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June 15, 1978

Director, Nuclear Reactor Regulation
Att: Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DRP-20 -
PALISADES PLANT - PROPOSED TECHNICAL
SPECIFICATION CHANGES RELATED
TO POWER DISTRIBUTION LIMITS

Transmitted herewith are three (3) original and thirty-seven (37) conformed copies of a proposed change to the Technical Specifications for the Palisades Plant, Docket 50-255, License DPR-20.

The purpose of this change is to modify allowable power distribution limits as determined by Exxon Nuclear Company, Inc and reported in XN-NF-78-16 entitled, "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MW_t" dated June 1, 1978. Forty (40) copies of this report were transmitted directly from Exxon to the NRC by a letter dated June 15, 1978.

Consumers Power requests that the proposed changes be reviewed as rapidly as possible so that current operating restrictions related to power distribution limits may be eliminated by September 1, 1978. To maintain the current Technical Specification power distribution limits the plant must operate with group 4 rods partially inserted at power levels above 85% full power.

Pursuant to 10 CFR 170.22 the requested changes involve a single issue. As such, it is concluded that the request should be classified as Class III. A check for the appropriate amount is attached.

David P. Hoffman

David P Hoffman
Assistant Nuclear Licensing Administrator

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CC: JGKeppler, USNRC

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CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

For the reasons hereinafter set forth, it is requested that the Technical Specifications contained in Provisional Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on October 16, 1972 for the Palisades Plant be changed as described in Section I below:

I. Change

A. Add new definitions for Assembly Radial Peaking Factor -F_r^A and Total Radial Peaking Factor -F_r^T to Section 1.1 as indicated on attached Technical Specification page changes.

B. Delete references to Hot Channel Factors in Section 2.1 as indicated on attached Technical Specification page changes.

C. Change Section 3.10.3.a as indicated on attached Technical Specification page changes.

D. Add new Section 3.10.3.g as indicated on attached Technical Specification page changes.

E. Change Basis for Section 3.10 as indicated on attached Technical Specification page changes.

F. Add new Section 3.11.g as indicated on attached Technical Specification page changes.

G. Delete Figures 3-7 and 3-8 and Replace Figure 3-9 with new 3-9 included in attached Technical Specification page changes.

II. Discussion

The primary purpose of these Technical Specification changes are to modify the axial limitations on linear heat rate. The previous limit was based on a single analyzed axial shape, while the new limit is based on a sensitivity study involving both LOCA and DNBR analyses performed by Exxon Nuclear (see XN-NF-78-16, "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MW_t").

A. The definitions added to Section 1.1 to be consistent with assumptions used in the Exxon sensitivity study (XN-NF-78-16).

B. The references in Section 2.1 have been deleted and the appropriate information included in the new version of Section 3.10.3.

C. The changes to Section 3.10.3.a incorporate the results of the axial power distribution sensitivity study as presented in XN-NF-78-16. There are different limits for ENC type and D type fuel because of the different number of fuel rods and the different engineering factors assumed in the analysis. The maximum LHGR for D fuel is the same as the current Technical Specification.

D. The new Section 3.10.3.g provides limits for core radial peaking factors consistent with those used in the LOCA and DNB analysis in XN-NF-78-16.

E. Changes to the Basis for Section 3.10 have been made to be consistent with the Exxon analysis (XN-NF-78-16).

F. The new Section 3.11.g has been added to when radial peaking factors are to be determined and what action is necessary if the limits of 3.10.3.g are exceeded.

G. With the new Exxon analysis (XN-NF-78-16) and the inclusion in 3.10.3.a of various uncertainty factors Figures 3-7 and 3-8 are no longer required. The new Figure 3-9 is identical to Figure 2-1 of the Exxon report XN-NF-78-16.

III. Conclusion(s)

Based on the foregoing, both the Palisades Plant Review Committee and the Safety and Audit Review Board have reviewed these changes and recommend their approval.

CONSUMERS POWER COMPANY

By C R Bilby

C R Bilby, Vice President
Production & Transmission

Sworn and subscribed to before me this 15 day of June 1978.

Linda R. Thayer

Linda R Thayer, Notary Public
Jackson County, Michigan
May commission expires July 9, 1979.

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."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ 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.

2.1 SAFETY LIMITS - REACTOR CORE (Contd)

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 $\leq 1.45 \div P$ and the calculated value of F_r^T shall be limited to $\leq 1.77 \div P$, where P is the core thermal power in fraction of core rated thermal power (2,530 Mw_t).

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

Basis (Contd)

rotor will not exceed acceptable limits.[5] 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.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

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.

3.11 IN-CORE INSTRUMENTATION (Contd)

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.

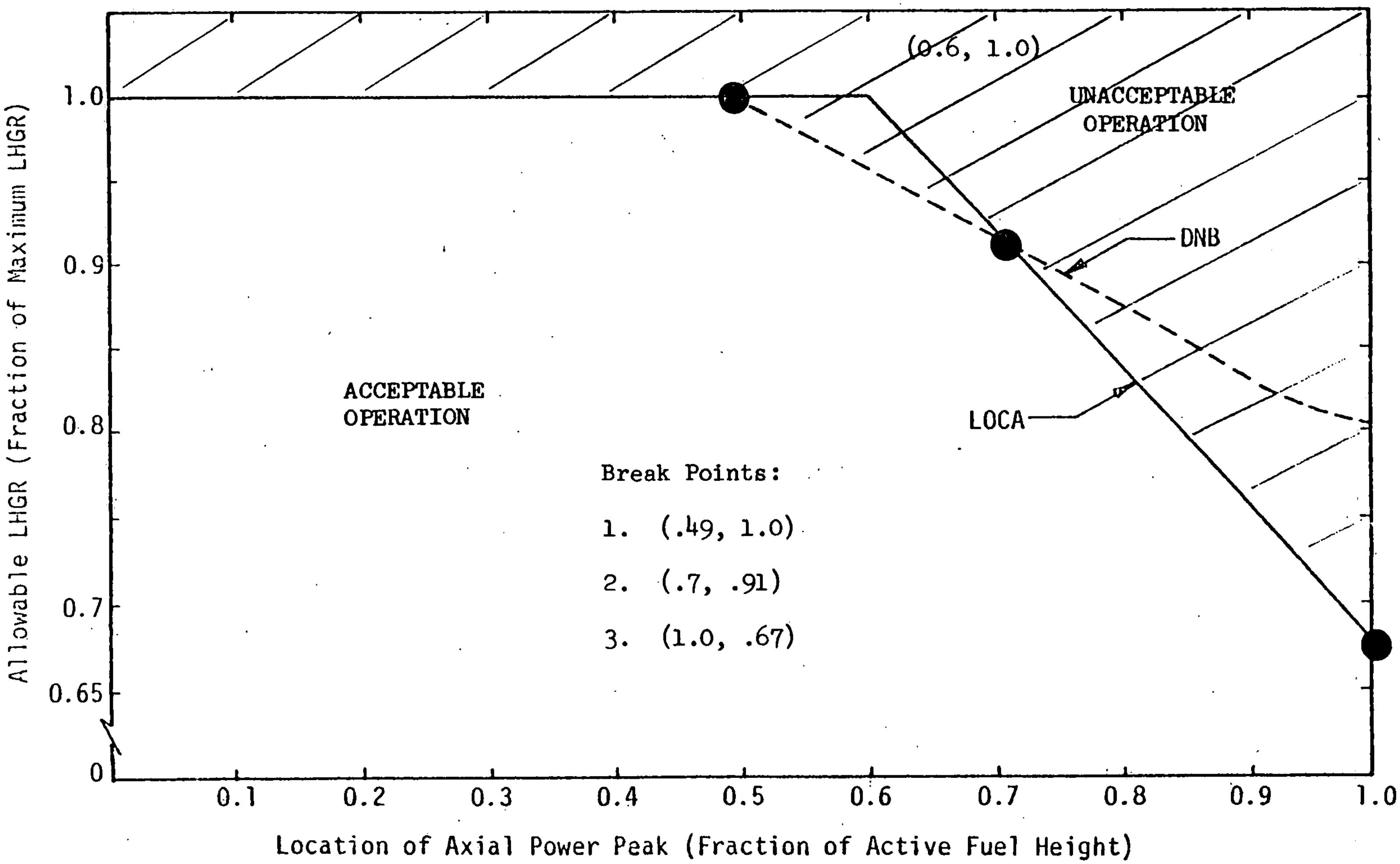
If the data logger for the in-core readout is not in operation for more than two hours, power will be reduced to provide margin between the actual peak linear heat generation rates and the limit and the in-core readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the

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.



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