Docket No. 50-255 April 29, 1976

Consumers Power Company
ATTN: R. B. Sewell
Nuclear Licensing Administrator
212 West Michigan Avenue
Jackson, Michigan 49201

Gentlemen:

The Commission has issued the enclosed Amendment No. 21 to Provisional Operating License No. DPR-20 for the Palisades Plant. The amendment consists of changes to the Technical Specifications, and is in response to your requests dated July 9, 1975, and January 30 and April 5, 1976, as supplemented and amended.

This amendment (1) revises provisions in the Technical Specifications related to the replacement of fuel assemblies in the Palisades core with fuel assemblies of a different design, constituting refueling of the core for operation with Cycle 2 at power levels up to 2200 MWt (100% power), (2) incorporates operating limits in the Technical Specifications based on an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46, (3) modifies various limits established in accordance with the Commission's Interim Acceptance Criteria, and (4) terminates the further restrictions imposed by the Commission's December 27, 1974 Order for Modification of License, and imposes instead limitations established in accordance with the Commission's Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 10 CFR Section 50.46.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/S/

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures: See next page

OFFICE➤						
SURNAME➤						
DATE➤						

Enclosures:
1. Amendment No. 21 to DPR-20
2. Safety Evaluation
3. Federal Register Notice

cc w/encl:
See next page

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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 21
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 July 9, 1975, and January 30 and April 5, 1976, as supplemented and amended, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the 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. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 29, 1976

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered revised pages:

ii	3-29
iii	3-33, 3-59, 3-60, 3-62, 3-63
2-5	3-65
3-1	3-66, 4-9
3-1a	5-3
3-3	6-26

Add the following new pages:

3-29a	3-86
3-66a	3-87
3-84	4-70
3-85	

Remove Fig. 2-4 (page following page 2-13). No replacement.

Revise Appendices B and C as follows:

Remove entire contents of both appendices including title pages.

TABLE OF CONTENTS (Contd)

TABLE OF CONTENTS (Cont'd)

Table 2.3.1

Reactor Protective System Trip Setting Limits

	Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating	Two Primary Coolant Pumps Operating
1. High Power Level(1)	≤ 106.5% of Rated Power	≤ 45% of Rated Power(4) (Continuous operation not permitted)	≤ 25% of Rated Power(4) (Continuous operation not permitted)
2. Low Primary Coolant Flow(2)	≥ 95% of Primary Coolant Flow With 4 Pumps Operating	≥ 71% of Primary Coolant Flow With 4 Pumps Operating	≥ 46% of Primary Coolant Flow With 4 Pumps Operating
3. High Pressurizer Pressure	1950 ± 20 psia	1950 ± 20 psia	1950 ± 20 psia
4. Thermal Margin/Low Pressure(2)(3)	P_T ≥ Applicable Limits To Satisfy Figure 2-3	Replaced by High-Power Level Trip and 1750 Psia Minimum Low-Pressure Setting	Replaced by High-Power Level Trip and 1750 Psia Minimum Low-Pressure Setting
5. Low Steam Generator Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level
6. Low Steam Generator Pressure(2)	≥500 Psia	≥500 Psia	≥500 Psia
7. Containment High Pressure	≤ 5 Psig	≤ 5 Psig	≤ 5 Psig

(1) Below 5% rated power, the trip setting may be manually reduced by a factor of 10.

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

(3) T_h and T_c in °F. Minimum trip setting shall be 1750 psia for two- and three-pump combinations. For four-pump operation, the minimum trip setting shall be 1650 psia for nominal operating pressures less than 1900 psia; and 1750 psia for nominal operating pressures 1900 psia and greater.

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

3.0 LIMITING CONDITIONS FOR OPERATION

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 shall be in operation whenever a change is being made in the boron concentration of the primary coolant.

b. Four primary coolant pumps shall be in operation whenever the reactor is operated continually above 5% of rated power (exception to this specification is permitted as described in Table 2.3.1, Item 1).

c. The minimum flow for various power levels shall be as shown in Table 2.3.1. The measured 'Four Primary Coolant Pumps Operating' reactor coolant vessel flow (as determined by reactor coolant pumps differential pressure and pump performance curves) shall be 124.0×10^6 lb/h or greater, when corrected to 532^oF.

In the event the measured flow is less than that required above, the limits specified in 3.19.1 shall be reduced by 2% for each 1% of reactor flow deficiency.

Continuous operation at power shall be limited to four pump operation. Following loss of a pump, thermal power shall be reduced as specified in Table 2.3.1 and appropriate corrective action implemented. With one or more pumps out of service, return the pumps to service (return to four pump operation) or be in hot standby (or below) within 24 hours. Start-up (above hot standby) with less than four pumps is not permitted.

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

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

(1) Maximum steam generator operating transient differential of 1530 psi.

(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 10 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 operating pressure shall be 1800 psia.

g. The reactor coolant temperature at the inlet to the reactor vessel shall be no greater than 525°F (indicated) during steady state operation above 80% of rated power.

The maximum steam generator operating transient differential pressure is expected to occur during a loss of load transient. The loss of load accident, initiated at a nominal reactor coolant system pressure of 2100 psia and a nominal high pressurizer pressure trip of 2400 psia is analyzed in Section 14.12 of the FSAR. Results of this analysis indicate that the maximum steam generator differential pressure is 1530 psi (assuming that reactor control is in the automatic mode, and that steam dump, bypass and pressurizer relief valves function). This pressure differential is approximately 11% greater than that allowed during normal operation, so that substantial safety margin exists between this pressure differential and the pressure differential required for tube rupture.

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 ECCS analysis has been conducted at a vessel flow of 124.0×10^6 lb/h, and the primary system flow areas and loss coefficients used in the analysis were forced to agree with this flow. The DNB analysis (assuming 122% margin to overpower) has also been performed at this flow with a 3% uncertainty. The ECCS limits associated with this flow rate, which may be more restrictive than the DNB limits, are specified in Section 3.19.1.

In the event the measured flow is less than the required flow, a decrease in allowable thermal limits is required. This decrease in thermal limits, at twice the percentage by which flow is decreased, conservatively maintains the power to flow ratio and provides adequate margin for transients and accidents.

References

(1) FSAR, Sections 6.1.2.2, 14.3.2.

(2) FSAR, Section 4.3.7.

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 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 186 inches (55.5%) and a maximum level of 198 inches (59%) with a boron concentration of at least 1720 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 coolant 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 open with HS-3027 and 3056 positioned to maintain them open.

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

3.3.2 During power operation, the requirements of 3.3.1 may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the requirements of 3.3.1 within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of 3.3.1 are not met within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

a. One safety injection tank may be inoperable for a period of no more than one hour.

b. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

that 25% of their combined discharge rate is lost from the primary coolant system out the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR Figures 14.17.9 to 14.17.13. These demonstrate that the maximum fuel clad temperatures that could occur over the break size spectrum are well below the melting temperature of zirconium (3300°F).

Malfunction of the Low Pressure Safety Injection Flow control valve could defeat the Low Pressure Injection feature of the ECCS; therefore, it is disabled in the 'open' mode (by isolating the air supply) during plant operation. This action assures that it will not block flow during Safety Injection.

The inadvertent closing of any one of the Safety Injection bottle isolation valves in conjunction with a LOCA has not been analyzed. To provide assurance that this will not occur, these valves are electrically locked open by a key switch in the control room. In addition, prior to critical the valves are checked open, and then the 480 volt breakers at MCC 9 are opened. Thus, a failure of a breaker and a switch are required for any of the valves to close.

References

(1) FSAR, Section 9.10.3.
(2) FSAR, Section 6.1.

3.10.3 Power Distribution Limits

a. If the quadrant to core average power tilt exceeds 15%, except for physics tests, then:

   (1) The hot channel factors shall promptly be demonstrated to be less than design values $F_q^N = 3.62$ $F_{\Delta H}^N = 1.94$ or,

   (2) Immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

b. If the power in a quadrant exceeds core average by 10% for a period of 24 hours or if the power in a quadrant exceeds core average by 20% at any time, immediate action shall be initiated to reduce reactor power below 50% until the situation is remedied.

c. If the power in a quadrant exceeds the core average by 15%, and if the hot channel factors cannot be demonstrated promptly to be within design limits, then the overpower trip set point shall be reduced to 80% and the thermal margin low-pressure trip set point (P_{trip}) shall be increased by 400 psi.

d. If the power in a quadrant exceeds core average by 5% for a period of 30 days, immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

e. If the ratio of the power in the upper half to the lower half of the core is not within the range of 0.5 to 1.8 as indicated by the in-core and out-of-core detectors, then immediate action shall be taken to reduce the power below 75% of rated power until the situation is remedied.

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

3.10.4 Misaligned or Inoperable Control Rod or Part-Length Rod

a. A control rod or a part-length rod is considered misaligned if it is out of position from the remainder of the bank by more than 8 inches.

b. A control rod is considered inoperable if it cannot be moved by its operator or if it cannot be tripped. A part-length rod is considered inoperable if it is not fully withdrawn from the core and cannot be moved by its operator. If more than one control rod or part-length rod becomes misaligned or inoperable, the reactor shall be placed in the hot shutdown condition within 12 hours.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

c. If a control rod or a part-length rod is misaligned, hot channel factors must promptly be shown to be within design limits or reactor power shall be reduced to 75% or less of rated power within 2 hours. In addition, shutdown margin and individual rod worth limits must be met. Individual rod worth calculations will consider the effects of xenon redistribution and reduced fuel burnup in the region of the misaligned control rod or part-length rod.

3.10.5 Regulating Group Insertion Limits

a. To implement the limits on shutdown margin, individual rod worth and hot channel factors, the limits on control rod regulating group insertion shall be established as shown on Figure 3-6. These limits may be revised during fuel cycle life based on physics calculations and physics data obtained during plant start-up and subsequent operation.

b. The sequence of withdrawal of the regulating groups shall be 1, 2, 3, 4.

c. An overlap of control banks in excess of 40% shall not be permitted.

d. If the reactor is subcritical, the rod position at which criticality could be achieved if the control rods were withdrawn in normal sequence shall not be lower than the insertion limit for zero power shown on Figure 3-6.

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, 3.10.2.b, 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.

PALISADES PLANT
CONTROL ROD INSERTION LIMITS
ALLOWED POWER LEVEL (% OF 2200 Mwt) VS CONTROL ROD INSERTION (%) BY ROD GROUP
FOR OPERATION BELOW 1900 psia

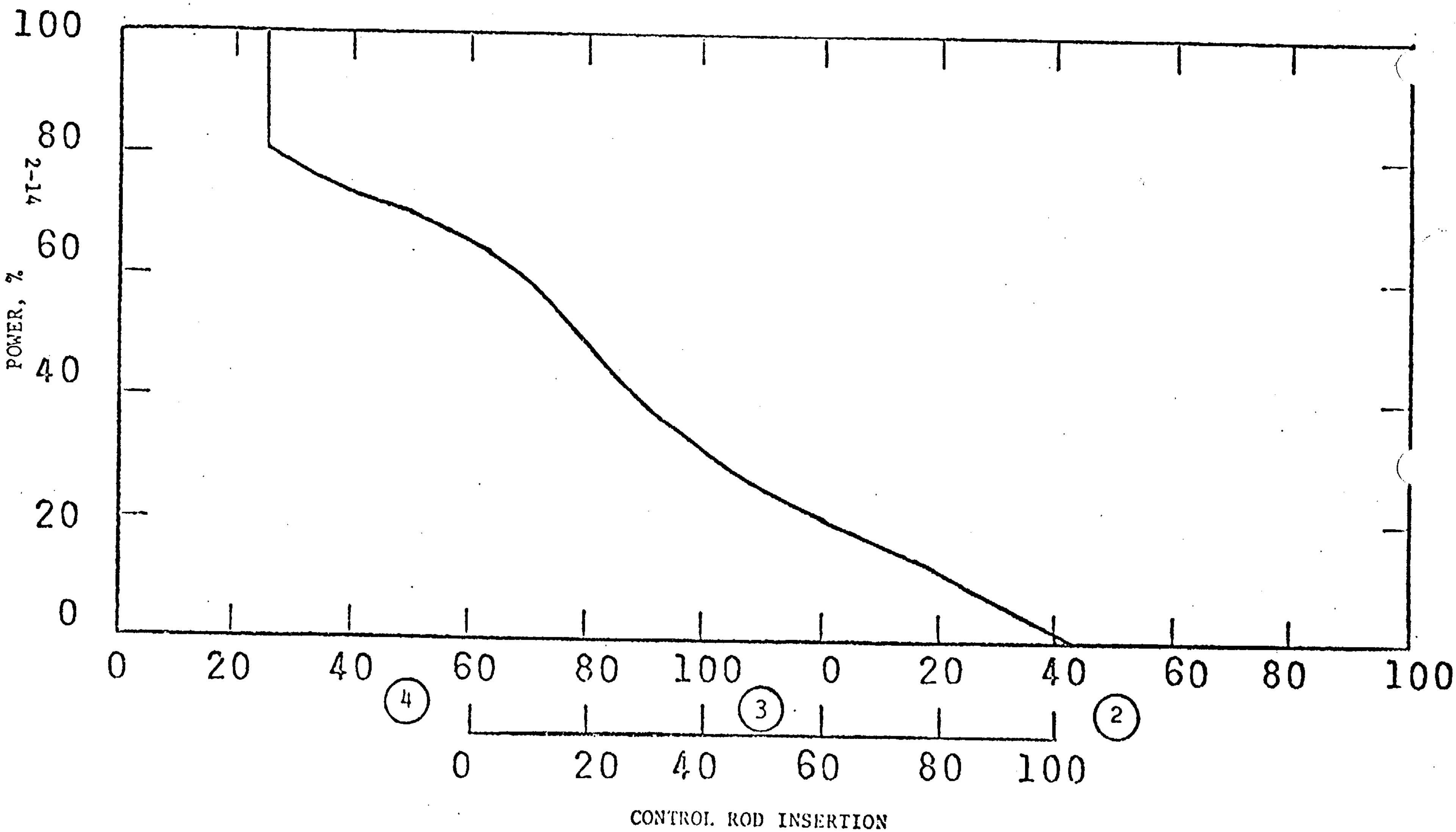


Fig. 3-6
Amendment No. 21

value; core outlet thermocouples; and in-core detectors.[3] These values will form the basis for the calculation of peaking factors. Calibration of the out-of-core detectors will take into account the local and total power distribution.

Studies have been performed to show that with this ratio (0.5 - 1.8), total peaking factors are not exceeded. Tests will be performed during start-up testing to verify this relationship.

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 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") 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.

For a "dropped" control rod during operation at high power, a turbine runback will automatically decrease the maximum power to 70% of rated power.[4] Continued operation with that rod fully inserted will only be permitted if the hot channel factors, shutdown margin and ejected rod worths 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 Spec. 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

3.11 IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the in-core instrumentation system.

Objective

To specify the functional and operability requirements of the in-core instrumentation system.

Specification

a. Sufficient in-core instrumentation shall be operable whenever the reactor is operating at or above 50% rated power (65%of rated power if no dropped or misaligned rods are present) in order to: (1) Assist in the calibration of the out-of-core detectors, and (2) check gross core power distribution. As a minimum, 50% of the incore detectors and not less than 10 individual detectors per quadrant, which shall include 2 detectors at each of the four axial levels, shall be operable.

b. For power operation above 85% of rated power, in-core detector alarms generated by the data logger shall be set, based on the latest power distribution obtained, such that the peak linear power does not exceed the limit specified in Section 3.19. If four or more coincident alarms are received, the validity of the alarms shall be immediately determined and, if valid, power shall be immediately decreased below alarm set point and a power distribution map obtained. If a power distribution is not obtained within 24 hours of the alarm conditions, power shall be reduced to 85% of rated power.

c. The in-core detector alarm set points shall be established based on the latest power distribution maps, normalized to the kW/ft limit defined in Section 3.19.

d. Power distributions shall be evaluated every week or more often as required by plant operations.

e. The data logger can be inoperable for two hours. If at the end of two hours, it is not available, the power level shall not exceed 85% of rated power.

f. If the ata logger for the in-cores is t in operation for more than two hours and reactor power is at or above 50% of rated power (65% of rated power if no dropped or misaligned rods are present), readings shall be taken and logged on a minimum of 10 individual detectors per quadrant (to include at least 50% of the total number of detectors in a 10-hour period) at least each two hours thereafter or the reactor power level shall be reduced to less than 50% of rated power (65% of rated power if no dropped or misaligned rods are present). If readings indicate a local power level equal to or greater than the alarm setpoint, the action specified in 3.11.b shall be taken.

Bases

A system of 45 in-core flux detector and thermocouple assemblies and a data display, alarm and record functions has been provided.(1) The out-of-core nuclear instrumentation calibration includes:

a. Calibration (axial and azimuthal) of the split detectors at initial reactor start-up and during the power escalation program.

b. A comparison check with the in-core instrumentation in the event abnormal readings are observed on the out-of-core detectors during operation.

c. Calibration check during subsequent reactor start-ups.

d. Confirm that readings from the out-of-core split detectors are as expected and that the ratio of the top and bottom detector readings is acceptable.

Core power distribution verification includes:

a. Measurement at initial reactor start-up to check that power distribution is consistent with calculations.

b. Subsequent checks during operation to insure that power distribution is consistent with calculations.

c. Indication of power distribution in the event that abnormal situations occur during reactor operation.

If the data logger for the in-core readout is not in operation for more than two hours, power will be reduced to provide margin between the actual peak linear heat generation rates and the limit and the in-core readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the manpower available, the reactor power will be reduced further to minimize the probability of exceeding the peaking factors. The time interval of two 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 data logger is returned to service.

<u>Reference</u> - (1) FSAR, Section 7.4.2.4.

3.19 Linear Heat Generation Rate Limit Associated With LOCA Considerations

Applicability

Applies to fuel linear heat generation rates.

Objective

To delineate the requirements regarding fuel linear heat generation rates associated with a postulated Loss of Coolant Accident.

Specification

3.19.1 The linear heat generation rate with appropriate consideration of normal flux peaking, measurement-calculational uncertainty, engineering factor, increase in linear heat rate due to axial fuel densification, power measurement uncertainty, and flux peaking augmentation shall not exceed 14.19 kW/ft.

The measurement-calculational uncertainty shall be 10%, the engineering factor shall be 3%, the increase in linear heat rate due to axial densification shall be 1.75% (as applied to hot dimensions), the power measurement uncertainty shall be 2%, and the flux peaking augmentation factor shall be as given in Figure 3-7 for uncollapsed fuel and Figure 3-8 for collapsed fuel. Augmentation factors for pressurized densification resistant ENC fuel and pressurized high density CE fuel shall be 1.0.

Bases

To maintain the integrity of the fuel cladding under the conditions of a postulated Loss of Coolant Accident (LOCA), the Emergency Core Cooling Systems (ECCS) must satisfy certain criteria set forth by the US Nuclear Regulatory Commission in

5

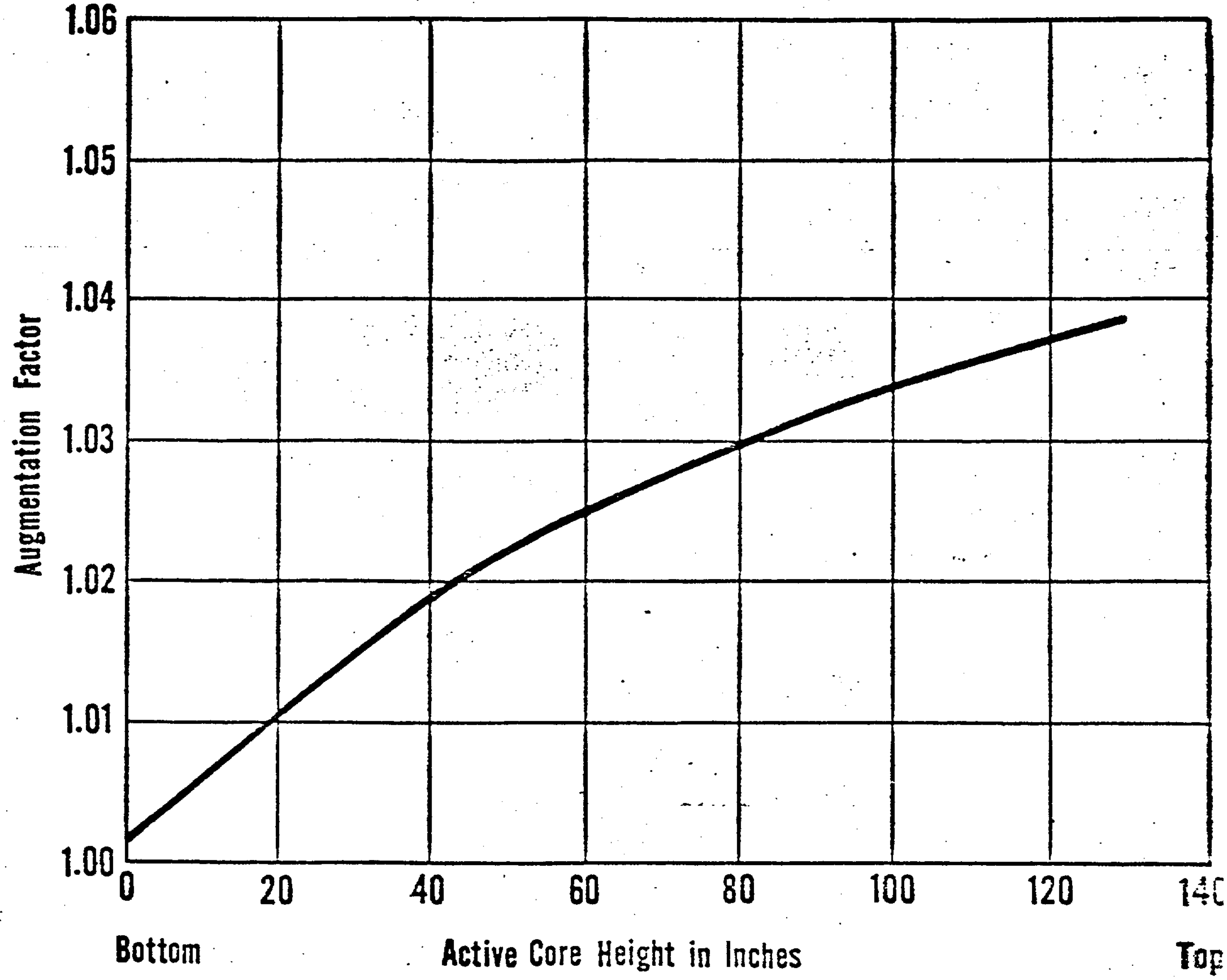
10 CFR 50.46(b). These criteria assure that under LOCA conditions the temperature and oxidation of the cladding will be controlled such that the uranium dioxide pellets will be maintained in a coolable geometry. These criteria are summarized below:

1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation, including effects of cladding thinning and rupture.
3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The computer codes which predict cladding temperature and oxidation under LOCA conditions must be approved by the US Nuclear Regulatory Commission in accordance with Appendix K of 10 CFR 50. The results of these computer code calculations depend on ECCS performance characteristics and fuel design. Analyses performed with approved codes and techniques for each fuel design type taking into consideration anticipated operating conditions will be kept on file at the plant and at the General Office. These analyses provide safety limits given in terms of allowable linear heat generation rates in kW/ft for each fuel design type.

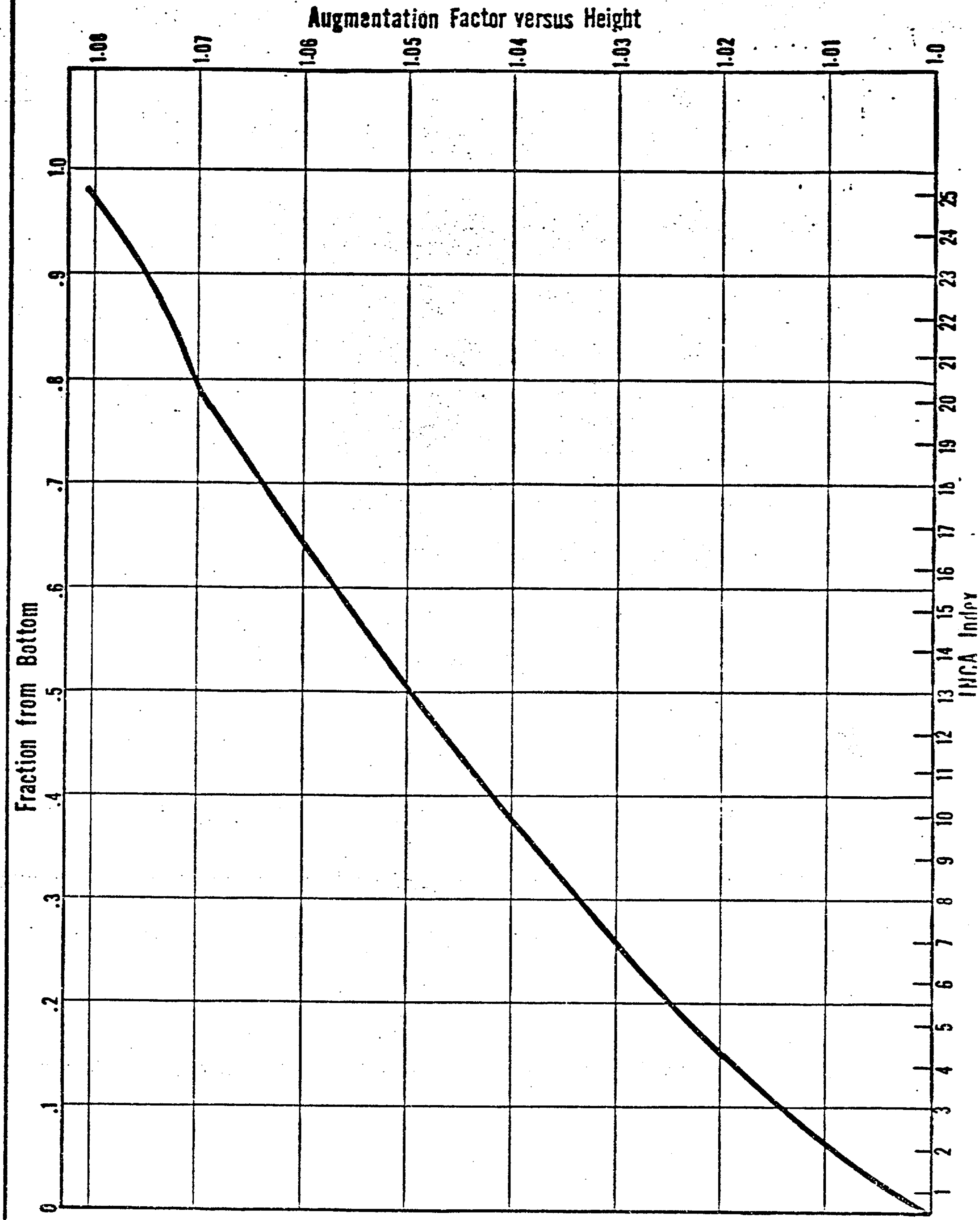
Appropriate factors for measurement-calculation uncertainty, engineering factor and shortening of the fuel pellet stack are specified to insure that linear heat generation rate limits are not exceeded during steady state operation.

Amendment No. 21



Amendment No. 21

Augmentation Factor Versus Height Monte Carlo Computation for Palisades	Palisades Technical Specifications	Figure 3-7



Amendment No. 21

Augmentation Factor [From Enclosure to P-CE-4059 10/10/75]	Palisades Technical Specifications	Figure 3-8

Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Cont'd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
15. Boric Acid Heat Tracing System	a. Check	D	a. Observe temperature recorders for proper readings
16. Main Steam Isolation Valve Circuits	a. Check	S	a. Compare four independent pressure indications.
	b. Test (3)	R	b. Signal to meter relay adjusted with test device to verify MSIV circuit logic.
17. SIRW Tank Temperature Indication and Alarms	a. Check	M	a. Compare independent temperature readouts.
	b. Calibrate	R	b. Known resistance applied to indicating loop.
18. Low Pressure Safety Injection Flow Control Valve CV-3006	a. Check	P	a. Observe valve is open with air supply isolated.
19. Safety Injection Bottle Isolation Valves	a. Check	P	a. Ensure each valve open by observing valve position indication and valve itself. Then lock open breakers (at MCC-9) and control power (key switch in control room).
20. Safety Injection miniflow valves CV-3027, 3056	a. Check	P	a. Verify valves open and HS-3027 and 3056 positioned to maintain them open.

Notes: (1) Calibration of the sensors is performed during calibration of Item 5(b), Table 4.1.1.

(2) All monthly tests will be done on only one channel at a time to prevent protection system actuation.

(3) Calibration of the sensors is performed during calibration of Item 7(b), Table 4.1.1.

S - Each Shift
D - Daily
M - Monthly
Q - Quarterly
R - Each Refueling Shutdown, But Not to Exceed 16 Months
P - Prior to Each Start-up if Not Done Previous Week

4.15 Primary System Flow Measurement

Applicability

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

Objective

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

Specification

After each refueling outage, or after plugging 10 or more steam generator tubes, a primary system flow measurement shall be made with four primary coolant pumps in operation before the reactor is made critical.

Basis

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

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) (Contd)

5.3.2 Reactor Core and Control

a. The reactor core shall approximate a right circular cylinder with an equivalent diameter of about 136 inches and an active height of about 132 inches.

b. The reactor core shall consist of approximately 43,000 Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall be grouped into 204 assemblies.

A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution.

c. The fully loaded core shall contain approximately 211,000 pounds UO_2 and approximately 56,000 pounds of Zircaloy-4.

Poison may be placed in the fuel bundles for long-term reactivity control.

d. The core excess reactivity shall be controlled by a combination of boric acid chemical shim, cruciform control rods, and mechanically fixed boron rods where required. Forty-five control rods shall be distributed throughout the core as shown in Figure 3-5 of the FSAR. Four of these control rods may consist of part-length absorbers.

5.3.3 Emergency Core Cooling System

An emergency core cooling system shall be installed consisting of various subsystems each with internal redundancy. These subsystems shall include four safety injection tanks, three high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in Section 6 of the FSAR.

5.4 FUEL STORAGE

5.4.1 New Fuel Storage

a. Unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less

6.9.3.3. Special Reports

a. Special reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable referenced specification:

Area	Specification Reference	
Prestressing, Anchorage, Liner and Penetration Tests	4.5.4 4.5.5	90 Days After Completion of the Test*
Primary System Surveillance Evaluation and Review	4.3	Five Years

*A test is considered to be complete after all associated mechanical, chemical, etc., tests have been completed.

b. Bimonthly status reports on the program to improve the reliability of the paths to prevent post-LOCA boron precipitation shall be submitted to the Division of Operating Reactors until completed.

c. The results of the fuel surveillance program (CPCo letters dated March 20 and April 8, 1976) shall be reported to NRC prior to the re-use of Cycle 2 fuel for Cycle 3.

6.10 RECORD RETENTION

(Records not previously required to be retained shall be retained as required below commencing with the effective date of Technical Specification Change No. 20. A system for efficient record retrieval shall be in effect not later than June 1976.)

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Reportable Occurrences.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

SAFETY EVALUATION

OF THE

PALISADES CYCLE II RELOAD

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

AMENDMENT NO. 21

TO

PROVISIONAL OPERATING LICENSE DPR-20

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

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Safety Evaluation
License Number DPR-20
Docket Number 50-255
Palisades Plant

1.0 Introduction

By letters dated July 9, 1975, and January 30 and April 5, 1976, Consumers Power Company (the licensee) requested changes in the Technical Specifications appended to Provisional Operating License No. DPR-20 for the Palisades Plant. The proposed changes relate to the replacement of all fuel assemblies in the Palisades core with fuel assemblies of a different design, constituting refueling of the core for operation in Cycle II at power levels up to 2200 MWt (100% power). In addition, the proposed changes include operating limits based on an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46, as required by the Commission's Order for Modification of License dated December 27, 1974.

1.1 Discussion

The Palisades core consists of 204 fuel assemblies, each having a 15 x 15 array of fuel rods. Control is provided by top-entry cruciform control blades (rather than rods), dissolved boric acid in the primary coolant, and burnable poison rods containing B_4C and Al_2O_3. The Cycle II core is unusual for a reload in that it contains no previously exposed fuel. This core utilizes a three batch, two-zone configuration with 68 high-enrichment Combustion Engineering (CE) assemblies located at the core circumference, and a mixed array of low-enrichment and poisoned high-enrichment Exxon assemblies in the interior.

The licensee provided the needed technical information for our review including a general description of the reload core, detailed mechanical design data on the reload fuel, the results of the nuclear and thermal-hydraulic evaluation (which include the effects of steam generator tube plugging), accident and transient analysis of the new core, and a detailed startup test program in support of the Cycle II reload application. Since this is the first application for a Palisades reload with Exxon fuel assemblies, Exxon Nuclear Company (ENC) has provided documentation on the ENC ECCS performance analysis models and computer codes.

From our review of the available reload information, and subject to the requirements described in the following sections, we conclude that it is acceptable for the licensee to proceed with Core II operation in the manner proposed. Our review and evaluation of the licensee's Core II reload submittals is discussed in the following sections.

2.0 Mechanical Design

Palisades Cycle II fuel reload will consist of 136 assemblies of ENC fuel denoted as assemblies E and F and 68 assemblies of CE fuel denoted as type D.

The fuel rods are in a 15 x 15 lattice array. They are clad with Zircaloy-4 and are prepressurized with helium. The E and F type fuel rods are slightly different in design from the previous Combustion Engineering fuel. The cladding is 19% thicker, the fuel-cladding gap is larger and the fuel pellet length-to-diameter ratio has been reduced to provide the fuel rod with more resistance to pellet-cladding interaction failures.

The rod bundles contain 10 Zircaloy spacer grids with Inconel springs to laterally locate the fuel rods and Zircaloy guide bars.

The Combustion Engineering type D fuel has thicker cladding, shorter pellets, a larger pellet-cladding gap and a higher internal pressure to increase resistance to pellet cladding interaction failures relative to that of the Cycle I fuel.

The licensee has used ENC analytical methods to show that the Exxon and CE fuel will operate safely during Cycle II.

Fuel densification effects were considered by the licensee. The GAPEXX[23] code was used to calculate fuel stored energy taking fuel densification into account. The COLAPX[23] code was used to predict the time at which cladding collapse due to axial gaps in the pellet stack would occur. Cladding collapse is not predicted for Cycle II. Both of the above ENC codes have been reviewed and approved by the staff.

The licensee has taken into account the increase in linear heat generation rate due to the shortening of the fuel stack.

The licensee has considered power spikes caused by the formation of axial gaps due to fuel densification in both the LOCA and DNB analyses. In both cases the licensee has concluded that the power spike need not be considered. In the case of the ECCS analysis justification is based on the topical report WCAP 8359 "Effects of Fuel Densification Power Spikes on Clad Thermal Transients." For DNB analyses, the licensee justifies neglecting power spikes based on the topical report WCAP 8219. The staff concludes that this is acceptable.

The licensee has not considered the effects of rod bowing on DNB and LOCA analyses. In the case of DNB analyses the staff finds the effects of rod bowing to be within the envelope of other thermal performance margins, based on experimental data in the Westinghouse topical report WCAP 8176 "Effect of a Bowed Rod on DNB," which has been reviewed and accepted by the staff. The topical report shows that at the maximum pressure expected for a Palisades anticipated transient the effect of rod bowing on DNB is negligible.

In the case of LOCA analysis, the staff has determined that the amount of bowing expected at the end of Cycle II will not have a significant effect on the calculated results. This is discussed further in Section 3.0.

Because the fuel design for Palisades has some new features, the licensee has initiated a fuel surveillance program which is described in the response to question 1.6[9]* Eight CE and sixteen ENC fuel assemblies will be inspected. This inspection will consist of a visual inspection of 100% of the peripheral rods on these rod bundles. In addition, detailed dimensional inspections will be conducted on two type D (CE) fuel assemblies and two type E and two type F (ENC) fuel assemblies.

The licensee has agreed to report the results of fuel surveillance prior to the re-use of Cycle II fuel in Cycle III. In the case of any abnormalities discovered during this surveillance program, the staff will require additional inspections to be performed. The staff concludes that this surveillance program is acceptable.

The Exxon fuel design for Palisades Core II is similar to that supplied by Exxon for previous cores. The cladding is Zircaloy-4, used as the cladding material in previous fuel supplied for the Yankee-Rowe Core XII and H. B. Robinson Core IV reload cores. Forty assemblies were loaded into Yankee Rowe XII, and fifty-two assemblies were loaded into H. B. Robinson. The enrichment of the fuel for Palisades is in the range of that used in the previously mentioned cores. The general dimensions of the fuel rod (including diametral gap which is of importance for stored energy) are within the range of PWR fuel designs previously irradiated successfully by the industry.

*Letters dated March 20 and April 8, 1976

Relative to other PWR fuel designs which have operated successfully to design burnups, the Palisades fuel design has several design features which should give confidence that the fuel rods will maintain their integrity throughout irradiation. For example, the Exxon fuel pellets have a length to diameter ratio of less than one and a thicker cladding. These changes result in a greater resistance to pellet-cladding interaction over that of the previous Palisades fuel. A high internal pressure of helium gives additional resistance to both pellet-cladding interaction and cladding collapse, and decreases the fuel stored energy (for the same rod power) compared to the fuel irradiated in the previous cycle.

Approximately 800 bundles manufactured by Exxon are in-core, in PWRs and BWRs, with burnups ranging from first cycle to 25,000 MWD/MTU. Approximately 10% of these have exposures between 15-20,000 MWD/MTU. Based on sipping results and surveillance of representative assemblies, no failures have been observed or detected.

The licensee has also described his fuel design methods in an ENC topical report(2) which, when approved, may be referenced for future fuel reloads of this type. Review of this report by the staff is in progress, and sections of this report regarding detailed analytical methods related to the effects of fuel densification have been reviewed and found acceptable. Other sections of this report remain to be reviewed in more detail, and therefore have not been considered in our conclusion that the Exxon fuel design is acceptable for Cycle II. Rather, this conclusion is based on (1) other fuel design analytical methods cited above which have been reviewed and approved by the staff, (2) the similarity of the reload fuel (with improvements as noted) to that used in Cycle 1 which was previously found acceptable, and (3) the successful operating experience achieved to date with Exxon fuel. These factors, taken together, provide reasonable assurance that Exxon fuel will demonstrate acceptable performance. This will be confirmed by a fuel surveillance program at the end of Cycle II.

3.0 Nuclear Design

The Cycle II core contains no exposed fuel, and therefore is neutronically similar to a first core. The reloading scheme will place unpoisoned, unexposed 2.73 w/o enriched Combustion fuel (Type D) around the circumference of the core, and a mixture of 3.04 w/o enriched Exxon fuel with burnable poison (Type E) and 1.5 w/o enriched unpoisoned Exxon fuel (Type F) in the interior of the core. Since only the Type E fuel contains burnable poison, Core II has a higher critical boron concentration than Core I. The delayed neutron fraction is close to that of the first cycle, and much higher than that of a typical reload.

The Type E fuel has a uniform enrichment, while the Type D Combustion fuel contains a lower enrichment in the outer row of rods. However, the placement of the burnable poison rods in the Type E fuel is such that its local flux distribution is more uniform than that of the Type D fuel, and the limiting local peaking factor remains 1.21.

The licensee's calculation of control bank worths for Cycle II indicates that there is a substantial excess margin over a 1% design shutdown margin allowance throughout the cycle life. In addition, an uncertainty allowance has been allowed for calculation of rod worth. The licensee will also perform an extensive set of startup measurements which will provide additional verification that the shutdown margin will be maintained throughout Cycle II operation.

The nuclear calculations for Cycle II have been performed by Exxon Nuclear Company, using methods which have been generically approved.(22-26) In addition, the extensive set of startup measurements mentioned above will provide experimental verification for the BOC values of the rod bank worths, ejected rod worth, dropped rod worth, moderator coefficients, and power coefficient. This procedure is acceptable.

Peak linear heat generation rates (LHGR) for Cycle II are restricted as indicated in the section on ECCS analysis. The licensee has proposed changes to the Technical Specifications which would administratively limit the actual peak LHGR to less than the LOCA limit. Power distributions will be measured by means of the incore detector system weekly or more often as required by plant operations. These power distributions will then be used to generate alarm setpoints on the individual incore detectors. Total power will be reduced (and the base power distribution updated) if these setpoints are exceeded. We find this acceptable.

The interpolative coefficients used by the process computer's incore power distribution program to calculate power maps have been pre-calculated by the licensee and Exxon Nuclear Company using approved methods equivalent to those used for Cycle I.

The measured axial and radial peaking factors, a local peaking factor of 1.21, factors of:

- 10% for measurement-calculational uncertainty
- 3% engineering factor
- 1.75% for stack height shortening due to fuel densification
- 2% for total thermal power measurement uncertainty

and an axially variable flux peaking augmentation factor will be combined to calculate a conservative measured LHGR to be compared with the LOCA limit.

Section 3.19 of the Proposed Technical Specifications does not include a LOCA penalty for rod bow. The licensee has argued that no rod bow is expected for the Cycle II. The staff does not agree with this position, and instead calculates a maximum single rod displacement of 0.047 in. during Cycle II. Based on information supplied by the licensee in their March 20 submittal, such a displacement would cause an increase of 0.9% in local LHGR. This corresponds to statistically increasing the 3% engineering factor from 3% to 3.13%. Because the present penalties total to more than 17%, the effect of rod bow is negligible and no additional penalty for rod bow is required.

Section 3.10.3 of the Technical Specifications would allow indefinite full-power operation with quadrant to core average power tilts of up to 10%. The licensee has justified this unusually high figure by taking credit for the more detailed power distribution provided by incore monitoring. The staff agrees that incore monitoring can in principle ensure that the allowable kw/ft will not be exceeded. However, several concerns remain:

- the certified version of the incore monitoring code uses 1/8 core symmetry. The validity of the 10% measurement-calculational uncertainty in Section 3.19 has not yet been justified for asymmetric operation.
- The accuracy of incore power distribution monitoring decreases in the presence of a long-term tilt.

In addition, it is rather surprising that a tilt limit of 10% is necessary for reactor operations. The staff concludes that operation with large long-term tilts is unjustified by the information available at this time. Therefore, we will require that power be reduced to 75% after 30 days of operation with a quadrant to core average power tilt in excess of 5%.

Because of the credit taken for the incore system to justify operation of the reactor with perturbed power distributions such as may be caused by dropped control rods or high axial offsets, the incore detectors must be operable whenever the reactor is operated at significant power. Sections 3.11.a and 3.11.f of the proposed Technical Specifications allow reactor operation with no incore monitoring up to 75% power. The staff does not agree that the available information justifies operation with no incores with only 25% margin. Based on a design peaking factor of 3.62 (as discussed on p. 3-59 of the present Technical Specifications), a peaking factor of 2.89 used in the LOCA analysis (from letter dated April 5, 1976, p. 16), the nominal peaking factors given in the letter dated April 8, 1986, the dropped rod incident analysis on p. 14.4-2 of the FSAR, a measurement-calculational uncertainty of 10%, and a short-term tilt limit of 10%, the staff concludes that operation above 50% power (or 65% power if no dropped or mis-aligned rods are present) without incore monitoring is unjustified, and we will require that sections 3.11.a and 3.11.f be appropriately modified.

Section 3.11.a of the Technical Specifications requires that at least 10 individual detectors per quadrant (including two detectors at each of the four axial levels) be operable when the reactor is operating at high power. The licensee has argued that the low power density and high number of incore detectors justify a greater number of failed detectors than permitted in other Combustion Engineering reactors. These other reactors are typically required to have 75% of their incores operable. The staff agrees that 75% are not required for operation of the Palisades reactor. However, the present Specification corresponds to only 2 1/2 detector strings per quadrant and more than three quarters of the incore system out of service. This number of detectors is sufficient for the original purposes of the system, but is not necessarily adequate if the incore system is to be used for ensuring that LOCA limits will not be violated. This is especially true since the licensee wishes to take credit for the incore system to allow operation with a dropped rod. Therefore, we will require that, in addition to ten operable detectors per quadrant (which must include 2 detectors at each of the four axial levels), at least 50% of the total number of incore detectors be operable whenever the reactor is operating at or above 50% rated power. (65% if no dropped or mis-aligned rods are present).

Sections 3.11.b and 3.11.e of the proposed Technical Specifications allow indefinite operation without automatic reading of the incore detectors by the data logger at power levels below 85% of the level permitted by the LOCA limit. The staff finds this Specification unacceptable because the reactor operator, under the circumstances which invoke this Specification, would have no rapid means of measuring the LOCA limit to which the derate would be tied. Therefore, we will require that the "85% of the value defined in Section 3.19" in Sections 3.11.b and 3.11.e be replaced by "85% of rated power." In addition, we will require that Section 3.11.f be modified such that if reactor power is greater than 50% of rated (or 65% of rated if no dropped or mis-aligned rods are present) and the data logger is not in use for automatic scanning and alarm generation, manual incore monitoring shall take place such that at least 50% of the total number of detectors are manually read in a 10 hour period.

Section 3.10.3 of the Technical Specifications allows the use of part-length rods to control axial Xenon oscillations. Palisades is the only Combustion plant which allows the use of part-length rods. The licensee has argued that the in-core detector system will be used to constrain the power distribution within the design ehvelope. The staff agrees that the incore system could be used this way provided that the incore monitoring program (INCA) were executed after part-length rod movement and periodically during Xenon stabilization, the resulting power distribution were compared with the power distributions used in the transient analysis, and sufficient margin to DNB resulted to allow the plant to withstand transients. However, the available information does not include the power distributions used in the transient analysis. It is likely that these analyses are equivalent to those used in other CE plants. It has already been demonstrated that these other plants can, under some circumstances, violate the design envelopes by using part-length rods. The staff concludes that the available information does not justify the use of part-length rods. We will therefore require that these rods remain withdrawn from the core except for rod exercises and physics tests. Part-length rod insertion is acceptable for (1) physics tests, since resulting power distributions are closely monitored under test conditions, and (2) rod exercise purposes, since insertion of about 6" necessary for the exercise test has an insignificant effect on power distribution.

We conclude that the proposed Technical Specifications, subject to the requirements set forth above, are acceptable because they will effectively limit the reactor power to a level consistent with the linear heat generation rate used in the LOCA analysis. The licensee has agreed to the changes discussed above.

4.0 Thermal and Hydraulic Design

The thermal-hydraulic analysis of the Cycle II core shows the following results:

a. The ENC and CE fuel assemblies are thermally and hydraulically compatible.

b. The minimum departure from nucleate boiling ratios (MDNBR) for both fuel types are always greater than 1.30 for normal operation and anticipated transients.

The thermal-hydraulic analysis included both experimental[6] measurements and theoretical calculations.[7,8] ENC has performed hydraulic flow tests to evaluate the compatibility between the CE type D and the ENC type E and F fuel assemblies. The results of these tests show that although there were some differences in the pressure drop distributions between the upper and lower tie plates and the bare rods and spacers, the difference in flow through the ENC and CE assemblies is small. This difference of flow has been considered in the analysis and this flow differential is acceptable.

The adequacy of the ENC fuel for meeting MDNBR requirements has been verified with transient analyses performed at 102% power. The results of the transient calculations are discussed later in this evaluation.

DNB calculations show that the MDNBR is greater than 1.30 for both ENC and CE fuel assemblies under the operating conditions of Cycle II. Additional margin is provided by the fact that the steady state DNB calculations were performed at a power level of 2684 MWt while Palisades will be licensed for only 2200 MWt for Cycle II.

We find the MDNBR values acceptable (>1.30). We conclude from our review that the thermal and hydraulic design of the Cycle II core is acceptable.

5.0 Transient and Accident Analysis

5.1 ECCS Cooling Performance (LOCA) Analysis

5.1.1 Evaluation Model

The licensee has evaluated through ENC the Palisades ECCS cooling performance using a calculational model that conforms to the requirements of 10 CFR Part 50, Appendix K.

The calculational model used by ENC for Palisades is similar to the approved H. B. Robinson ECCS performance evaluation model addressed in the staff's Safety Evaluation of September 11, 1975 and its supplement.[15] The model has been modified for Palisades by including: additional axial nodalization, the FLECHT correlation for short cores and skewed power profiles, a large reverse K factor in the junction between the intact leg on the broken loop and the vessel to prevent reverse flow of steam, injection pressure penalties for a 60^{0} injection angle instead of a 90^{0} angle, and an allowance for flow communication between the two halves of the broken cold leg. In addition the radiation model was not employed. Also, we have reviewed the use of the H. B. Robinson model for the Palisades ECCS performance evaluation with respect to the differences in plant design, particularly the shorter core, the thinner fuel rods, and the different accumulator arrangement in the Palisades facility. We have determined that the H. B. Robinson model conservatively accommodates these differences. We conclude that the application of the modified H. B. Robinson model to the Palisades plant is acceptable.

Since the ECCS analysis has been conducted assuming four reactor coolant pumps are in operation, we have added restrictions in the Technical Specifications which prohibit operation above 5% power for more than 24 hours with less than four reactor coolant pumps running. The intent of this restriction is to prohibit sustained operation with less than 4 pumps operating, pending receipt and approval of further analyses in support of such operation. Such a 24-hour period allows a reasonable length of time to restore an inoperable pump to service, and avoids undesirable and unnecessary further plant transients, such as a manual scram or rapid plant shutdown which otherwise would be required. In addition, up to 12 hours is allowed in order to conduct reactor internals noise measurements in different coolant pump combinations. This interval of time is adequate to make these measurements, and is less than the 24 hour period above which we find acceptable. Long-term operation of the facility without having conducted these tests (when necessary) represents a greater risk than that incurred by permitting less than 4-pump operation for no more than 12 hours to perform the tests. Operation at 5% of rated power or less with less than 4 pumps operating is acceptable because such a power level embodies large conservatisms that provide adequate assurance that the ECCS criteria would be met.

5.1.2 Break Spectrum

Using the acceptable evaluation model described in the preceding section, the licensee provided in the January 30, 1976, March 8, 1976 and April 8, 1976 submittals the results of the analysis of a limited break spectrum. The worst break location was identified as a break in the cold leg at the pump discharge. The worst single failure, previously identified by a CE analysis, is the failure of a low pressure safety injection pump to start. ENC performed a series of break size calculations at that location and assuming the worst single failure. The calculations were performed for double ended guillotine breaks with discharge coefficients of 1.0, 0.8 and 0.6, and for split breaks with areas of 9.818 ft^2 (equivalent in area to the double ended guillotine break of the pump discharge line), 7.854 ft^2, and 5.891 ft^2.

From the results of the above calculations, it has been determined that the 9.818 ft^2 split break is most limiting. The maximum peak clad temperature was shown to be 2146°F which is below the acceptance limit of 2200°F as specified in 10 CFR 50.46(b). In addition, the maximum local metal/water reaction of less than 10% and the total core wide metal/water reaction of less than 1% were within the allowable limits of 17% and 1%, respectively. These calculations were done using a total peaking factor of 2.876. Based on this analysis, the licensee proposed to limit the peak linear heat generation rate (LHGR) to 14.19 kw/ft.

We have reviewed the above results and agree that the break spectrum has been defined sufficiently to assure that the worst break size and location for Palisades has been identified and analyzed. We find the break spectrum calculations acceptable. Therefore, it is our finding that operation with the reload core consisting of CE and ENC fuel assemblies is acceptable and meets the requirements of 10 CFR 50.46.

5.1.3 ECCS Containment Pressure Evaluation

The ECCS containment pressure calculations for Palisades were done using the ENC ECCS evaluation model. The NRC staff reviewed ENC's model and published a Safety Evaluation Report on September 11, 1975, and a Supplement on November 28, 1975. We concluded that ENC's containment pressure model was acceptable for ECCS evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for Palisades by letters dated July 9, 1975 and August 14, 1975. Consumers Power Company has reevaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for the ECCS analysis. This evaluation was based on measurements within the containment and from as-built drawings to which a margin was added. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Palisades is reasonably conservative and, therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

5.1.4 Single Failure Criterion

In CENPD 132, August 1974, Combustion Engineering described an analysis of the possible single failures that can occur within the ECCS. It was concluded that the worst single failure for the large break in Combustion Engineering plants was the loss of one of the low pressure pumps, and this assumption was used in the ECCS evaluation of Palisades performed by Exxon. The staff reported in its Status Report regarding the Combustion Engineering ECCS Evaluation Model, October 1974, that it found Combustion Engineering's generic evaluation of the single failure criterion acceptable but added that the satisfaction of the single failure criterion specified in Appendix K to 10 CFR 50 should be confirmed individually for each plant. The licensee has reviewed his plant with regard to single failures, and this review is documented in his July 9, 1975 and April 5, 1976 submittals. Proposed Technical Specification changes resulting from the single failure criterion review were also included in the April 5 submittal.

We have performed an evaluation of the Palisades Plant ECCS regarding the single failure criterion in the following specific areas:

1. Emergency Safeguards Actuation System
2. Onsite Emergency Power System
3. Electrical Interlocks
4. Qualification Status of Electrical Equipment
5. Electrical and Physical Separation Criteria
6. Electrically Operated Fluid System Components
7. Submerged Electrical Equipment

Following is a summary of our review.

5.1.4.1 Emergency Safeguards Actuation System

The Emergency Safeguards Actuation System (ESAS) is a protection system that initiates operation of various engineered safeguards equipment to mitigate the consequences of a Loss of Coolant Accident. The ESAS monitors two variables, low pressurizer pressure and high containment pressure, in order to detect the loss of integrity of the boundary of the reactor coolant system. The pressurizer and containment have four pressure instruments each in order to derive a safety injection signal (SIS). Each pressurizer pressure instrument and each containment pressure instrument are powered from one of four preferred A-C sources. The actuation logic is such that any two out of four pressurizer low-pressure or any two out of four containment high-pressure signals initiate the SIS which, in turn, actuates two redundant safety injection actuation circuits.

Based upon a review of the information the licensee has provided and of the previous evaluation of this system at the operating license review stage, we find this design meets the basic single failure criterion and is acceptable.

5.1.4.2 Onsite Emergency Power System

The onsite emergency power system supplies electrical power to the engineered safeguards equipment whenever there is a total loss of offsite power. The electrical power and control buses are divided into two channels and the loads into two groups. Each channel consists of the following buses and power sources: one 2400 volt bus, one 480 volt load center, one 480 volt motor control center, one D-C distribution center, one battery, two battery chargers, two preferred A-C buses, two inverters, and one diesel generator.

In addition, each channel is capable of furnishing power to equipment load groups which meet the minimum requirements to safely shut down the reactor. Furthermore, each channel is capable of providing sufficient electrical power to all functions necessary to operate the systems which mitigate the consequences of a Loss of Coolant Accident.

The design of the onsite power distribution system meets the fundamental single failure requirements with the exception of two cross train interties. One of these cross train interties is a swing bus. The other is a continuous cross channel connection. The staff has discussed the interties with the licensee. Resolution of these two items is described in Section 5.1.4.3, Electrical Interlocks.

5.1.4.3 Electrical Interlocks

An electrical interlock is used as a means of preventing redundant channels from being tied together, thus compromising electrical independence.

The licensee has identified three interties which connect redundant channels together. One intertie is an electrical interlock between the breakers connecting redundant 480 volt emergency buses together. This electrical interlock prevents manual initiation of breaker closure that could tie the redundant buses together.

Since a fault on either redundant bus would trip its incoming breaker and operator action might be to attempt to close the tie breakers which may cause transfer of the fault to the other redundant bus, the licensee has agreed(7)* to implement procedural changes that require the operator to clear the fault and close the incoming breaker prior to taking other action. If there is a fault that cannot be cleared, no attempt to close the tie breakers will be permitted.

With this change to the plant's administrative procedures, there is sufficient assurance that with a single failure of the interlock, redundant buses will not be compromised by manual operator action. We find this change acceptable.

The second intertie involves a swing bus arrangement which automatically transfers a non-safety related 120 volt instrument bus between two redundant safety related 480 volt motor control center (MCC) power supplies. Redundant 480 volt to 120 volt transformers are installed in each of the power supplies. If a fault is postulated on the instrument A-C bus (Y01), an automatic transfer on undervoltage may swing the Y01 bus from one safety-related 480 volt MCC to the redundant safety-related MCC. It may be possible for the automatic transfer scheme to reflect the fault to both redundant safety-related power sources.

*Letter dated April 7, 1976

This arrangement satisfies the general requirements of GDC-17, but is not in conformance with the more specific requirements of IEEE Std 308-1971 and the recommendations of Regulatory Guide 1.6. However, since both safety-related buses have protective circuit breakers upstream of this possible fault, the staff believes that the likelihood of this postulated fault adversely affecting plant safety is remote, and therefore acceptable for Cycle II. However, the licensee has agreed(10)* to (1) evaluate the impact on plant safety of the removal of the automatic transfer scheme on bus Y01 and (2) propose and implement a design change on the swing bus that meets the requirements of IEEE Std 308-1971 and the recommendations of Regulatory Guide 1.6, or submit additional justification for the present design. The staff will review and resolve this item prior to Cycle III operation.

The third intertie involves the D-C distribution system buses, D10 and D20, and the redundant 480 volt motor control center power supplies, MCC-1 and MCC-2.

Engineering Safeguards Channel 1 power is normally fed to D-C bus D10 from MCC-1 through battery charger No. 1. Channel 2 power is similarly fed to D-C bus D20 from MCC-2 through battery charger No. 2. However, there is also a continuous intertie between DC bus D20 of channel 1 and MCC-1 of channel 2 through battery charger No. 4.

*Letter dated April 15, 1976

Likewise, there is a continuous intertie between redundant DC bus D10 of channel 2 and MCC-2 of channel 1 through battery charger No. 3.

This concept of providing a continuous cross train bus tie adds little to the availability of power to the DC bus of either train. However, this continuous cross train bus tie does provide a greater likelihood of compromising the independence of redundant power sources. This type of design is not in conformance with the requirements of GDC-17, IEEE Std 308-1971 and with the recommendations of R.G. 1.6. The licensee has, therefore, agreed to open both breakers on either side of battery chargers Nos. 3&4 and use them as "spares" during normal plant operation.

In addition the licensee will amend his administrative procedures to prevent battery chargers Nos. 2&4 and battery chargers Nos. 1&3 from being tied to their respective 125 volt D-C buses simultaneously. However, the licensee will manually connect battery charger No. 3 to D-C bus D10 only after two circuit breakers in each feeder line of battery charger No. 1 are opened. Similarly, manual connection of battery charger No. 4 to D-C bus D20 will be permitted only after two circuit breakers in each feeder line of charger No. 2 are opened. These manual cross train ties are to be permitted only for the time required to repair or maintain the dedicated charger.

With the elimination of the continuous cross train bus tie through the battery charger, the staff finds the D-C distribution system design acceptable.

5.1.4.4 Qualification Status of Electrical Equipment

The qualification requirements for safety-related equipment are a measure of the equipment's ability to withstand the design basis seismic and environmental conditions. All safety-related equipment, controls and emergency electric power systems are Class 1. Class 1 equipment is that equipment (1) whose failure could cause uncontrolled release of radioactivity or (2) which is essential for immediate and long term operation following a Loss Of Coolant Accident. The equipment is designed to withstand the appropriate seismic loads simultaneously with other applicable loads without loss of function. In addition, a seismic disturbance will not affect the operation of safety systems.

The licensee has considered the ability of vital components, including electrical equipment and cables, to withstand the environment of the containment in the unlikely event of LOCA. Samples or prototypes of vital components in the Palisades Plant that would be required to operate in the containment accident environment were tested in such an environment. The simulated environmental conditions for these components included the application of heat, humidity, heat aging, pressure, shock and vibration over a controlled period of time. Samples of the electrical cabling used within the Palisades containment were also tested under simulated accident environmental conditions of radiation, temperature, humidity, and pressure.

These tests were completed satisfactorily when the plant was licensed for operation in 1971 and found by us to be an acceptable demonstration of the capability of safety-related components within the containment to withstand the post-LOCA environment. This is acceptable for Cycle II operation.

5.1.4.5 Electrical and Physical Separation Criteria

Engineered safeguards circuit separation includes separation of sensors, control and power devices, protective devices, power sources and the interconnecting cables.

Cable is carried in raceway systems consisting of rigid and flexible conduit, electromechanical tubing, galvanized steel cable tray, junction boxes, containment penetrations and raceways within equipment cabinets. Power, control and instrument cables are run in separate raceway systems, except for a few cases in which control and instrument cables occupy the same cable tray but are separated by a metal barrier. When power, control and instrument trays occupy the same area, they are arranged vertically with the power trays on top, the control trays in the center and the instrument cable trays on the bottom. A minimum vertical clearance of 1 foot is maintained between trays. All engineered safeguards raceways are located in tornado protected areas of the building or are embedded.

The raceways and containment penetrations for these circuits are divided into two systems. Where four channels are to be accommodated, a metal barrier is provided within the appropriate cable trays of each raceway system. All other parts of the system (conduits, junction boxes, containment penetrations, etc), as appropriate, are duplicated to form a total of four raceway subsystems. The interconnecting cables for any one channel are run in their respective raceway systems or subsystems.

Physical separation is maintained between the two raceway systems. This is provided by running the raceways in separate rooms, by providing such distance between them to assure that a single accident will not affect both raceways or by fireproof barriers between the raceways. Penetration of the barriers is minimized, but when penetration is necessary, the penetrating raceways are sealed.

The power source for driven equipment and the control power for that system are supplied from the sources in one channel.

Although Palisades does not meet the most recent separation criteria, there has been sufficient consideration given to the layout and separation of electrical cable and equipment so as to allow continued power operation.

5.1.4.6 Electrically Operated Fluid System Components

Each motor operated and air operated valve in the ECCS has been reviewed for compliance with Branch Technical Position EICSB-18 to determine if a single failure malfunction of the operator could have an adverse effect on the ECCS. In each case, the valve was assumed to fail to, or malfunction to, the most adverse position rather than the normal failed position.

In most cases, it was concluded that redundancy of systems and/or valves provides for proper functioning of the ECCS, with the qualifications discussed below.

5.1.4.6.1 Safety Injection Tank Isolation Valves

Removing power to the operators of motor operated valves while the valve is in the preferred position is an acceptable means for preventing malfunction in certain instances, and for this reason the licensee has proposed to remove power to the four motor operated safety injection tank isolation valves, MO-3041, -3045, -3049 and -3052 in the open position. The valves will be locked in the open position prior to achieving critical power by opening key lock switches in the control room and by locking open the breakers of the valve operators.

An inadvertent restoration of electric power to the valve operators would be indicated in the control room, and open valve position will be verified prior to reactor operation. The Technical Specifications for Cycle II have included provisions for the above requirements.

We conclude that this is an acceptable method for preventing malfunction of the safety injection tank isolation valves.

5.1.4.6.2 Mini-Flow Bypass Valves

In the event of a small break LOCA, the reactor coolant system pressure could remain relatively high for a period of time preventing flow in the low pressure safety injection pumps, resulting in overheating and damage to the pumps. To prevent overheating and pump damage, an orificed mini-flow bypass line has been provided which allows a small flow of coolant from the discharge of the low pressure safety injection pumps back to the refueling water storage tank. The mini-flow bypass line must be open during the injection phase of a LOCA until the reactor coolant system pressure falls below the shutoff head of the low pressure safety injection pumps. However, this line must be closed to allow isolation of the refueling water storage tank and containment during recirculation.

For this purpose, two in-series solenoid actuated, pneumatic valves have been provided in the mini-flow bypass line. Since a single failure resulting in the closure of either of these valves during pumped injection could result in pump damage, the licensee proposed a design modification in his submittal dated April 7, 1976.

The design modification is to electrically place two contacts in series in the solenoid valve coil such that both contacts will have to fail closed before valve motion will occur. One contact is to be energized by a low water level signal generated by 2 out of 4 logic whose inputs are derived from sensors on the safety injection refueling water tank (SIRWT). The other contact is energized by the control room operator manually closing a hand switch. Therefore, in order to energize the coil and cause valve closure, the contacts associated with the hand switch in conjunction with those associated with a low level in the SIRWT will have to close. The staff finds this design acceptable as an interim fix for making the mini-flow bypass lines single failure proof. However, for long term plant operation the licensee has agreed (10)* to implement a control circuit design change that precludes the possibility of undetected failures and one single failure causing spurious valve closure. In addition this design change must meet all the requirements of EICS BTP-18. The Technical Specifications

*Letter dated April 15, 1976

include our requirements regarding the mini-flow bypass line valves.

5.1.4.6.3 Shutdown Cooling Flow Control Valve

In the licensee's July 9, 1975 submittal, it was noted that a single failure of the Shutdown Cooling Flow Control Valve (CV-3006) to the closed position would result in a reduction in ECCS performance. CV-3006 is a solenoid actuated pneumatic valve and is located in the common discharge line from the two low pressure safety injection pumps. In a submittal dated February 25, 1976, the licensee proposed locking the valve in the open position by isolating the air supply. Since the valve is an air-to-close valve, this procedure would prevent inadvertent valve closure from a spurious electrical signal to the solenoid. Valve position indication in the control room would also be retained.

Since the long term cooling procedures designed to prevent precipitation of boric acid require that valve CV-3006 be closed, it will be necessary to restore the air supply at the time such procedures are required (about 12 hours following a LOCA). The licensee has verified that the air supply may be restored to the valve when required, and the Technical Specifications have been amended to provide for the above requirements. We conclude that the proposed operating procedures regarding control valve CV-3006 are acceptable.

5.1.4.7 Submerged Electrical Equipment

As the result of a Loss of Coolant Accident (LOCA), the containment sump will be filled with borated water. At the LOCA flood level certain electrical equipment will become submerged.

The licensee has surveyed the containment building, and all the electrical equipment which would be below the LOCA flood level has been identified.

Following is a summary of the staff's review of the information submitted by the licensee on this subject. Section 5.1.4.7.1 discusses the submergence of those motor operated valves within containment which are required following a LOCA. Section 5.1.4.7.2 is more general, addressing the overall effects of submergence, and includes our review of the plant breaker and fuse coordination scheme.

5.1.4.7.1 Submerged Valves Inside Containment

The licensee has surveyed the lower regions of the containment and reported in a submittal dated February 25, 1976 that the only motor operated valve that could become submerged in a post-LOCA environment is MO-3008. Valve MO-3008 is in the discharge line from the low pressure safety injection pumps to one of the four cold leg injection points and receives a safety injection actuation signal to open within seconds after a LOCA. The licensee has stated that submerging the valve would not affect its performance because it would not become submerged until well after it had opened (about 20 minutes following a LOCA). It is stated that if flooding the valve were to cause its failure, it would fail "as is" in the open position. We are in concurrence with the licensee's assessment, however we have evaluated the consequences of failing the valve to the closed position upon submergence and determined that even with the valve failed closed (including the worst single failure of loss of one low pressure safety injection pump), there is still adequate flow to cool the core.

We conclude that flooding of motor operated valve MO-3008 will not prevent proper operation of the ECCS.

5.1.4.7.2 Overall Effects of Electrical Equipment Submergence

The licensee has performed a detailed review of his equipment and its associated circuitry to determine the safety significance of this flooding including the effects on class IE electrical power sources serving this equipment. Breaker and fuse coordination was specifically addressed.

There are over 100 items including transmitters, switches, pumps, transformers, valves, and power supplies that will become submerged. Most are 120 volt low power equipment. Some are 480 volt equipment. Most items are not safety related; some items are.

Nevertheless, the licensee has indicated that the feeder breakers or fuses are designed such that they would open to clear the fault prior to the incoming bus breaker opening.

Based on the information submitted and our review thereof, the staff concludes that the design is acceptable for Cycle II with respect to mitigating the consequences of flooding of electrical equipment due to a LOCA.

5.1.4.8 Conclusions

The licensee has performed a single failure analysis of the ECCS and the staff has evaluated the submittal. Based upon an evaluation of this submittal and the discussions in the preceding sections, the Emergency Core Cooling System meets the basic single failure requirements for the Emergency Safeguards Actuation System.

With certain design and Technical Specification changes as discussed in the preceding sections the following areas meet the single failure requirements.

1. Onsite Emergency Power System
2. Electrically Operated Fluid System Components
3. Electrical Interlocks

Based on the licensee's submittal, the staff has determined that the design in the following areas is acceptable for the next refueling interval.

1. Qualification Status of Electrical Equipment
2. Submerged Electrical Equipment

The licensee has designed Palisades so that electrical and physical separation criteria in effect at the time of its design were met. This is acceptable for the next refueling interval. In conjunction with a review of the adequacy of fire protection for all nuclear power plants, the licensee will be required to further evaluate cable separation during Cycle II.

5.1.5 Boric Acid Concentration Effects During Post LOCA Long Term Cooling

The ECCS is required to provide adequate cooling for the reactor core following a LOCA. Long term removal of residual heat is provided by continuous evaporation of liquid in the reactor vessel which may result in high concentrations of boric acid and other materials in the vessel. If the solubility limit is exceeded, precipitation of boric acid will occur resulting in possible blockage of the coolant flow paths and a degradation in cooling capability.

At the request of the NRC staff, the licensee reviewed his ECCS equipment and emergency operating procedures and submitted for the staff's review submittals dated June 27 and August 20, 1975 including results of computer analyses and proposed emergency operating procedures designed to prevent an excessive buildup of boric acid during post LOCA long term cooling. The licensee included in his submittals some proposed system modifications which would enhance the reliability of his long term cooling system. The staff evaluation is discussed below.

The procedures proposed by the licensee require that for the first 12 hours after LOCA, boric acid solution be injected into the cold leg. After that time, the cold leg injection should be supplemented, either by suction from the hot leg through the shutdown cooling line (hot leg suction), or by injection through the pressurizer auxiliary spray line (hot leg injection). These modes of operation will assure sufficient flushing through the core and will prevent the buildup of boric acid. (The licensee's analyses indicate that a core flushing rate of 20 gpm is sufficient to prevent the precipitation of boron, while his proposed post-LOCA long term cooling procedures specify a minimum flushing flow of 100 gpm for either the primary flushing method, hot leg suction, or for the backup method, hot leg injection).

The staff has performed independent calculations which have confirmed that the licensee's analyses were conservative.

In the licensee's May 22 and June 27, 1975 submittals, it was reported that a review was being conducted to determine the post-LOCA environmental qualifications of the components required in the long term cooling procedures aimed at preventing boron precipitation. In a letter dated August 27, 1975, the licensee identified eleven areas in which further equipment qualification and/or modification would be required. A qualification program was also proposed, and the licensee's progress with this program has been reported in subsequent status reports dated November 5, 1975 and March 8, 1976. Prior to startup of Cycle II, the licensee completed qualification of the equipment required for the primary method used in preventing boron precipitation (hot leg suction method). This involved qualifying the motors and power cables to two motor operated valves. The licensee has stated that he has been unable to obtain some of the valve replacement parts and electrical components required for completing the qualification of the back-up procedure (hot leg injection through the pressurizer) prior to the scheduled Cycle II startup. The equipment has been ordered, and the licensee has stated that upon receiving the necessary parts, the required modifications will be completed either during a power outage or the next refueling outage.

In addition to the above, an evaluation is being performed of electrical power cable separation between the two systems proposed for preventing boron precipitation. The licensee intends to report his findings on this subject in the near future along with a program for correcting any deficiencies in cable separation.

We have reviewed the boron concentration evaluation performed by the licensee, and we conclude that the proposed methods for operation of the ECCS will prevent exceeding boron solubility limits within the reactor vessel during post-LOCA long term cooling. The licensee is proceeding with the program defined for improving the reliability of his backup system and progress reports are being provided to the staff on a bimonthly basis.

5.1.6 Conclusions

Based on our review, we conclude that:

1. The ECCS cooling performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR Part 50.46(b). In addition the plant will also conform to the two remaining criteria, i.e., the maintenance of a coolable geometry and long term cooling.

2. The licensee's submittals regarding the containment pressure calculations are acceptable.
3. The information submitted by the licensee along with certain design and Technical Specification changes demonstrate conformance to the single failure criterion as discussed in the previous sections.

4. Adequate systems and procedures exist to provide long term cooling to the reactor vessel and prevent boron precipitation.

5.2 Rod Ejection Incident

The licensee has determined that the ejected rod worth for Core II will be within the limits analyzed in the Cycle I analysis. This worth will be confirmed experimentally during the Cycle II startup program. This is acceptable.

5.3 Rod Drop Incident

The licensee has determined that the rod drop parameters for Core II will be within the limits analyzed previously in the Cycle I analysis. The worth of the dropped rod will be checked experimentally during the startup program. This is accceptable.

Rod Withdrawal Incident

The licensee and Exxon have reanalyzed the rod withdrawal transient from full power using the Exxon PTSPWR2 code.[13] The staff has reviewed this code, together with the additional information submitted in support of the Palisades transient and accident analysis.[5,9] We find this analysis acceptable.

The rod withdrawal was not reanalyzed from low power because of the more restrictive rod insertion limits in effect for Cycle II. The minimum DNBR and the maximum system pressure calculated for the full-power case were 1.69 and 1822 psia respectively. These values are well within the design bounds, and the analysis is therefore acceptable.

5.5 Loss of Coolant Flow Incidents

The analysis of the reference cycle showed the loss of coolant flow incidents, pump coastdown and locked rotor, to be the most limiting with respect to DNB. ENC's reanalysis of these incidents resulted in MDNBR's of 1.62 and 1.39 for the pump coastdown and locked rotor cases respectively. Of the transients and accidents reanalyzed by ENC, the loss of flow- locked rotor incident resulted in the highest pressurizer pressure, with a value of 1909 psia. This is well below the Technical Specification limit of 2750 psia and the analysis is acceptable.

5.6 Loss of Load Incident

The loss of load incident was analyzed for the reference cycle using the most limiting parameters during the core life. This incident was limiting with respect to system pressure. For Cycle II the system operating pressure is 1800 psia while the reference cycle pressure was 2100 psia. The other Cycle II input parameters are also more favorable than those for the reference cycle. Therefore, the results for the reload cycle will be bounded by those for the reference cycle.

5.7 Other Transients and Accidents

The remaining transients and accidents in the licensee's FSAR are either not affected by the proposed core design changes or the input parameters are less favorable than for the reload cycle. Therefore, the results for the reload cycle will be bounded by those for the reference cycle. We find this acceptable.

6.0 Conclusion

Based on our evaluation of the application and available reload information as set forth above, and subject to the requirements set forth above, we conclude that it is acceptable for the licensee to proceed with Cycle II operation in the manner proposed.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level beyond that previously authorized and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

DATE:

7.0 References

1. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model, Palisades Type Combustion Engineering Plant Large Break Example Problem", XN-75-64, December 7, 1975.

2. "Generic Fuel Design for 15 x 15 Reload Assemblies for Palisades", XN-75-59, November 1975.

3. "Palisades Cycle 2 Reload Fuel Licensing Data Submittal", XN-75-66, December 15, 1975.

4. "Palisades LOCA Analyses Using the ENC WREM-Based PWR ECCS Evaluation Model", XN-76-4, January 1976 and Supplement 1, February 1976.

5. "Plant Transient Analysis for the Palisades Nuclear Power Plant", XN-75-67, December 23, 1975.

6. "Analysis of the Palisades DNB and Mixing Test", XN-75-65(P), January 14, 1976.

7. "Palisades Thermal Hydraulic Design Report for Cycle 2 Core (Batches A, B, C1, D & E Fuels)", XN-76-3(P), February 1976.

8. "Palisades Thermal Hydraulic Design Report for Cycle 2 Core, Supplement 1: Analysis of Core Flow Reduction Due to Anticipated Steam Generator Tube Plugging", XN-76-3(P) Supp 1, March 1976.

9. Letters from David A. Bixel, CPC, to Robert A. Purple, NRC, dated August 14, 1975, August 20, 1975, August 27, 1975, November 5, 1975, December 15, 1975, January 30, 1976, February 6, 1976, February 25, 1976, March 8, 1976, March 20, 1976, March 23, 1976, March 26, 1976, March 29, 1976, April 5, 1976, April 8, 1976, and April 15, 1976.

10. Letter from Ralph B. Sewell, CPC, to Robert A. Purple, NRC, dated May 22, 1975, June 27, 1975, July 9, 1975, April 7, 1976 and April 15, 1976.

11. Letter from J. V. Strimaitis, CE, to Robert A. Purple, NRC, dated July 9, 1975.

12. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model", XN-75-41, Volumes I, II, III, Supplements 1-7.

13. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)", XN-74-5 Rev. 1, May 15, 1975.

14. "Definition and Justification of Exxon Nuclear Company DNB Correlation for PWR's", XN-75-48, October 6, 1975.

15. "Safety Evaluation Report Regarding Review of the Exxon Nuclear Company Pressurized Water Reactor Generic ECCS Codes and the H. B. Robinson Reactor ECCS Evaluation Model for Conformance to All Requirements of Appendix K to 10 CFR 50 by the Office of Nuclear Reactor Regulation", USNRC, September 11, 1975 and Supplement No. 1, November 28, 1975.

16. "Status Report by the Directorate of Licensing in the Matter of Combustion Engineering, Inc., ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K", October 1974.

17. "Calculational Methods for the CE Large Break LOCA Evaluation Model", CENPD-132P, August 1974.

18. "Calculative Methods for the CE Small Break LOCA Evaluation Model", CENPD-137, August 1974.

19. "Effects of Fuel Densification Power Spikes on Clad Thermal Transients", WCAP 8359, August 1974.

20. "Fuel Densification Experimental Results and Model for Reactor Application", WCAP 8219, March 1975.

21. "Effect of a Bowed Rod on DNB", WCAP 8176, May 4, 1974

22. "XPOSE, The Exxon Nuclear Revised LEOPARD," XN-CC-21 Rev. 2, April 1975.

23. "XTG: A Two-Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing," XN-CC-28 (A) Rev. 3, January 1975.

24. "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors," XN-75-27, June 1975.

25. "Core Physics Methods and Data used as Input to LOCA Analyses," XN-75-43 (A), August 1975.

26. "XTRAN-PWR: A Computer Code for the Calculation of Rapid Transients in Pressurized Water Reactors with Moderator and Fuel Temperature Feedback," XN-CC-32, October 1975.

27. Letter from G. F. Owsley, ENC, to R. A. Purple, NRC, dated April 16, 1976.

28. K. R. Merckx, "Cladding Collapse Calculational Procedure," JN-72-23, November 1972.

29. K. P. Galbraith, "GAPEX: A Computer Code Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," XN-73-25, August 13, 1973.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Provisional Operating License No. DPR-20 issued to Consumers Power Company which revised Technical Specifications for operation of the Palisades Plant, located in Covert Township, Van Buren County, Michigan. The amendment is effective as of its date of issuance.

The amendment (1) revises provisions in the Technical Specifications related to the replacement of fuel assemblies in the Palisades core with fuel assemblies of a different design, constituting refueling of the core for operation with Cycle 2 at power levels up to 2200 MWt (100% power), (2) incorporates operating limits in the Technical Specificatons based on an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46, (3) modifies various limits established in accordance with the Commission's Interim Acceptance Criteria, and (4) terminates the further restrictions imposed by the Commission's December 27, 1974 Order for Modification of License, and imposes instead limitations established in accordance with the Commission's Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 10 CFR Section 50.46.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Provisional Operating License in connection with this action was published in the FEDERAL REGISTER on February 23, 1976 (41 FR 8002). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated July 9, 1975, and January 30 and April 5, 1976, as supplemented and amended, (2) Amendment No. 21 to License No. DPR-20, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. 20555 and at the Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan 49006.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 29th day of April 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors