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. 57
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated February 26, 1980, 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.

8007100299

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. 57 , 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

G. P. Lien for

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 6, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 57

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.

PAGES

i

1-2

1-3*

1-4*

3-58

3-59

3-61

3-63

3-66

(add new page)...................... 3-87a

*There are no changes to the provisions contained on these pages; the Technical Specifications have merely been repositioned.

PALISADES PLANT
Technical Specifications

TABLE OF CONTENTS

1.1 REACTOR OPERATING CONDITIONS (Contd)

Low Power Physics Testing

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

Interior Fuel Rod

Any fuel rod of an assembly that is not on that assembly's periphery.

Total Interior Rod Radial Peaking Factor - $F_r^{\Delta H}$

The maximum product of the ratio of individual assembly power to core average assembly power times the highest interior local peaking factor integrated over the total core height including tilt.

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.

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 intitiating 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 DEFINTIONS

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.

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.

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 $F_A(Z)$, and the linear heat generation rate in any interior fuel rod at the peak power elevation shall not exceed 14.33 kW/Ft x $F_B(Z)$ where the function $F_A(Z)$ is shown in Figure 3.9 and $F_B(Z)$ is shown in Figure 3.10. If the power distribution is double peaked, both peaks shall

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

3.10.3 Power Distribution Limits (Contd)

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) An engineering uncertainty factor (which includes fuel column shortening due to densification and thermal expansion) of 1.03, and

(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 $\leq 1.45 (1.0 + 0.5 (1 - P))$, the calculated value of F_r^T shall be limited to $\leq 1.77 (1.0 + 0.5 (1 - P))$, and the calculated value of $F_r^{\Delta H}$ shall be limited to $\leq 1.66 (1.0 + 0.5 (1 - P))$, where P is the core thermal power in fraction of core rated thermal power (2530 MW_t).

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

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, 3.10.1.b, 3.10.2.b, 3.10.3.f, 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.

3.10.8 Center Control Rod Mialignment

The requirements of Specifications 3.10.4.1, 3.10.4.a, 3.10.5 and S-5.1 may be suspended during the performance of physics tests to determine the isothermal temperature coefficient and power coefficient provided that only the center control rod is misaligned and the limits of Specification 3.10.3 are maintained.

Basis

Sufficient control rods shall be withdrawn at all times to assure that the reactivity decrease from a reactor trip provides adequate shutdown margin. The available worth of withdrawn rods must include the reactivity defect of power and the failure of the withdrawn rod of highest worth to insert. The requirement for a shutdown margin of 2.0% in reactivity with 4-pump operation, and of 3.75% in reactivity with less than 4-pump operation, is consistent with the assumptions used in the analysis of accident conditions (including steam line break) as reported in XN-NF-77-18 and additional analysis.(5) The change in insertion limit with reactor power shown on Figure 3-6 insures that the shutdown margin requirements for 4-pump operation is met at all power levels. The 2.5-second drop time specified for the control rods is the drop time used in the transient analysis.(5)

The maximum individual rod worth of inserted control rods and associated peaking factors have been used to demonstrate reactor safety for the unlikely event of a rod ejection accident as described in Reference 5. The maximum worth of an inserted control rod will not exceed the values of the specification for the regulating group insertion limits of Figure 3-6.

The limitation on linear heat generation rate ensures that, in the event of a LOCA, the Nuclear Regulatory Commission criteria set forth in 10 CFR 50.45(b) will be met.(6) In addition, the limitation on linear heat rate and interior fuel rod linear heat rate ensures that the minimum DNBR will be maintained above 1.30 during anticipated transients, and that fuel damage (if any) during Condition IV events such as locked

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

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 values 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, F_r^T and $F_r^{\Delta H}$ 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.

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, F_r^T or $F_r^{\Delta H}$ 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 $[(1.77 \div F_r^T) \times 2530\ MW_t]$, $[(1.45 \div F_r^A) \times 2530\ MW_t]$ or $[(1.66 \div F_r^{\Delta H}) \times 2530\ MW_t]$, 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. A four level, five level or six level system may be used.(1)(2) 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.

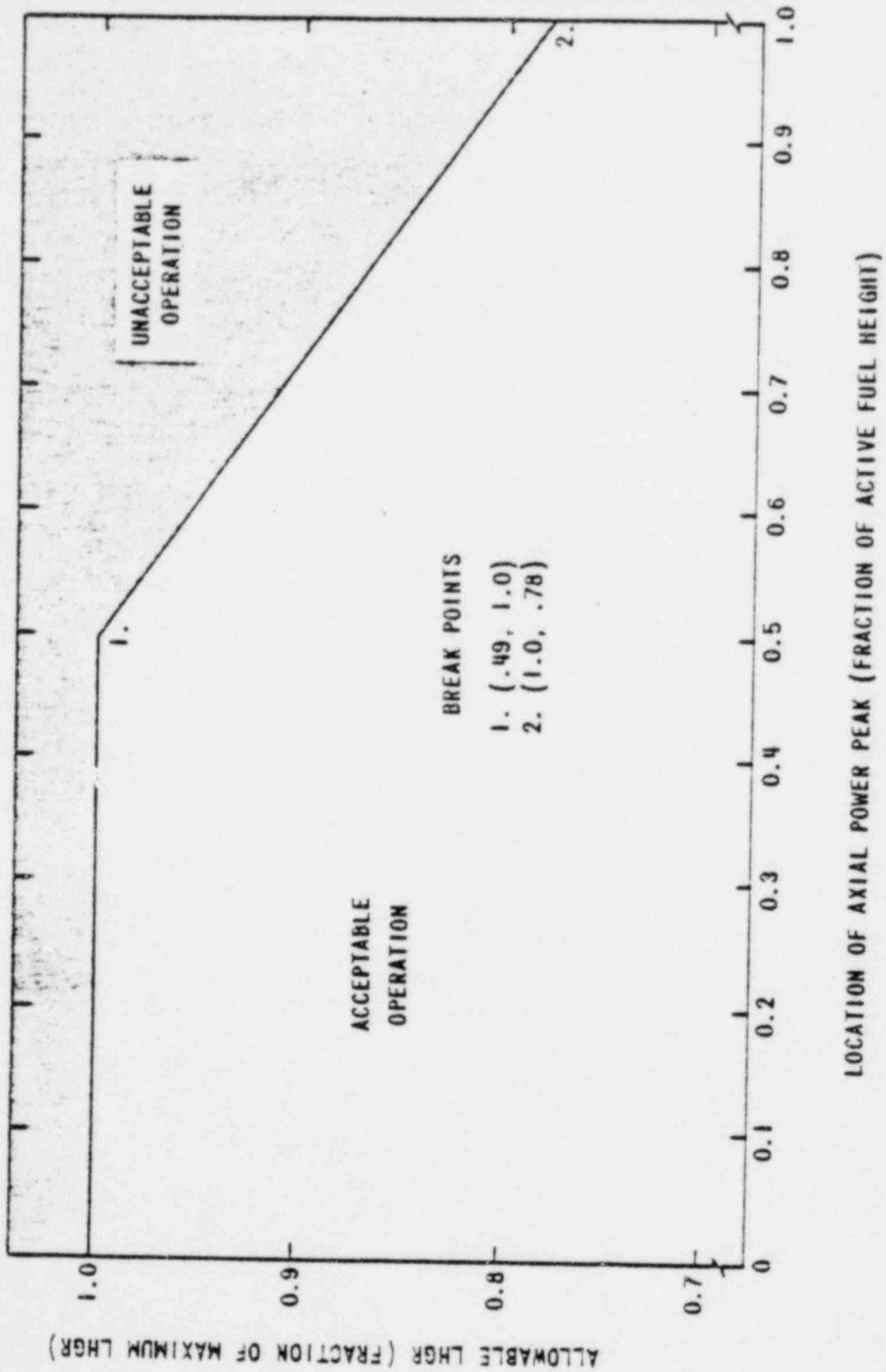
Core power distribution verification includes:

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



ALLOWABLE LHGR AS A FUNCTION OF PEAK POWER LOCATION	Palisades Technical Specifications	Figure 3-10