DISTRIBUTION
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NRC PDR
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VStello
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SEP 08 1978

Docket No. 50-255

Consumers Power Company
ATTN: Mr. David Bixel
Nuclear Licensing Administrator
212 West Michigan Avenue
Jackson, Michigan 49201

Gentlemen:

The Commission has issued the enclosed Amendment No. 43 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your requests dated June 15, 1978, as supplemented by letters dated August 15, 21 and 25, 1978.

This amendment changes the Palisades Technical Specifications relating to the limits on axial power distribution.

Copies of our Safety Evaluation and the Notice of Issuance also are enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 43 to License No. DPR-20
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OFFICE	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2		
SURNAME	HSmith	RDSilver:ah	BBerson	DLZiemann		
DATE	8/1/78	9/1/78	9/7/78	9/8/78		

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

September 8, 1978

Docket No. 50-255

Consumers Power Company
ATTN: Mr. David Bixel
Nuclear Licensing Administrator
212 West Michigan Avenue
Jackson, Michigan 49201

Gentlemen:

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U. S. Environmental Protection
Agency
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230 South Dearborn Street
Chicago, Illinois 60604

*(w/cy. of CPC filings 6/15/78, 8/15/78, 8/21/78 and 8/25/78)

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. 43
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated June 15, 1978, as supplemented by letters dated August 15, August 21 and August 25, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 43, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 43

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the following pages and by inserting the enclosed pages. The revised pages contain the captioned amendment number and marginal lines indicating the area of change.

Remove	Insert
1-2	1-2
1-3	1-3
1-4	1-4
2-2	2-2
3-58	3-58
3-59	3-59
3-63	3-63
3-64	3-64
3-66	3-66
--	3-66a
3-86	3-86 (Blank Page)
3-87	3-87
3-87a	--

1.1 REACTOR OPERATING CONDITIONS (Contd)

Low Power Physics Testing

Testing performed under approved written procedures to determine control rod worths and other core nuclear properties. Reactor power during these tests shall not exceed 10^{-2}% of rated power, not including decay heat and primary system temperature and pressure shall be in the range of 260°F to 538°F and 415 psia to 2150 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions therefore in these Technical Specifications.

Shutdown Boron Concentrations

Boron concentration sufficient to provide $k_{eff} \leq 0.98$ with all control rods in the core and the highest worth control rod fully withdrawn.

Refueling Boron Concentration

Boron concentration of coolant at least 1720 ppm (corresponding to a shutdown margin of at least 5% $\Delta\rho$ with all control rods withdrawn).

Quadrant Power Tilt

The difference between nuclear power in any core quadrant and the average in all quadrants.

Assembly Radial Peaking Factor - F_r^A

The assembly radial peaking factor is the maximum ratio of individual fuel assembly power to core average assembly power integrated over the total core height, including tilt.

Total Radial Peaking Factor - F_r^T

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

1.2 PROTECTIVE SYSTEMS

Instrument Channels

One of four independent measurement channels, complete with the sensors, sensor power supply units, amplifiers and bistable modules provided for each safety parameter.

Reactor Trip

The de-energizing of the control rod drive mechanism (CRDM) magnetic clutch holding coils which releases the control rods and allows them to drop into the core.

1.2 PROTECTIVE SYSTEMS (Contd)

Reactor Protective System Logic

This system utilizes relay contact outputs from individual instrument channels to provide the reactor trip signal for de-energizing the magnetic clutch power supplies. The logic system is wired to provide a reactor trip on a 2-of-4 or 2-of-3 basis for any given input parameter.

Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

Engineered Safety Features System Logic

This system utilizes relay contact outputs from individual instrument channels to provide a dual channel (right and left) signal to initiate independently the actuation of engineered safety feature equipment connected to diesel generator 1-2 (right channel) and diesel generator 1-1 (left channel). The logic system is wired to provide an appropriate signal for the actuation of the engineered safety feature equipment on a 2-of-4 basis for any given input parameter.

1.3 INSTRUMENTATION SURVEILLANCE

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. This determination shall, where feasible, include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, interlocks or trip and shall be deemed to include the channel functional test.

1.4 MISCELLANEOUS DEFINITIONS

Operable

A system or component is operable if it is capable of fulfilling its design functions.

Operating

A system or component is operating if it is performing its design functions.

1.4 MISCELLANEOUS DEFINITIONS (Contd)

Control Rods

All full-length shutdown and regulating rods.

Containment Integrity

Containment integrity is defined to exist when all of the following are true:

a. All nonautomatic containment isolation valves and blind flanges are closed.

b. The equipment door is properly closed and sealed.

c. At least one door in each personnel air lock is properly closed and sealed.

d. All automatic containment isolation valves are operable or are locked closed.

e. The uncontrolled containment leakage satisfies Specification 4.5.1.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μC/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

$\overline{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

Safety

Safety as used in these Technical Specifications refers to those safety issues related to the nuclear process and for example does not encompass OSHA considerations.

probability at a 95% confidence level than DNB will not occur which is considered an appropriate margin to DNB for all operating conditions.[1] The curves of Figures 2-1, 2-2 and 2-3 represent the loci of points of thermal power, primary coolant system pressure and average temperature of various pump combinations for which the DNBR is ≥ 1.3. The area of safe operation is below these lines. For 3- and 2-pump operation, the limiting condition is void fraction rather than DNBR. The void fraction limits assure stable flow and maintenance of DNBR greater than 1.3. Flow maldistribution effects of operation under less than full primary coolant flow have been evaluated via model tests.[2] The flow model data established the maldistribution factors and hot channel inlet temperatures for the thermal analyses that were used to establish the safe operating envelopes presented in Figures 2-1 and 2-2. These figures were established on the basis that the thermal margin for part-loop operation should be equal to or greater than the thermal margin for normal operation.

The reactor protective system is designed to prevent any anticipated combination of transient conditions for primary coolant system temperature, pressure and thermal power level that would result in a DNBR of less than 1.3.[3]

References

[1] FSAR, Section 3.3.3.5.
[2] FSAR, Section 3.3.3.3, Appendix C.
[3] FSAR, Section 14.1.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to operation of control rods and hot channel factors during operation.

Objective

To specify limits of control rod movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.

Specifications

3.10.1 Shutdown Margin Requirements

a. With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 2%.

b. With less than four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 3.75%.

c. At less than the hot shutdown condition, boron concentration shall be shutdown boron concentration.

d. If a control rod cannot be tripped, shutdown margin shall be increased by boration as necessary to compensate for the worth of the withdrawn inoperable rod.

e. The drop time of each control rod shall be no greater than 2.5 seconds from the beginning of rod motion to 90% insertion.

3.10.2 Individual Rod Worth

a. The maximum worth of any one rod in the core at rated power shall be equal to or less than 0.6% in reactivity.

b. The maximum worth of any one rod in the core at zero power shall be equal to or less than 1.2% in reactivity.

3.10.3 Power Distribution Limits

a. The linear heat generation rate at the peak power elevation z shall not exceed:

15.28 kW/ft x $F_A(Z)$ for ENC fuel types

14.12 kW/ft x $F_A(Z)$ for D type fuel

where the function $F_A(Z)$ is shown in Figure 3.9. If the power distribution is double peaked, both peaks shall satisfy the criterion. Appropriate consideration shall be given to the following factors:

(1) A flux peaking augmentation factor of 1.0,

(2) A measurement calculational uncertainty factor of 1.10,

3.10.3 Power Distribution Limits (Contd)

(3) An engineering uncertainty factor (which includes fuel column shortening due to densification and thermal expansion) of:
1.03 for ENC fuel types and
1.05 for D-type fuel

(4) A thermal power measurement uncertainty factor of 1.02.

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

(1) The linear heat generation rate shall promptly be demonstrated to be less than that specified in Part a, or

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

c. 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.

d. If the power in a quadrant exceeds the core average by 15% and if the linear heat generation rate cannot be demonstrated promptly to be within 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.

e. 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.

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

g. The calculated value of F_r^A shall be limited to ≤ 1.45 $(1.0 + 0.5\ (1 - P))$ and the calculated value of F_r^T shall be limited to $\leq 1.77\ (1.0 + 0.5\ (1 - P))$, where P is the core thermal power in fraction of core rated thermal power (2,530 MW_t).

Basis (Contd)

rotor will not exceed acceptable limits.[5][7] The axial power distribution term ensures that the operating power distribution is enveloped by the design power distributions. Appropriate factors for measurement-calculational uncertainty, engineering factor and shortening of the fuel pellet stack are specified to ensure that the linear heat generation rate limit is not exceeded.

When a flux tilt exists for a sustained time period (24 hours) and cannot be corrected or if a flux tilt reaches 20%, reactor power will be reduced until the tilt can be corrected. A quadrant to core average power tilt may be indicated by two methods: Comparison of the output of the upper or lower sections of the ion chamber with the average value 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.

The insertion of part-length rods into the core, except for rod exercises or physics tests, is not permitted since it has been demonstrated on other CE plants that design power distribution envelopes can, under some circumstances, be violated by using part-length rods. Further information may justify their use. Part-length rod insertion is permitted for physics tests, since resulting power distributions are closely monitored under test conditions. Part-length rod insertion for rod exercises (approximately 6 inches) is permitted since this amount of insertion has an insignificant effect on power distribution.

For a control rod misaligned up to 8 inches from the remainder of the banks, hot channel factors will be well within design limits. If a control rod is misaligned by more than 8 inches, the maximum reactor power will be reduced so that hot channel factors, shutdown margin and ejected rod worth limits are met. If in-core detectors are not available to measure power distribution and rod misalignments > 8 inches exist, then reactor power must not exceed 75% of rated power to insure that hot channel conditions are met.

The limitations on F_r^A and F_r^T are provided to ensure that the assumptions used in the analysis for establishing the DNB margin, linear heat rate, thermal margin/low pressure and high power trip set points remain valid during operation at the various allowable control rod group insertion limits.

Continued operation with that rod fully inserted will only be permitted if the hot channel factors, shutdown margin and ejected rod worth limits are satisfied.

In the event a withdrawn control rod cannot be tripped, shutdown margin requirements will be maintained by increasing the boron concentration by an amount equivalent in reactivity to that control rod. The deviations permitted by Specification 3.10.7 are required in order that the control rod worth values used in the reactor physics calculations, the plant safety analysis, and the Technical Specifications can be verified. These deviations will only be in effect for the time period required for the test being performed. The testing interval during which these deviations will be in effect will be kept to a minimum and special operating precautions will be in effect during these deviations in accordance with approved written testing procedures.

Violation of the power dependent insertion limits, when it is necessary to rapidly reduce power to avoid or minimize a situation harmful to plant personnel or equipment, is acceptable due to the brief period of time that such a violation would be expected to exist, and due to the fact that it is unlikely that core operating limits such as thermal margin and shutdown margin would be violated as a result of the rapid rod insertion. Core thermal margin will actually increase as a result of the rapid rod insertion. In addition, the required shutdown margin will most likely not be violated as a result of the rapid rod insertion because present power dependent insertion limits result in shutdown margin in excess of that required by the safety analysis.[5]

References

(1) FSAR, Section 14.

(2) FSAR, Section 3.3.3.

(3) FSAR, Section 7.4.2.2.

(4) FSAR, SEction 7.3.3.6.

(5) XN-NF-77-18.

(6) XN-NF-77-24.

(7) XN-NF-78-16

Specification (Contd)

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 set point, the action specified in 3.11.b shall be taken.

g. F_r^A and F_r^T shall be determined whenever the core power distribution is evaluated. If either F_r^A or F_r^T is found to be in excess of the limit specified in Section 3.10.3.g; within six hours thermal power shall be reduced to less than $[(1.77 \div F_r^T) \times 2530\ MW_t]$ or $[(1.45 \div F_r^A) \times 2530\ MW_t]$, whichever is lower.

Basis

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.

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

3.11 IN-CORE INSTRUMENTATION (Contd)

Basis (Contd)

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.

Reference

(1) FSAR, Section 7.4.2.4.

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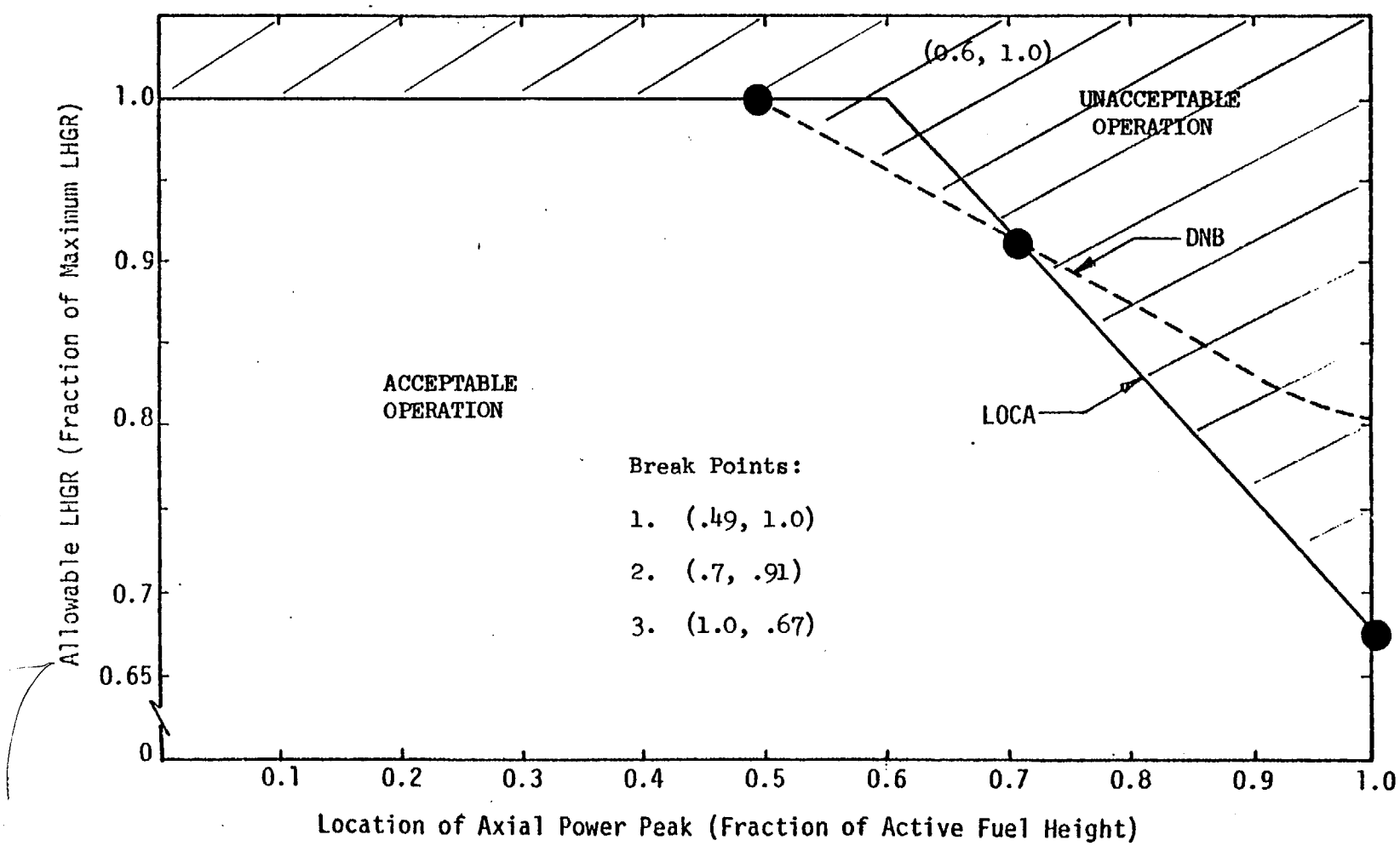

Allowable LHGR as a function of peak power location.	Palisades Technical Specifications	Figure 3-9

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 43 TO LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 Introduction

By letter dated June 15, 1978,[1] and supplemented by letters dated August 15, 21 and 25,[2,3,11] Consumers Power Company (the licensee) requested amendment of the Palisades Technical Specifications. The effect of this amendment is to extend the axial power distribution envelope to allow a higher percentage of the total power to be generated in the upper portion of the core.

The licensee has supported this amendment by examining the effect of the revised allowed axial power distributions on the accident and transient analyses. The details are discussed in the following sections.

2.0 Nuclear Design

The neutronic analysis performed by the licensee in support of this amendment consisted of two independent parts: a verification of the ability of the axial power distributions assumed in the accident and transient analyses to bound actual operating conditions, and a recalculation of core kinetics parameters assuming power distributions peaked high in the core.

2.1 Axial Power Distribution X/L Correlation

The licensee used the data base from the Exxon power distribution control study for PWRs.[4,5] This data base consists of more than 2000 axial power distributions produced by simulated maneuvers, and (within certain limitations enforced by the Technical Specifications) bounds all possible states of the reactor throughout the cycle. The licensee examined each axial power distribution with a positive axial offset, and verified that the axially integrated power fraction above the peak power plane (the upper peak power plane for double-peaked distributions) was within 10% of the fraction of active fuel length

above the peak power plane. That is, the fraction of total power generated in the volume above the peak power plane is always bounded by the expression:

$$(1.00 \pm 0.10)\ (1 - X/L)$$

where X is the height of the peak power plane and L is the total core height. On this basis the staff concludes that the axial power distribution X/L correlation is acceptable for use in the accident and transient analyses.

2.2 Impact on Core Kinetics Parameters

The upper portion of the reactor core operates at higher temperatures than does the bottom. Because axial power distributions peaked high in the core give more importance weighting to the upper portion, the core-average kinetics parameters may be altered by these axial power distributions. The licensee has re-calculated the kinetics parameters using the same methods as previously and assuming a range of axial power distributions, as was done for the accident and transient analyses.

The results of the calculations showed no significant change in Doppler coefficient, delayed neutron fraction, and rod worths; a better scram reactivity function; and a more negative moderator temperature coefficient. The licensee then reanalyzed selected accidents and transients to verify that the 115% overpower analysis continues to be bounding. Further discussion of this reanalysis is contained in Section 3.0.

On this basis the staff concludes that the core kinetics parameters used in the accident and transient analyses are acceptable.

3.0 Thermal and Hydraulic Design

The licensee's thermal-hydraulic analysis for the revised axial power profiles[1] shows that the minimum departure from nucleate boiling ratio (MDNBR) is never less than the allowed 1.30 for normal operation and anticipated transients. The licensee, using the thermal-hydraulic code XCOBRA-III C,[6] established the axial power distribution limits such that the MDNBR $\geq$ 1.30 at the steady state, 115% (2910 MWt) overpower design condition. Additional analyses confirmed that the 115% overpower condition enveloped the expected decrease in thermal margin during operational transients. The transient analyses were initiated from 102% power (2580.6 MWt) and the results of these analyses are discussed in the following section.

4.0 Transient and Accident Analyses

The postulated transients and accidents reported in XN-NF-77-18, (7), that would be affected by the revised axial power distribution envelope were reanalyzed by the licensee. Table 4.1 lists those events which were reanalyzed. These events were all analyzed for an axial power peaked at 0.9 of active fuel height. The limiting transients (loss of load and 4-pump coastdown) were also analyzed with the axial power peaked at 0.7 and 0.8 of active fuel height. These two events were further analyzed for the power peaked at 0.7 of active fuel height with the integrated power above the peak skewed by ± 10%.

The results of these analyses are summarized in Table 4.2. The licensee states that for a given event the plant transient response followed the same trend for each of the power distributions. As can be seen from Table 4.2, the MDNBR is not very sensitive to these changes in power distribution. The licensee's analysis confirmed that for operational transients the MDNBR is never less than 1.30 and the transients are bounded by the 115% overpower design condition. The licensee has demonstrated, to our satisfaction, using standard ENC calculational methods, that the consequences of these transients and accidents are acceptable.

The licensee states that for the events which were not reanalyzed the reference analyses are not adversely affected by the revised power distribution limits. The staff has reviewed the reference analyses and methodology agrees that the consequences of the events not reanalyzed would not be adversely affected by the revised power distribution limits because the key safety parameters for these events are not affected. Therefore, we conclude that the consequences of events not reanalyzed continue to be acceptable.

TABLE 4.1

REANALYZED TRANSIENTS AND ACCIDENTS

Control Rod Withdrawal
Control Rod Drop
Excessive Load
Loss of Load
Single Rod Withdrawal
Four Pump Coastdown
Locked Rotor
Steam Line Break

TABLE 4.2

Event	MDNBR 0.6*	0.7	0.8	0.9
Rod Withdrawal @ 1.25 x 10^{-4} $\Delta\rho$/sec from 102% power (EOC)	1.46			1.45
Rod Drop Case 1 (BOC)†	1.35			1.71
Rod Drop Case 2 (EOC)†	1.40			1.47
Four Pump Coastdown	1.39	** (1.41) 1.43 (1.45)	1.41	1.43
Locked Rotor	1.27			1.33
Excessive Load	1.74			1.67
Loss of Load Case 1 ††	1.57			1.53
Loss of Load Case 2 ††	1.39	(1.43) 1.44 (1.47)	1.43	1.41
Steam Line Break	1.30			1.34
Single Rod Withdrawal	1.44			1.50

* 0.6 (0.7, 0.8, 0.9) = fraction of active fuel height at which peak power occurs. The results for the 0.6 height case are from XN-NF-77-18.

** Top value: Skewing factor = 1.1
Middle value: Skewing factor = 1.0
Bottom value: Skewing factor = 0.9

Rod Drop Case 1, dropped rod worth ($\Delta\rho$) = -0.12, F_R = 1.64, EOC.

Case 2, dropped rod worth ($\Delta\rho$) = -0.12, F_R = 1.66, BOC.

† Loss of Load Case 1, initiated from primary pressure 2110 (psia), pressurizer relief valve and spray inoperable, atmospheric steam dump and condenser bypass operable.

†† Loss of Load Case 2, initiated from primary pressure 2010 (psia), pressurizer relief valve and spray operable, atmospheric steam dump and condenser bypass inoperable.

5.0 ECCS Performance Evaluation

The LOCA analysis presented in XN-NF-77-24[8] identified the limiting break to be a double ended guillotine break located in the pump discharge line with a discharge coefficient of 0.6. The licensee has reanalyzed this break with the axial power peaked at 0.6, 0.7, 0.8, and 0.9 of core height. In addition, the break was analyzed for the power peaked at 0.8 core height with the integrated power above the peak skewed by ±10%.

The radial peaking factor for these analyses has been decreased to 1.45 from the 1.50 value used in XN-NF-77-24.[8] This lower radial peaking factor is now consistent with the value used in the DNB analyses. Reducing the radial peaking factor results in better fluid conditions and heat transfer during a LOCA. This, in turn, allows the axial peaking factor to be increased more than enough to offset the decrease in radial peaking. Therefore, the maximum allowable LHGR limit is increased from 14.68 KW/ft to 15.28 KW/ft (including 2% power measurement uncertainty).

The licensee's LOCA analysis assumed that an additional 500 steam generator tubes were plugged beyond the present conditions. The analysis, therefore, provides considerable conservatism with regard to primary coolant flow under the present steam generator conditions and also allows for some additional plugging should it become necessary.

The results of these analyses are summarized in Table 5.1. As can be seen from the Table, as the location of peak rises in the core, the peak linear heat generation rate (PLHGR) is reduced. This is necessary since it takes longer for the ECCS fluid to quench the higher core elevations during reflood. As indicated, the predicted values of peak clad temperature, local clad oxidation, and hydrogen generation are below their respective limits of 2200°F, 17% and 1% as specified in 10 CFR 50.46(b).

Small break LOCAs were generically evaluated for Combustion Engineering 2560 MWt series plants in CENPD-137P.[9] This report concludes that the small break is much less limiting than the large break. This generic report is applicable to Palisades with the revised power profiles since the report assumes an axial power distribution peaked at 0.85 core height with a PLHGR of 15.25 KW/ft. This is conservative for Palisades because the revised Technical Specifications will limit the LHGR at 0.85 core height to less than 12.0 KW/ft and for small breaks a power distribution peaked high in the core is limiting.

Based on our review, we conclude that the Palisades Plant can operate with the revised power distribution limits for the remainder of Cycle 3 and will conform to the peak clad temperature, maximum local oxidation, hydrogen generation, coolable geometry, and long term cooling criteria of 10 CFR 50.46(b), provided that the PLHGR limits in Section 3.10.3a of the revised Technical Specifications are not exceeded.

TABLE 5.1

RESULTS OF ECCS CALCULATIONS

Peak Location (inches)	(X/L)	LHGR (KW/ft)	Axial (F_z)	Skewing Factor	PCT °F	Local Clad Oxidation	Hydrogen Generation
79.08	0.6	15.0	1.51	1.00	2081	<10%	<1.0%
92.26	0.7	13.7	1.39	1.00	2155	<12%	<1.0%
105.44	0.8	12.6	1.27	1.00	2154	<12%	<1/0%
118.62	0.9	11.3	1.15	1.00	2135	<13%	<1.0%
105.44	0.8	12.6	1.27	0.90	2112	<11%	<1.0%
105.44	0.8	12.6	1.27	1.10	2172	<13%	<1.0%

6.0 Technical Specification Changes

6.1 Definitions for assembly radial peaking factor F_r^A and total radial peaking factor F_r^T have been added to Section 1.1. These quantities are used in the new Section 3.10.3.g, discussed below and are acceptable.

6.2 References to hot channel factors have been deleted from the bases for Section 2.1. The appropriate information is now included in the basis for the new version of Section 3.10.3. This is acceptable because the hot channel factor limitations are limiting conditions for operation (LCO) and are more appropriately discussed in LCO sections than safety limit sections such as Section 2.1.

6.3 Two changes have been made in Section 3.10.3.2 as follows:

6.3.1 The linear heat generation rate (LHGR) limit has been increased to 15.28 KW/ft X $F_A(Z)$ for ENC fuel and to 14.12 KW/ft X $F_A(Z)$ for D type fuel, where the $F_A(Z)$ function is an axial shaping function. This function is shown in Figure 3.9 of the Technical Specifications. These KW/ft limits and the new $F_A(Z)$ function are the primary subject of this Technical Specification change request, and are acceptable for the reasons provided under the discussions of Thermal and Hydraulic Design, Transient and Accident Analyses, ECCS Performance Evaluation, and Neutronics Analysis.

6.3.2 The engineering uncertainty factor of 1.03 for has been modified to include an allowance of 0.87% for fuel column shortening due to densification and thermal expansion. This change is acceptable because on pp 93-97 of Reference 10, the licensee shows that his engineering uncertainty factor for ENC fuel is 1.021 for a statistical combination of the manufacturing tolerances and product specifications at the 95% confidence level, and his net increase in local heat flux due to axial fuel column shortening is calculated to be 0.87%.

An engineering uncertainty factor of 1.05 is specified for type D fuel. This includes the previously approved factors of 1.03 engineering uncertainty and 1.0175 for fuel column shortening.

6.4 Specifications have been added for F_r^A and F_r^T (defined in 6.1 above) with full power values of 1.45 and 1.77 respectively. These are acceptable because they were assumed values in the analysis supporting the LHGR limit discussed in 6.3.1 above. These carry a part power fraction, P, dependence factor of 1.0+0.5(1-P) as proposed by the licensee in supplement 3 to his proposed Technical Specification change request.(11) This P dependence factor was justified in Reference 7 in connection with the review of 2530 MWt operation of the Palisades reactor.

6.5 Changes have been made to the bases of Section 3.10 which are consistent with the licensee's analyses,(1) and are therefore acceptable.

6.6 A new Section 3.11.g has been added to provide for monitoring of the radial peaking factors F_r^A and F_r^T discussed in Section 6.4 above. This specification calls for determining that these factors are within their limit whenever the core power distribution is evaluated. This is done weekly, which we find also an acceptable frequency for F_r^A and F_r^T monitoring.

6.7 Figures 3-7 and 3.8 have been deleted because they no longer required in view of the Technical Specification changes discussed herein. Figure 3.9 has been replaced with the correct function supported by the change discussed in Section 6.3.1 above.

7.0 Environmental Consideration

We have determined that the amendment does not involve a change in effluent types or total amounts nor an increase in power level 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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

8.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: September 8, 1978

References

1. Letter, David P. Hoffman (CPC) to Director, Nuclear Reactor Regulation (NRC), dated June 15, 1978, enclosing "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MWt," XN-NF-78-16, June 1, 1978.

2. Letter, David P. Hoffman (CPC) to Director, Nuclear Reactor Regulation (NRC), dated August 15, 1978.

3. Letter, David P. Hoffman (CPC) to Director, Nuclear Reactor Regulation (NRC), dated August 21, 1978.

4. J. S. Holm and F. B. Skogen, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors," XN-76-40, September, 1976.

5. J. S. Holm and F. B. Skogen, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors -- Phase 2," January, 1978.

6. K. P. Galbraith and T. W. Patten, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation," XN-75-21, April 1, 1975.

7. G. E. Koester et. al., "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt," XN-NF-77-18, July 18, 1977.

8. W. C. Kayser, "LOCA Analysis for Palisades at 2530 MWt Using the ENC WREM-II PWR ECCS Evaluation Model," CENPD-137P, August, 1974.

9. "Calculative Methods for the C-E Small Break LOCA Evaluation Model," CENPD-137P, August, 1974.

10. "Palisades Thermal Hydraulic Design Report for Cycle 2 Core," XN-76-3, March, 1976.

11. Letter, David P. Hoffman (CPC) to Director, Nuclear Reactor Regulation (NRC), dated August 25, 1978.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 43 to Provisional Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised the Technical Specifications for operation of the Palisades Plant (the facility) located in Covert Township, Van Buren County, Michigan. The amendment is effective as of its date of issuance.

The amendment changes the Palisades Technical Specifications relating to the limits on axial power distribution.

The application for the amendment complies 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. Prior public notice of this action was not required since the amendment does not involve a significant hazards consideration.

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 impact statement or negative declaration and 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 application for amendment dated June 15, 1978, and supplements thereto dated August 15, 21 and 25, 1978, (2) Amendment No. 43 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, N. W., Washington, D. C. 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 8th day of September, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Docket No.(s) 50-255 Date: 6/26/78

William O. Miller
License Fee Management Branch
Office of Administration

FACILITY AMENDMENT CLASSIFICATION - Palisades

Applicant: Consumers Power Co.

License No.(s) DPR-20 Mail Control No: 781730031

Application Dated: 6/15/78 Fee Remitted; ✓ Yes ___ No

Applicant's Fee Classification: Class ___ I, ___ II, ✓ III, ___ IV, ___ V, ___ VI, ___ None

Amendment No: 43 Date of Issuance 9/8/78

✓ 1. This application has been reviewed by DOR/DPM in accordance with Section 170.22 of Part 170 and is properly categorized.

☐ 2. This application is incorrectly classified and should be properly categorized as Class ____. Justification for reclassification: _____

☐ 3. Additional information is required to properly categorize the license amendment: _____

☐ 4. The application was filed (a) ___ by a nonprofit educational institution, (b) ___ by a Government agency, (c) ___ pursuant to written NRC recommendations and the amendment will be issued for the convenience of the Commission, and (d) ___ Other (state reason therefor): _____

Richard D. Silver for D. F. Ziemann
Division of Operating Reactors/Project Management

☐ 5. This application has been reviewed and is exempt from fees.

6/26/
H Smith
ORPM
R. Silver
Date

Appl. attached

William O. Miller, Chief
License Fee Management Branch