July 15, 1992

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. 150 TO FACILITY OPERATING LICENSE NO. DPR-20 (TAC NO. M81066)

The Commission has issued the enclosed Amendment No. 150 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 February 3, 1992.

This amendment revises the Palisades Technical Specifications (TS) to delete a surveillance test requirement which is no longer appropriate following modifications to the Reactor Protection System (RPS). Additionally, format and editorial changes were made to Section 2.0 of the TS to enhance clarity. A revised Basis was also submitted to correct an error and clarify the discussions dealing with three pump operation.

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/

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

Enclosures:
1. Amendment No.150DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD3-1	PM:PD3-1	PE:PD3-1	SICB	OGC	D:PD3-1
	MKS	aam	aam for			
NAME	MShuttleworth	AMasciantonio	DDiec	SNewberry	RBachmann	LMarsh
DATE	5/26/92	5/22/92	5/22/92	6/1/92	6/2/92	7/1/92

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DF01

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

July 15, 1992

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Palisades Plant
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Sincerely,

A.S. Masciantonio

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

Enclosures:
1. Amendment No.150DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

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Palisades Plant

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Monroeville, Pennsylvania 15146

DATED: July 15, 1992

AMENDMENT NO. 150 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
A. Masciantonio OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
Tech Branch that had input in package (Principal Contributor of SE)
ACRS (10)
GPA/PA
OC/LFMB
W. Shafer, R-III

cc: Plant Service list

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
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 February 3, 1992, 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. 150 , 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

John Stang

L. B. Marsh, Director For
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: July 15, 1992

ATTACHMENT TO LICENSEE AMENDMENT NO. 150

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 are of change.

REMOVE	INSERT
i	i
2-1	2-1
2-2	2-2
2-3	B 2-1
2-4	B 2-2
2-5	B 2-3
2-6	B 2-4
2-7	B 2-5
2-8	-----
2-9	-----
4-3	4-3
4-5	4-5

PALISADES PLANT TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS - APPENDIX A

Amendment No. 31, 51, 57, 85, 108, 118, 128, 1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Reactor Core

The Minimum DNBR of the reactor core shall be maintained greater than or equal to the DNB correlation safety limit.

Correlation	Safety Limit
XNB	1.17
ANFP	1.154

Applicability

Safety Limit 2.1 is applicable in HOT STANDBY and POWER OPERATION.

Action

2.1.1 If a Safety Limit is exceeded, comply with the requirements of Specification 6.7

2.2 Safety Limit - Primary Coolant System Pressure (PCS)

The PCS Pressure shall not exceed 2750 psia.

Applicability

Safety Limit 2.2 is applicable when there is fuel in the reactor.

Action

2.2.1 If a Safety Limit is exceeded, comply with the requirements of Specification 6.7

2.3 Limiting Safety System Settings - Reactor Protective System (RPS)

The RPS trip setting limits shall be as stated in Table 2.3.1.

Applicability

Limiting Safety System Settings of Table 2.3.1 are applicable when the associated RPS channels are required to be OPERABLE by Specification 3.17.1.

Action

2.3.1 If an RPS instrument setting is not within the allowable settings of Table 2.3.1, immediately declare the instrument inoperable and complete corrective action as directed by Specification 3.17.1.

Amendment No. ~~31~~, ~~25~~, ~~43~~, ~~118~~, ~~137~~, 150

TABLE 2.3.1

REACTOR PROTECTIVE SYSTEM TRIP SETTING LIMITS

	RPS Trip Unit	Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating
1.	Variable High Power	≤15% above core power, with a minimum of ≤30% RATED POWER and a maximum of ≤106.5% RATED POWER.	≤15% above core power with a minimum of ≤15% RATED POWER and a maximum of ≤49% RATED POWER.
2.	PCS Flow	≥95% Full PCS Flow.	≥60% Full PCS Flow.
3.	High Pressure Pressurizer	≤2255 psia.	≤2255 psia.
4.	Thermal Margin/ Low Pressure	(a)	(a)
5.	Steam Generator Low Water Level	Above the feedwater ring center line.	Above the feedwater ring center line.
6.	Steam Generator Low Pressure	≥500 psia.	≥500 psia.
7.	Containment High Pressure	≤3.70 psig.	≤3.70 psig.

(a) The pressure setpoint for the Thermal Margin/Low Pressure Trip, P_{trip}, is the higher of two values, P_{min} and P_{var}, both in psia:

$$P_{min} = 1750$$
$$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9493$$

where:

$QA = -0.720(ASI) + 1.028$; when $-0.628 \leq ASI < -0.100$
$QA = -0.333(ASI) + 1.067$; when $-0.100 \leq ASI < +0.200$
$QA = +0.375(ASI) + 0.925$; when $+0.200 \leq ASI \leq +0.565$

ASI = Measured ASI when $Q \geq 0.0625$
ASI = 0.0 when $Q < 0.0625$

$QR_1 = 0.412(Q) + 0.588$; when $Q \leq 1.0$
$QR_1 = Q$; when $Q > 1.0$

Q = Core Power/Rated Power
T_{in} = Maximum primary coolant inlet temperature, in °F.

ASI, T_{in}, and Q are the existing values as measured by the associated instrument channel.

Amendment No. ~~31~~, ~~80~~, ~~118~~, ~~138~~, 150

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 in Cycle 8. 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 in Cycle 8 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

2.0 BASIS - Safety Limits and Limiting Safety System Settings

2.2 Basis - Primary Coolant System Safety Limit

The primary coolant system[1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The Primary Coolant System design pressure is 2500 psia. The maximum allowable Primary Coolant System transient pressure is limited by the pressure vessel limit (ASME Code, Section III) of 110% of design pressure and by the piping, valve, and fitting limit (ASA Section B31.1) of 120% of design pressure. The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings of the reactor High Pressure Trip, primary safety valves, and secondary safety valves have been established to assure never reaching the primary coolant system safety limit. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by the normal setting of the atmospheric steam dump and turbine bypass valves of 900 psia.

References

(1) Updated FSAR, Section 4.
(2) Updated FSAR, Section 4.3.

Amendment No ~~25~~, ~~118~~,150

2.0 BASIS - Safety Limits and Limiting Safety System Settings

2.3 Basis - Limiting Safety System Settings

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 with a specified upper limit. 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.[5]

2. Primary Coolant System (PCS) Low Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.[2] Flow in each of the four coolant loops is determined from pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is determined, for the RPS flow channels, by summing the loop pressure drops across the steam generators and correlating this pressure sum with the sum of steam generator differential pressures which exists at 100% flow (four pump operation at full power t_{ave}). The normal flow with three pumps operating is 74.7% of Full PCS Flow. Full PCS flow is that flow which exists at RATED POWER, at full power T_{ave}, with four pumps operating.

   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.[5]

   Provisions are made in the reactor protective system to permit operation of the reactor at reduced power if one coolant pump is taken out of service. These low-flow and high-flux settings have been derived in consideration of instrument errors and response times of equipment involved to assure that thermal margin and flow stability will be maintained during normal operation and anticipated transients.[4] For reactor operation with one coolant pump inoperative, core power must be reduced and then the Variable High Power and Low Flow setpoints must be adjusted to the three pump values before the pump may be stopped.

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

2.3 Basis - Limiting Safety System Settings (continued)

3. High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the primary and secondary safety valves to prevent primary system overpressure (Specification 3.1.7). In the event of loss of load without reactor trip, the temperature and pressure of the primary coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. This setting is consistent with the trip point assumed in the accident analysis.[8]

4. Thermal Margin/Low Pressure (TM/LP) Trip

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 equations given in Table 2.3.1.

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 PCS pressure and a trip signal is generated when the measured pressure for that channel 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} + ΔP.

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

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

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.[6]

   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-91-176, Table 15.0.7-1
(2) Updated FSAR, Section 7.2.3.3.
(3) EMF-91-176, 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

TABLE 4.1.1

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

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Power Range Safety Channels	a. Check[7]	S	a. Comparison of four-power channel readings.
	b. Check[3]	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparators alarms.
	c. Test	M[2]	c. Internal test signal.
	d. Calibrate[6]	R	d. Channel alignment through measurement/adjustment of internal test points.
2. Wide-Range Neutron Monitors	a. Check	S	a. Comparison of channel indications.
	b. Test	P	b. Internal test signal.
	c. Calibrate	R	c. Channel alignment through measurement/adjustment of internal test points.
3. Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M[2]	c. Bistable trip tester.[1]
4. Thermal Margin/Low Pressurizer Pressure	a. Check:[8]	S	a. Check:
	(1) Temperature input		(1) Comparison of four separate calculated trip pressure set point indications.
	(2) Pressure input		(2) Comparison of four pressurizer pressure indications. Same as 5(a) below.)
	b. Calibrate	R	b. Calibrate:
	(1) Temperature input		(1) Known resistance substituted for RTD coincident with known pressure and power input.
	(2) Pressure input		(2) Part of 5(b) below.
	c. Test	M[2]	c. Bistable trip tester.[1]
5. High-Pressurizer Pressure	a. Check [8]	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M[2]	c. Bistable trip tester.[1]

TABLE 4.1.1

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

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

NOTES:

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

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

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

(4) Deleted

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

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

(7) Required if the reactor is critical.

(8) Required when PCS is >1500 psia.

FREQUENCY Notation

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. DRP-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated February 3, 1992, the Consumers Power Company (CPC or the licensee) submitted a request for changes to the Technical Specifications (TS). The requested changes would delete Footnote (4) from TS Table 4.1.1 "Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System." This Footnote addresses the testing of the low flow trip set points for non-operating reactor coolant pump combinations. Additionally, changes were proposed to TS Section 2.0, "Safety Limits and Limiting Safety System Settings." This section would administratively be revised to enhance clarity and consistency with other sections of the TS. The licensee previously submitted proposed changes (May 30, 1991 and November 1, 1991) for Safety Limits and Limiting Safety System Settings for Variable High Power set point and associated Cycle 10 reload. These changes were separately reviewed and are not included in this evaluation. In addition, a revised Basis for TS, Section 2.0 was also submitted to correct an error and clarify the discussions dealing with three pump operation.

2.0 EVALUATION

2.1 Reactor Protection System (RPS) modification

The original design of the RPS provided a single flow set point selector switch (FSSS) that allowed proper selections of the high power and low flow set points for the three available primary coolant pump modes of operations (2, 3, or 4 operating pumps). The trip set point for different modes was selected by turning the switch to the appropriate position, thereby electronically aligning to the proper potentiometer for each channel. Other set points and pump modes could not be independently selected and verified for surveillance testing without causing the reactor to trip. Footnote (4) in TS Table 4.1.1 was needed to ensure the low flow trip set points for non-operating reactor coolant pump combinations could be verified when shut down and within a specified time after resuming operation with a different pump combination.

In 1988, The NRC approved (Amendment No. 118) the licensee's request to install a variable high power trip to the RPS system. The variable high power trip set point selections were removed from the FSSS. The non-operating pump combination (three-pumps) for low flow trip settings were then conservatively set at the greater than four-pumps operating value (e.g., 110 percent flow) to prevent inadvertent selection of the FSSS. The reactor trip would occur due to a flow mismatch between an actual measured flow and flow set point for three-pumps setting, if the FSSS was moved from the four-pumps position. This action ensured that a loss of flow protection was not compromised.

During the recently completed refueling outage, the licensee implemented a design change to replace aging components for the RPS. The RPS modifications were performed under a 10 CFR 50.59 Safety Evaluation and consisted of replacing all four channels of the RPS power assemblies, trip unit assemblies, interconnection modules, bistable trip units, and auxiliary trip units as well as the incorporation of trip tester functions into the RPS power supply assemblies. The RPS component upgrades were based on analog-to-analog and were form-fit-function replacement with newer design. The RPS power supply front panel combines the functions that were previously contained in the trip tester unit with the indications currently found on the front panel. The test functions on the trip tester unit were implemented in a similar manner to that used in the original design.

Changes made included the rewiring of the Thermal Margin/Low Pressure (TM/LP) selector switch and the removal of the unnecessary FSSS switch. The TM/LP switch and Trip Test switch are functionally unchanged. As a result of the recent RPS modification, the licensee will continue to perform surveillance testing for normal operating pump mode and also has the capability to independently verify preset trip and trip set points at trip unit bistables for other non-operating pump combinations while the reactor is critical. Removal of the FSSS eliminates the potential of inadvertent selection of other than operating pump combination during power operation, thus preventing an inadvertent reactor trip. Therefore, Footnote (4) in TS Table 4.1.1 was no longer needed. The staff has reviewed the CPC submittal and has found the changes are acceptable.

2.2 Technical Specifications (TS) Section 2.0, "Safety Limits and Limiting Safety System Settings" format rearrangement and Basis revised.

The licensee administratively rearranged the TS Section 2.0 format to enhance clarity and consistency with other sections of the TS. Specifications 2.1, 2.2 and 2.3 were rewritten in the format: Specification - Application - Action. Information in the existing "Objective" statements was moved to the Basis. The licensee incorporated the Standard Technical Specifications, NUREG-0212, format for the "Action" statements for failure to meet a Limiting Safety System Setting.

The Basis section for TS Sections 2.1, 2.2, and 2.3 were grouped together and placed in appropriate location following the specifications section. The departure from nucleate boiling (DNB) safety limits have been moved from the Basis and were incorporated into the specification section and the TM/LP trip setting equations were moved to Table 2.3.1, with the rest of the set points limits.

Footnote 1, 2, 3, and 4 from Table 2.3.1, were deleted to reduce redundancy with other TS.

The staff has reviewed the Consumers Power Company, Palisades Plant submittal and has found that the proposed deletion of the surveillance requirement Footnote (4) in Table 2.3.1 and the administrative change to TS Section 2.0 format and the associated Basis to enhance clarity and consistency are acceptable.

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 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 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 accumulative 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 (57 FR 9441). 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.

5.0 CONCLUSION

We have 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) issuance of this amendment will not be inimical to the health and safety of the public.

Principal Contributor: David Diec

Date: July 15, 1992