawal event is initiated by an electrical or mechanical failure in the Rod Control System that causes the inadvertent withdrawal of a single control rod. The movement of a single rod out of sequence causes an insertion of positive reactivity and a local increase in the radial power peaking factor. The combinations of these factors challenge the DNB margin, therefore, reanalysis of the MDNBR and LHR was performed.

The licensee calculated the MDNBR for single control rod withdrawal event to be 1.19 and the peak LHR to be 18.5 kw/ft. For this event, the MDNBR is greater than the 95/95 DNBR limit for the ANFP correlation and the peak LHR is less than the 21 kW/ft limit for centerline melt.

3.0 CONCLUSION

The licensee has determined that some of the Chapter 15 accident analyses required reanalysis due to Cycle 11, Reload 0. Of the events that required reanalysis, two were reviewed in this safety evaluation - Single Control Rod Withdrawal (SRP 15.4.3) and RCP Rotor Seizure (SRP 15.3.3).

The staff has reviewed the submittal; in summary, the analyses predicted that the MDNBR will decrease and peak LHR will increase for the single rod withdrawal event in comparison to the Cycle 10 analysis. The predicted LHR for RCP rotor seizure increased over the Cycle 10 analysis. The predicted MDNBR limit for RCP rotor seizure is below the AFNP correlation limit, and the associated fuel failure is predicted to be an acceptably low value of 0.1% of all fuel pins.

Based on the submittal, the staff has concluded that the specified acceptable fuel design limits for the single rod withdrawal event would be met; namely, the fuel shall not experience centerline melt, i.e., LHR is less than 21 Kw/ft, and the DNBR shall have a minimum allowable limit such that there is a 95% probability with a 95% confidence interval that DNB has not occurred.

Although the predicted MDNBR is less than the 1.15 limit for the RCP rotor seizure accident, the licensee has satisfied the acceptance critera in that the potential radiological consequences are within the limits of 10 CFR 100. Therefore, the staff finds the proposed changes to the Cycle 11 radial peaking factors limits 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 Michigan 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 19476). Accordingly, the 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 the 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, SRXB

Dated: June 16, 1993

7.0 REFERENCES

1. Letter from G.B. Slade, Consumers Power, to USNRC, "Palisades Plant Technical Specification Change Request for Cycle 11," dated January 29, 1993.

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