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Docket No. 50-255

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 31 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 request dated August 12, 1977, as supplemented September 26, 1977.

This amendment authorizes operation of the Palisades Plant at power levels up to 2530 megawatts thermal.

Copies of the Safety Evaluation, Environmental Impact Appraisal and the Notice of Issuance/Negative Declaration are also enclosed.

Sincerely,

/S/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

JMcGough
DRoss
DEisenhut
CMiles
BHarless
JSaltzman
TJCarter
VStello

Enclosures:
1. Amendment No. 31 to DPR-20
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

cc w/encl: See next page

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

November 1, 1977

Docket No. 50-255

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

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A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 31 to DPR-20
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

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. 31
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 August 12, 1977, as supplemented September 26, 1977, 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 2.C.(2) of Provisional License No. DPR-20 is hereby amended to read as follows:

"(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31, 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

R. W. Reid for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 1, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 31

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A as follows:

1. Remove the following pages and insert identically numbered revised pages.

i	3-58
ii	3-59
iii	3-60
1-1	3-61
1-2	3-62
2-2	3-63
2-5	3-64
2-6	3-65
2-7	3-66
2-8	3-71
2-9	3-72
2-10	3-73
2-11	3-75
2-12	3-84
2-13	3-85
3-1	4-10
3-1a	4-11
3-3	4-12
3-3a	4-15b
3-25	4-29
3-29	4-35

2. Delete pages 2-10a and 3-66a

3. Add page 3-87a

Marginal lines indicate changed area.

PALISADES PLANT
TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (Contd)

TABLE OF CONTENTS (Contd)

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of these Technical Specifications:

1.1 REACTOR OPERATING CONDITIONS

Rated Power

A steady state reactor core output of 2530 MW_t.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmitic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition

The reactor is considered to be in a hot standby condition if the average temperature of the primary coolant (T_{avg}) is greater than 525°F and any of the control rods are withdrawn and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition

When the reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is greater than 525°F.

Refueling Shutdown Condition

When the primary coolant is at refueling boron concentration and T_{avg} is less than 210°F.

Cold Shutdown Condition

When the primary coolant is at shutdown boron concentration and T_{avg} is less than 210°F.

Refueling Operation

Any operation involving movement of core components when the vessel head is unbolted or removed.

Shutdown Margin

Shutdown margin shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming that all full-length control rods are fully inserted except for the single highest worth control rod which is assumed to be withdrawn.

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 therefor in these Technical Specifications.

Shutdown Boron Concentration

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.

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.

2.1 SAFETY LIMITS - REACTOR CORE (Contd)

probability at a 95% confidence level that 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.

The curves are based on the following nuclear hot channel factors:

3- and 2-Pump Operation: $F_q^N = 3.62$ and $F_{\Delta H}^N = 1.94$

4-Pump Operation: $F_q^N = 2.48$ and $F_{ROD}^N = 1.77$*

These limiting hot channel factors are higher than those calculated at rated power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. (Control rod insertion limits are covered in Specification 3.10.) Somewhat worse hot channel factors could occur at lower power levels because additional control rods may be in the core; however, the control rod insertion limits dictated by Figure 3-6 insure that the minimum DNBR is always greater at part-power than at rated power.

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]

*F_{ROD}^N = Peak Rod Power/Average Rod Power

References

(1) FSAR, Section 3.3.3.5.

(2) FSAR, Section 3.3.3.3, Appendix C.

(3) FSAR, Section 14.1.

TABLE -.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	≤ 39% of Rated Power(4) (Continuous Operation Not Permitted)	≤ 21% of Rated Power(4) (Continuous Operation Not Permitted)
2. Low Primary Coolant Flow(2)	≥ 95% of Primary Coolant Flow With Four Pumps Operating	≥ 71% of Primary Coolant Flow With Four Pumps Operating	≥ 46% of Primary Coolant Flow With Four Pumps Operating
3. High Pressurizer Pressure	≤ 2255 Psia	≤ 2255 Psia	≤ 2255 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 react 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

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

1. High Power Level - A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from some reactivity excursions too rapid to be detected by pressure and temperature measurements.

   During normal plant operation with all primary coolant pumps operating, reactor trip is initiated when the reactor power level reaches 106.5% of indicated rated power. Adding to this the possible variation in trip point due to calibration and instrument errors, the maximum actual steady state power at which a trip would be actuated is 112%, which was used for the purpose of safety analysis.[1]

   Provisions have been made to select different high power level trip points for various combinations of primary coolant pump operation as described below under "Low Primary Coolant Flow."[3]

   If reactor operation at less than 10% of rated power is required for an extended period of time, provisions have been made to allow the operator to decrease the indicated power range by a factor of 10, which will also decrease the high power level trip point by a factor of 10 to 10.65% of indicated rated power.[2] Administrative procedures will allow this range change to be made during reactor start-up and also between 5% and 8% of rated power when the reactor power is reduced to that level.

2. Low Primary Coolant Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly. Provisions are made in the reactor protective system to permit operation of the reactor at reduced power if one or two coolant pumps are taken out of service. These low-flow and high-flux settings have been derived in consideration of instrument errors and response times of equipment involved to assure that MDNBR $\geq$ 1.30 and flow stability will be maintained during normal operation[5, 13] and anticipated transients.[4] For reactor operation with one or two coolant pumps inoperative, the low-flow trip points and the overpower trip points must be manually changed to the specified values for the selected pump condition by means of a set point

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

Basis (Contd)

selector switch. Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves. The percent of normal core flow is shown in the following table:(5)

4 Pumps	100.0%
3 Pumps	74.7%
2 Pumps	48.7%

During four-pump operation, the low-flow trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors.(3) The high power level trip and the low primary coolant flow trip are reduced to compensate for the corresponding core flow reduction experienced with fewer than four pumps in operation. The trip points are shown in Table 2.3.1.

3. High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the primary and secondary safety valves to prevent primary system overpressure (Specification 3.1.7). In the event of loss of load without reactor trip, the temperature and pressure of the primary coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is at least 100 psi below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.(11)

4. Thermal Margin/Low-Pressure Trip - A reactor trip is provided to prevent operation with an MDNBR of less than 1.30 or under conditions of parallel channel flow instability which may lead to premature DNB. The thermal and hydraulic safety limits shown on Figures 2-1, 2-2, and 2-3 for two, three and four primary coolant pump operation, respectively, define the limiting values of primary coolant pressure, reactor inlet temperature, and core power level for which the criteria on MDNBR and parallel channel flow stability are met. For each mode of operation, a thermal margin/low-pressure

Basis (Contd)

(TM/LP) trip will occur before these limits are reached. Reference 13 forms the basis for Figure 2-3 for 4-pump operation. For 2- and 3-pump operation the flow instability criterion is more limiting than the MDNBR criterion. Reference 7 forms the basis for Figures 2-1 and 2-2.

The trip is initiated whenever the pressurizer pressure drops below the the minimum value given on Table 2.3.1, or a value computed as described below, whichever is higher. The computed value is a function of reactor inlet temperature and reactor outlet temperature, and takes the form $P_{Trip} = AT_H - BT_C - C$ where A, B, and C are constants and T_H and T_C are the hot and cold leg coolant temperatures, respectively. The minimum value of reactor coolant flow and the maximum expected values of axial and radial peaking factors are assumed in generating this trip function.

The TM/LP trip set points are derived from the 4-pump operation core thermal limits (Figure 2-3) through application of appropriate allowances for measurement uncertainties and processing errors. A maximum error of 165 psi is assumed to account for expected instrument drift and repeatability errors, process measurement uncertainties, flow stratification effects, and calibration errors. As such, a maximum error in the calcaulted set point of -165 psi has been assumed in the accident analysis.[12]

For two and three coolant pump operation, power is limited to 21% and 39% of rated power, respectively, for a maximum of 12 hours. During either of these modes of operation, the high power level trip in conjunction with the TM/LP trip (minimum set point = 1750 psia) and the secondary system safety values (set at 1000 psia) assure that the limits shown on Figures 2-1 and 2-2 will not be violated.

5. Low Steam Generation Water Level - The low steam generation water level reactor trip protects against the loss of feed-water flow accidents and assures that the design pressure of the primary coolant system will not be exceeded. The specified set point assures that there will be sufficient water inventory in the steam generator at the time of trip to provide a 15-minute margin before the auxiliary feedwater is required.[9]

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

Basis (Contd)

The setting listed in Table 2.3.1 assures that the heat transfer surface (tubes) is covered with water when the reactor is critical.

6. Low Steam Generator Pressure - A reactor trip on low steam generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the primary coolant. The setting of 500 psia is sufficiently below the rated load operating point of 739 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used in the accident analysis.(8)

7. Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down upon the initiation of the safety injection system. The setting of this trip is identical to that of the containment high-pressure safety injection signal.(10)

8. Low Power Physics Testing - For low power physics tests, certain tests will require the reactor to be critical at low temperature ($\geq$ 260°F) and low pressure ($\geq$ 415 psia). For these certain tests only, the thermal margin/low pressure, and low steam generator pressure trips may be bypassed in order that reactor power can be increased for improved data acquisition. Special operating precautions will be in effect during these tests in accordance with approved written testing procedures. At reactor power levels below 10^{-1}% of rated power, the thermal margin/low-pressure trip is not required to prevent fuel rod thermal limits from being exceeded. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown, should a steam line break occur during these tests.

References

(1) FSAR, Section 4.1.

(2) FSAR, Section 7.2.3.2.

(3) FSAR, Section 7.2.3.3.

(4) XN-NF-77-18, Section 3.3

(5) FSAR, Section 3.3.3.

(6) Deleted.

(7) FSAR, Section 3.3.6.

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

References (Contd)

(8) XN-NF-77-18, Section 3.8.

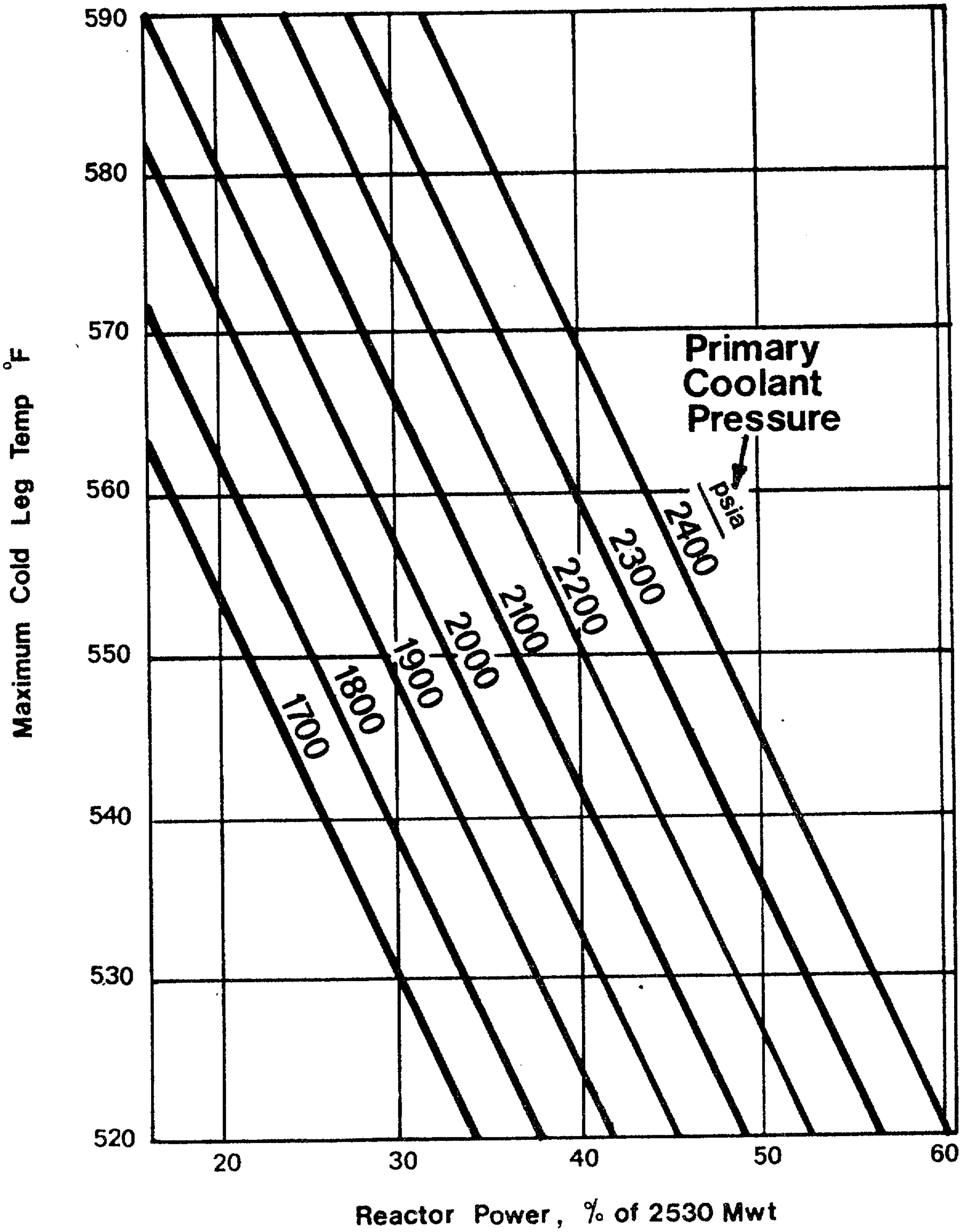
(9) XN-NF-77-18, Section 3.7.

(10) FSAR, Amendment No 17, Item 4.0.

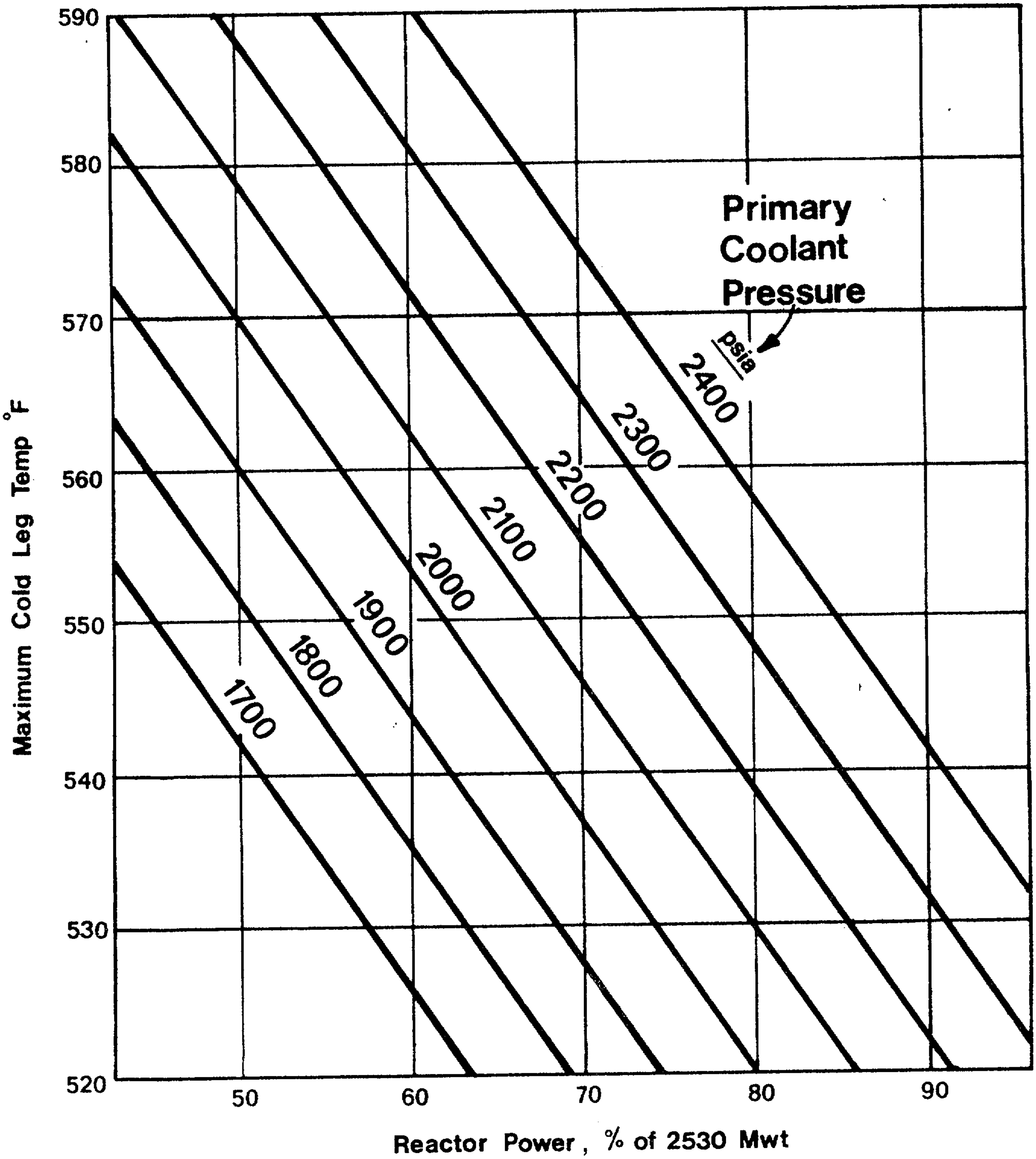
(11) XN-NF-77-18, Section 3.6.

(12) XN-NF-77-18, Section 3.1.

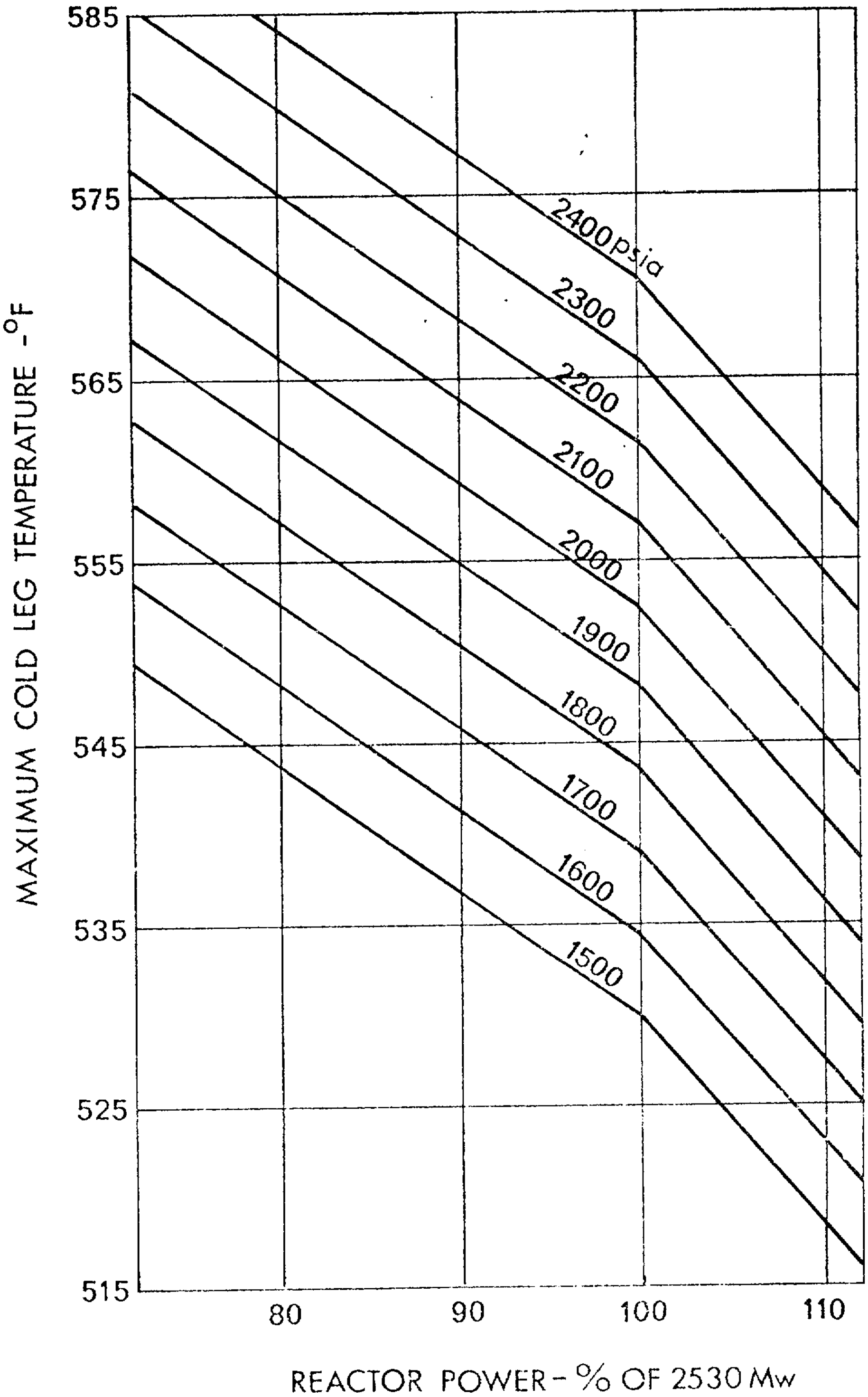
(13) XN-NF-77-22, Section 3.4.



Reactor Core Safety Limits 2 Pump Operation	Palisades Technical Specifications	Figure 2-1



Reactor Core Safety Limits 3 Pump Operation	Palisades Technical Specifications	Figure 2-2



Reactor Core Safety Limits 4 Pump Operation	Palisades Technical Specifications	Figure 2-3

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 vessel flow (as determined by reactor coolant pump differential pressures and pump performance curves) shall be 126.9×10^6 lb/h or greater, when corrected to 532°F.

In the event the measured flow is less than that required above, the limits specified on Figure 2-3 shall be reduced by 1°F in inlet temperature 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°F.

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

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

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.1 Operable Components (Contd)

(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 ten 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 not exceed 2100 psia.

g. The reactor inlet temperature (indicated) shall not exceed the value given by the following equation at steady state 100% power operation:

$$T_{inlet} \leq 538.0 + 0.03938\,(P-2060) + 0.00004843\,(P-2060)^2 + 1.0342\,(W-120.2)$$

Where: T_{inlet} = reactor inlet temperature in °F.

P = nominal operating pressure in psia.

W = total recirculating mass flow in 10^6 lb/h corrected to the operating temperature conditions.

Note: This equation is shown in Figure 3-0 for a variety of mass flow rates. *

3.1 PRIMARY COOLANT SYSTEM (Contd)

The maximum transient steam generator differential pressure is expected to occur during the loss of load accident. The loss of load accident initiated from hot full power operating conditions and assuming a high pressurizer trip of 2277 psia is analyzed in Reference 3. Results of this analysis indicate that the maximum steam generator differential pressure is less than 1530 psi for the worst case assuming pressurizer spray and relief valves inoperable and assuming steam dump and turbine bypass operable. The 1530 psi limit on transient 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 transient analyses were performed assuming a vessel flow at hot zero power (532°F) of 126.9×10^6 lb/h minus 6% to account for flow measurement uncertainty and core flow bypass.(3) A steady state DNB analysis was also performed (assuming 115% overpower, 50 psi for pressure uncertainty, 3% for flow measurement uncertainty, and 3% for core flow bypass) in a parametric fashion to determine the core inlet temperature as a function of pressure and flow for which the minimum DNBR at 115% overpower is equal to 1.30.(4) The result of this steady state DNB analysis was the following equation for limiting reactor inlet temperature:

$$T_{inlet} \leq 541.0 + 0.03938\,(P-2060) + 0.00004843\,(P-2060)^2 + 1.0342\,(W-120.2)$$

A temperature measurement uncertainty of 3°F was subtracted from this limit in arriving at the LCO given in Section 3.1.1.g. The nominal full power inlet temperature is 2°F less than the value given in Section 3.1.1.g to allow for drift within the temperature control band. Thus, a total uncertainty of 5°F is applied to the limiting reactor inlet temperature equation. The limits of validity of this equation are:

$$1850 \leq \text{Pressure} \leq 2250 \text{ Psia}$$

$$110.0 \times 10^6 \leq \text{Vessel Flow} \leq 130 \times 10^6 \text{ Lb/h}$$

References

(1) FSAR, Sections 6.1.2.2 and 14.3.2.

(2) FSAR, SEction 4.3.7.

(3) XN-NF-77-18.

(4) XN-NF-77-22.

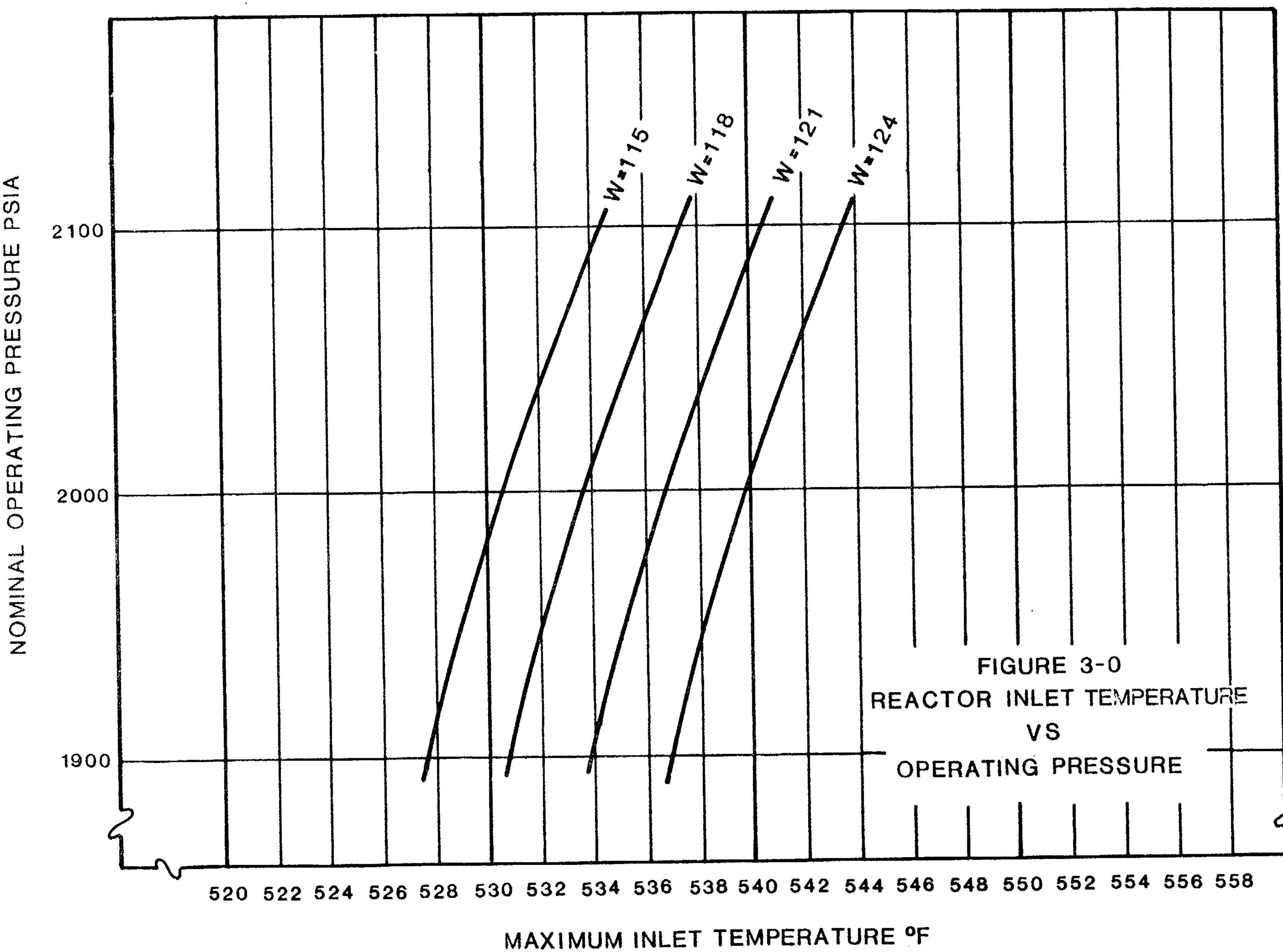


FIGURE 3-0
REACTOR INLET TEMPERATURE
VS
OPERATING PRESSURE

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.7 Primary and Secondary Safety Valves

Specifications

a. The reactor shall not be made critical unless all three pressurizer safety valves are operable with their lift settings maintained between 2500 psia and 2580 psia (± 1%).

b. A minimum of one operable safety valve shall be installed on the pressurizer whenever the reactor head is on the vessel.

c. Whenever the reactor is in power operation, a minimum of 23 secondary system safety valves shall be operable with their lift settings between 985 psig (± 10 psig) and 1025 (± 1%) psig.

Basis

The primary and secondary safety valves pass sufficient steam to limit the primary system pressure to 110 percent of design (2750 psia) following a complete loss of turbine generator load without simultaneous reactor trip while operating at 2650 MW_t.[1]

The reactor is assumed to trip on a "High Primary Coolant System Pressure" signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the secondary system safety valves. Conservative values for all system parameters, delay times and core moderator coefficient are assumed. Overpressure protection is provided to the portions of the primary coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of one valve's capacity. One valve, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

The total relief capacity of the 24 secondary system safety valves is 11.7 $\times 10^6$ lb/h. This is based on a steam flow equivalent to an NSSS power level of 2650 MW_t at the nominal 1000 psia valve lift pressure.

At the power rating of 2530 MW_t, a relief capacity of less than 11.2 $\times 10^6$ lb/h is required to prevent overpressurization of the secondary system of loss of load conditions, and 23 valves provide relieving capability of 11.2 $\times 10^6$ lb/h.[1, 2]

The ASME Boiler and Pressure Vessel Code, Section III, 1971 edition, Paragraph NC-7614.2(a) allows the specified tolerances in the lift pressures of safety valves.

References

(1) FSAR, Sections 4.3.4 and 4.3.7.

(2) XN-NF-77-18, Section 3.6.

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 but not more than 2000 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 but not more than 2000 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.

3.10 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 peak linear heat generation rate with appropriate consideration of normal flux peaking, measurement-calculational uncertainty, engineering factor, increase in linear heat rate due to fuel densification, power measurement uncertainty, and flux peaking augmentation shall not exceed the following value at any core elevations, Z:

$$14.12^{*}\ F_A(Z)$$

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

3.10.3 Power Distribution Limits (Contd)

The axial power distribution term F_A is a function of elevation in the core and is shown graphically in Figure 3-9.

The measurement-calculational uncertainty shall be 10%, the engineering factor shall be 3%, the increase in linear heat generation 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 for pressurized high density CE fuel shall be 1.0.

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).

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.

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 two 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. The 4-pump operation limits of Figure 3-6 do not apply for decreasing power level rapidly when such a decrease is needed to avoid or minimize a situation harmful to the plant personnel or equipment. Once such a power decrease is achieved, the limits of Figure 3-6 will be returned to by borating the control rods above the insertion limit within two hours. Limits more restrictive than Figure 3-6 may be implemented during fuel cycle life based on physics calculations and physics data obtained during plant start-up and subsequent operation. New limits shall be submitted to the NRC within 45 days.

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 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

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.

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 two 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. The limits of Figure 3-6 do not apply for decreasing power level rapidly when such a decrease is needed to avoid or minimize a situation harmful to the plant personnel or equipment. Once such a power decrease is achieved, the limits of Figure 3-6 will be returned to by borating the control rods above the insertion limit. 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 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.

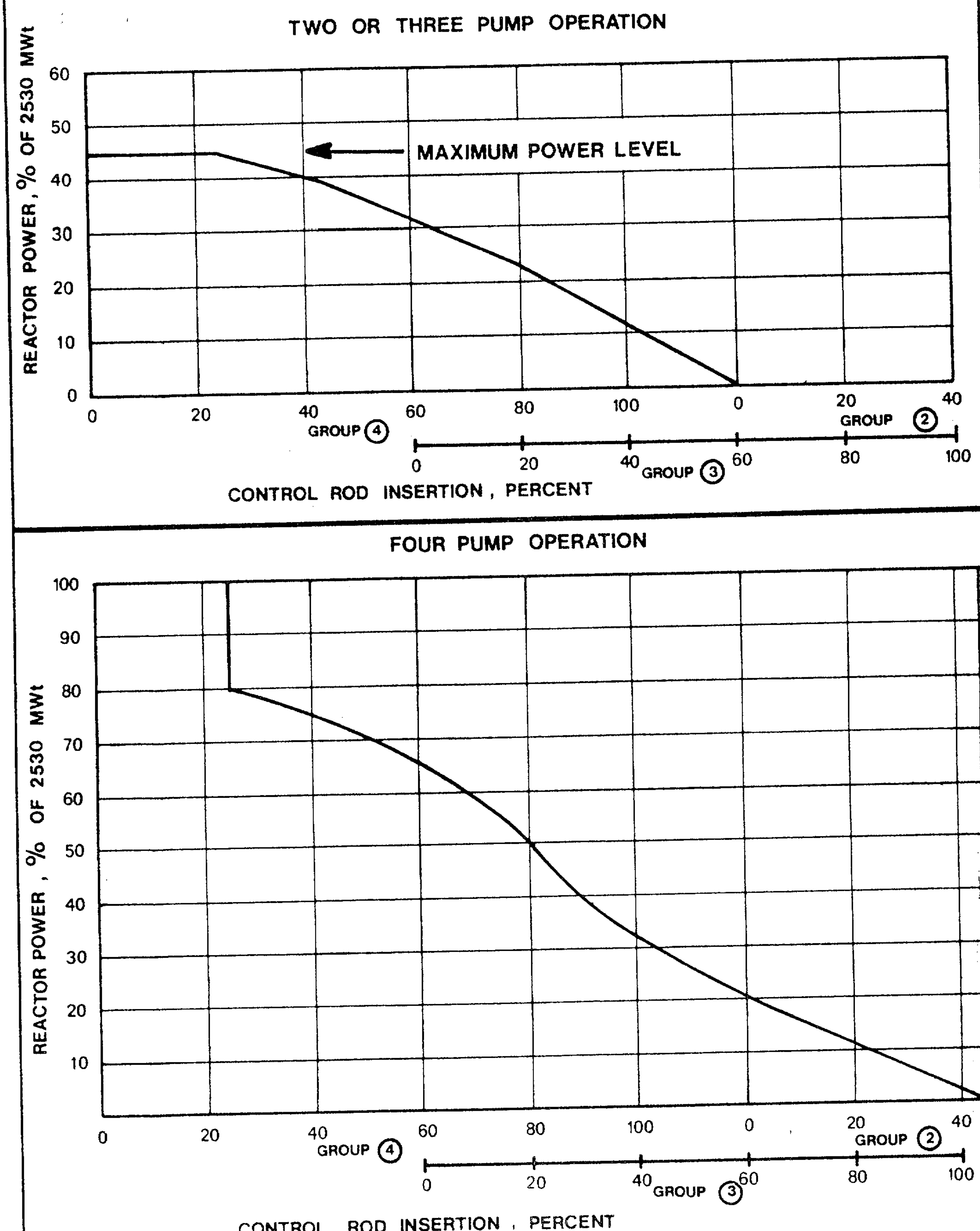
Basis

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 requirement 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 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



CONTROL ROD INSERTION LIMITS	PALISADES TECHNICAL SPECIFICATION	FIGURE 3-6

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

3.10.7 Low Power Physics Testing (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 distribution. 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.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

3.10.7 Low Power Physics Testing (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

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 in-core detectors and not less than 10 individual detectors per quadrant, which shall include two 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 3.10.3.a. 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 3.10.3.a.

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 data logger for the in-cores is not 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

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.

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 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.

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS

Applicability

This specification applies to the engineered safety features system initiation instrumentation settings.

Objective

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded.

Specifications

The engineered safety features system initiation instrumentation setting limits and permissible bypasses shall be as stated in Table 3.16.1.

Basis

a. High Containment Pressure - The basis for the 5 psig $\binom{+0.75}{-0.25}$ set point for the high-pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.[1, 2]

b. Pressurizer Low Pressure - The pressurizer low-pressure safety injection signal is a diverse signal to the high containment pressure safety injection signal. The settings include an uncertainty of -22 psia and are the settings used in the Loss of Coolant Accident analysis.[3]

c. Containment High Radiation - Four area monitors in the containment initiate an isolation signal under high radiation condition. The setting is based on the following analysis:

A 10 gpm primary coolant leak to the containment atmosphere is used based upon Specification 3.1.5. Primary coolant radioactivity concentration was assumed to be the maximum allowable by Specification 3.1.4.

Note: Added to this is the contribution from N^{16} whose equilibrium radioactivity in the primary coolant is estimated to be 121 μCi/cc. Semi-infinite cloud geometry and uniform mixing of radioactivity in the containment atmosphere was assumed. N^{16} equilibrium exists in containment atmosphere due to its short half-life, but all other radioactivity was assumed

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS (Contd)

Basis

to build up indefinitely. Calculations show that at the end of a 24-hour leakage period, the dose rate is approximately 20 R/h as seen by the area monitors. A large leak could exceed the 20 R/h setting rapidly and initiate isolation.(4)

d. Low Steam Generator Pressure - A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the primary coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a -22 psi uncertainty and was the setting used in the FSAR Section 14 analysis.(5)

e. SIRW Tank Low-Level Switches - Level switches are provided on the SIRW tank to actuate the valves in the injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 20 minutes following a safety injection signal.(5) The switchover point of 27 inches ($^{+0}_{-6}$) above tank bottom is set to prevent the pumps from running dry during the 60 seconds required to stroke the valves and to hold in reserve approximately 20,000 gallons of 1720 ppm borated water. No specific setting was used for the accident analyses stated in the FSAR Section 14.

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS (Contd)

f. Engineered Safeguards Pump Room Vent-Radiation Monitor - A process monitor is installed to provide an isolation signal upon high radioactivity levels in the engineered safeguards pump rooms. The setting is based on the following analyses:

To maintain acceptable dose levels at the site boundary, it is necessary not only to detect significant quantities of leakage into the east and west engineered safeguards pump rooms, but also to provide a reasonable trip point at which ventilation of the rooms will terminate. For this, the following analysis was performed: Primary coolant radioactivity concentration was assumed to be the maximum allowable by Specification 3.1.4. No fuel melting is assumed to occur. An average beta energy was calculated for each nuclide for purposes of converting individual isotopic radioactivity concentrations seen by the process monitor to count rates measured by the monitor. The design exhaust ventilation rate of 2400 cfm was assumed along with a 1 gpm leak rate into the room. This leakage is made up of primary coolant (81,800 gallons) diluted with up to 285,000 gallons of SIRW tank water and 7,480 gallons of safety injection tank water. This results in a total primary coolant radioactivity minus noble gases residing in 374,280 gallons of water. The results show that the process monitor registered a count rate of about 2.2×10^5 cpm. Normal background for this monitor is expected to be less than 1×10^3 cpm. Furthermore, due to the wide variation between normally expected background count rate and the count rate registered during a 1 gpm leak rate, detection of far smaller leak rates can be expected. This is especially true in the event of fuel meltdown. The relative safety implications of a 1 gpm leak rate into the engineered safeguards pump room are minor (per basis of Specification 3.1.5).[4]

TABLE 3.16.1

Engineered Safety Features System Initiation Instrument Setting Limits

Functional Unit	Channel	Setting Limit
1. High Containment Pressure	a. Safety Injection b. Containment Spray c. Containment Isolation d. Containment Air Cooler DBA Mode	5 - 5.75 Psig
2. Pressurizer Low Pressure	Safety Injection	$\geq$ 1550 Psia[1] for Nominal Operating Pressures < 1900 Psia $\geq$ 1593 Psia[2] for Nominal Operating Pressures $\geq$ 1900 Psia
3. Containment High Radiation	Containment Isolation	$\leq$ 20 R/h
4. Low Steam Generator Pressure	Steam Line Isolation	$\geq$ 500 Psia[3]
5. SIRW Low-Level Switches	Recirculation Actuation	$\leq$ 27-Inch $\left(\begin{smallmatrix}+0\\-6\end{smallmatrix}\right.$ Above Tank Bottom
6. Engineered Safeguards Pump Room Vent - Radiation Monitors	Engineered Safeguards Pump Room Isolation	$\leq 2.2 \times 10^5$ CPM

[1] May be bypassed below 1600 psia and is automatically reinstated above 1600 psia.

[2] May be bypassed below 1700 psia and is automatically reinstated above 1700 psia.

[3] May be bypassed below 550 psia and is automatically reinstated above 550 psia.

3.19 IODINE REMOVAL SYSTEM

Applicability

Applies to the operational status of the iodine removal system.

Objective

To define those conditions when it is necessary to have the Iodine Removal System operable.

Specification

3.19.1 During power operation the Iodine Removal System shall be operable with:

a. The Iodine Removal Hydrazine Tank, T-102, containing 350 gallons of 5.5 ± 0.5% w/o of hydrazine solution.

b. The Iodine Removal Make-up Sodium Hydroxide Tank, T-103, containing 5,100 gallons of 23.0 ± 0.5% w/o sodium hydroxide solution.

c. T-102 capable of supplying hydrazine solution to the water from the SIRW tank, T-58, and T-103 capable of supplying sodium hydroxide solution to the suction header between the containment sump and the spray and injection pumps.

d. With the Iodine Removal System inoperable, restore the system to operable status within 72 hours or be in hot shutdown condition, within the next 48 hours until operable status is achieved.

Bases

The Iodine Removal System acts in conjunction with the containment spray system to reduce the post-accident level of fission products in the containment atmosphere. Hydrazine is added to the water from the SIRW tank after a LOCA to provide for iodine retention. Sodium Hydroxide is added to the recirculated water after a LOCA to establish a neutral pH.

References

FSAR, Section 6.4

FSAR, Section 14.22

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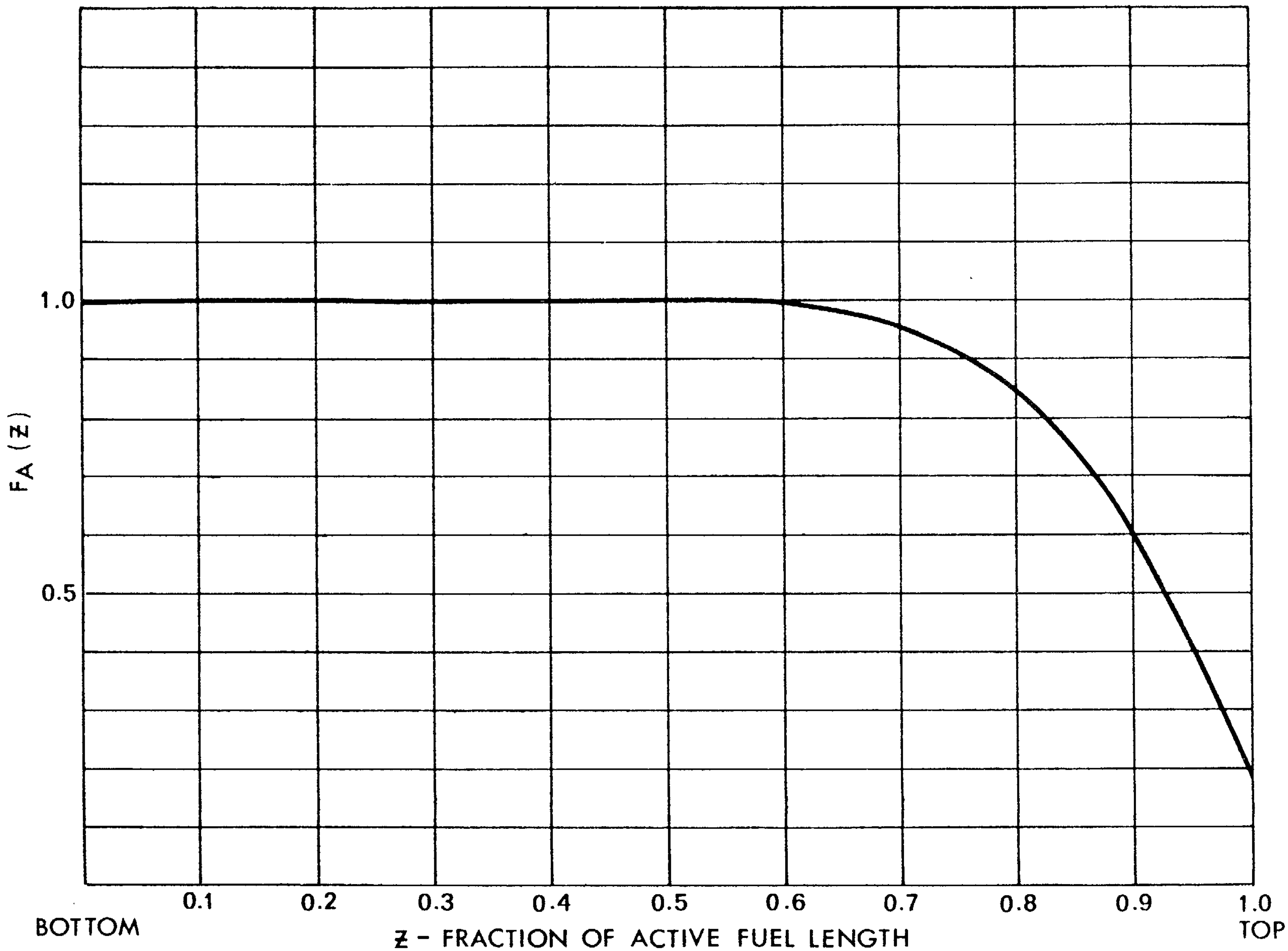

Axial Correction Factor For Peak Linear Heat Generation Rate	Palisades Technical Specifications	Figure 3-9

TABLE 4.1.3

Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Start-Up Range Neutron Monitors	a. Check	S	a. Comparison of both channel count rate indications when in service.
	b. Test	P	b. Internal test signals.
2. Primary Rod Position Indication System	a. Check	S	a. Comparison of output data with secondary RPIS.
	b. Check	M	b. Check of power dependent insertion limits monitoring system.
	c. Calibrate	R	c. Physically measured rod drive position used to verify system accuracy. Check rod position interlocks.
3. Secondary Rod Position Indication System	a. Check	S	a. Comparison of output data with primary RPIS.
	b. Check	M	b. Same as 2(b) above.
	c. Calibrate	R	c. Same as 2(c) above, including out-of-sequence alarm function.
4. Area and Process Monitors	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	M	c. Detector exposed to remote operated radiation check source.
5. Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
	b. Test	M	b. Battery check.
6. Environmental Monitors	a. Check	M	a. Operational check.
	b. Calibrate	A	b. Verify airflow indicator.
7. Pressurizer Level Instruments	a. Check	S	a. Comparison of six independent level readings.
	b. Calibrate	R	b. Known differential pressure applied to sensor.
	c. Test	M	c. Signal to meter relay adjusted with test device.

TABLE 4.1.3

Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
8. Control Rod Drive System Interlocks	a. Test	R	a. Verify proper operation of all rod drive control system interlocks, using simulated signals where necessary.
	b. Test	P	b. Same as 8(a) above, if not done within three months.
9. Flux-ΔT Power Comparator	a. Calibrate	R	a. Use simulated signals.
	b. Test	M	b. Internal test signal.
10. Calorimetric Instrumentation	a. Calibrate	R	a. Known differential pressure applied to feed-water flow sensors.
11. Containment Building Humidity Detectors	a. Test	R	a. Expose sensor to high humidity atmosphere.
12. Interlocks - Isolation Valves on Shutdown Cooling Line	a. Calibrate	R	a. Known pressure applied to sensor.
13. Service Water Break Detector in Containment	a. Test	R	a. Known differential pressure applied to sensors.
14. Control Room Ventilation	a. Test	R	a. Check damper operation for DBA mode with HS-1801 and isolation signal.
	b. Test	R	b. Check control room for positive pressure.

TABLE 4.1.3

Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls (Contd)

FREQUENCY NOTATION

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

**NOTE: This interval is included as on interval not included in the standard Technical Specifications but required by the present commitments.

Table 4.2.2 (continued)

Minimum Frequencies for Equipment Tests

13. Iodine Removal System

The Iodine Removal System shall be demonstrated operable:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

    1. Verifying that tanks T-102 and T-103 contain the minimum required volumes.

    2. Verifying the concentration of hydrazine in T-102 and sodium hydroxide in T-103.

c. At least once per 18 months, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position.

(3) Visual inspection shall be made for excessive leakage from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

b. Acceptance Criterion

The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

c. Corrective Action

Repairs shall be made as required to maintain leakage within the acceptance criterion of 4.5.3.b.

d. Test Frequency

Tests of the recirculation heat removal system shall be conducted at intervals not to exceed twelve months.

4.5.4 Surveillance for Prestressing System

a. Tendon inspection shall be accomplished in accordance with the following schedule:

1. One year after initial structural integrity test.
2. Three years after initial structural integrity test.
3. Five years after initial structural integrity test.
4. At five-year intervals thereafter for the life of the plant.

b. Surveillance tendons for the one-year inspection shall be the nine designated surveillance tendons plus V-104 and V-200. In addition, 15 vertical tendons shall be tested for lift-off forces only.

c. For the three-year inspection, the surveillance tendons shall consist of the 11 tendons inspected during the one-year test plus an additional 10 vertical tendons to be tested for lift-off force only. The additional 10 tendons shall be selected from tendons other than those tendons tested for lift-off force during the one-year inspection.

4.5 CONTAINMENT TESTS (Contd)

an important part of the structural integrity of the containment is maintained.

The basis for specification of a total leakage rate of $0.60L_a$ from penetrations and isolation valves is specified to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. This value allows for possible deterioration in the intervals between tests. The limiting leakage rates from the shutdown cooling system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a DBA. The test pressure (270 psig) achieved either by normal system operation or by hydrostatically testing gives an adequate margin over the highest pressure within the system after a DBA. Similarly, the hydrostatic test pressure for the return lines from the containment to the shutdown cooling system (100 psig) gives an adequate margin over the highest pressure within the lines after a DBA.[5]

A shutdown cooling system leakage of 1/5 gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the DBA. The engineered safeguards room ventilation system is equipped with isolation valves which close upon a high radiation signal from a local radiation detector. These monitors shall be set at 2.2×10^5 cpm, which is well below the expected level, following a loss-of-coolant accident (LOCA), even without clad failure. The 1/5 -gpm leak rate is sufficiently high to permit prompt detection and to allow for reasonable leakage through the pump seals and valve packings, and yet small enough to be readily handled by the sumps and radioactive waste system. Leakage to the engineered safeguards room sumps will be returned to the containment clean water receiver following an LOCA, via the equipment drain tank and pumps. Additional makeup

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

Introduction

By letters dated August 12 and September 26, 1977, Consumers Power Company (the licensee) requested changes to the Technical Specifications appended to Provisional Operating License No. DPR-20 for operation of the Palisades Plant in Van Buren County, Michigan. The proposed changes would permit operation of the Palisades Plant at an increased power level of 2530 Megawatts Thermal (MW_t). This represents a 15% increase over the presently authorized power level of 2200 MW_t.

Discussion

By letter dated January 22, 1974, the licensee requested authorization to increase the steady state power level of the Palisades Plant to 2638 MW_t. On that same date, the licensee also requested that the Palisades Provisional Operating License (POL) be converted to a Full Term Operating License (FTOL). The staff had originally issued the Palisades POL on March 24, 1971, for operation at one MW_t with POL amendments through October 16, 1972 increasing this authorization up to present steady state power level of 2200 MW_t. Our original thermal-hydraulics review of the Palisades reactor was based on operation at 2200 MW_t. The licensee indicated in the Palisades FSAR that the ultimate design power level was 2650 MW_t (subsequently reduced to 2638 MW_t). Before allowing operation at a power level greater than 2200 MW_t we stated that we would perform an additional safety evaluation to ensure that the Palisades reactor can be operated at the higher power level.

In June 1972, we issued a Final Environmental Statement (FES) which appraised the environmental impact of operations at 2638 MW_t for all postulated high level radioactive release accidents and for all non-radiological, environmental aspects of normal operations. For low level radiological releases at normal operations, the environmental impact was assessed at the then requested power level of 2200 MW_t.

With regard to the requested conversion of the Palisades POL to an FTOL, we are presently involved in the evaluation of the scope of review necessary to support such an action. Pending completion of our review of this licensing action, the licensee has elected to request an interim increase to a power level less than the 2638 MWt requested in its letter of January 22, 1974, and in support of this application has provided information regarding the neutronics, thermal hydraulics, transient and accident analyses, as well as proposed physics tests and Technical Specification changes for operation at 2530 MWt. (Reference 3 thru 11).

Evaluation

1.0 MECHANICAL DESIGN

The mechanical design of the cycle 2 fuel was reviewed and approved by the staff in our SER and license amendment dated April 29, 1977.

The time to clad collapse has been calculated by the licensee using the COLAPX [12] code. The results of the calculations show that clad collapse is not predicted for an assumed end of cycle 2 burnup of 15,000 MWD/MTU. The core is actually calculated to be capable of operation at 2530 MWt to a lesser core exposure of 13,140 MWD/MTU, thereby providing considerable margin to these calculations. Since the COLAPX code has been previously approved by the staff, we find these predictions to be acceptable and conclude that cladding collapse will not occur for the remainder of cycle 2 operation.

2.0 NUCLEAR DESIGN

Exxon Nuclear Company, a fuel supplier, has reevaluated the neutronics characteristics of the cycle 2 core for operation at the requested power level. It was assumed that at an exposure of approximately 10,500 MWD/MTU the core power level would be increased to 2530 MWt and maintained at that level until the end of cycle 2. Present exposure is approximately 11,500 MWD/MTU.

The largest power defect (sum of moderator temperature and Doppler defects) is calculated to be 1.84% $\Delta\rho$ at 15,000 MWD/MTU. A conservative reactivity allowance of 2.0% $\Delta\rho$ is provided for this by the available shutdown margin in the control rods at end of cycle condition.

The allowable nuclear peaking factors have been reduced for the increased power condition. The allowable radial peaking factor has been changed from 1.6 to 1.45 while the allowable axial peaking factor has decreased from 1.5 to 1.4.

The licensee is running the PDQ7 Code in order to generate the library to be used in INCA. INCA is Palisades computer program which calculates actual power distributions from in core measurements. The INCA library contains exposure dependent data for calculating coupling coefficients, millivolt-to-power conversion factors and one pin peaking factors. The staff has reviewed several comparisons of the licensee's PDQ calculations with reactor measurements of radial power distribution.[11] The agreement between the calculations and measurements is good. The largest errors tend to be in the low power assemblies along the core periphery. The peak assembly power has been predicted within 4% of the INCA value in the worst case. These errors are compensated for by the 10% physics uncertainty allowance incorporated in the operating limits in the Technical Specifications and are therefore acceptable for the remainder of cycle 2 operation.

The licensee will perform physics tests during the power increase to verify the predicted values of the neutronics parameters. This test program is discussed further in a later section.

3.0 THERMAL AND HYDRAULIC DESIGN

The licensee's thermal-hydraulic analysis for the higher power level shows that the minimum departure from nucleate boiling ratio (MDNBR) is never less than the allowed 1.30 for normal operation and anticipated transients. Extra margin is provided by the fact that the steady state DNB calculations were performed at 115% rated power (2910 MWt). In addition, the transient analyses were performed from an initial power of 102% power (2580.6 MWt). The results of the transient analyses are discussed in the following section of this report.

4.0 TRANSIENT AND ACCIDENT ANALYSES

A majority of the postulated transients and accidents analyzed in the Palisades FSAR were reanalyzed by the licensee for operation at a power level of 2530 MWt. Table 4.0-1 lists those incidents which were reanalyzed. The licensee has demonstrated, to the satisfaction of the staff, using standard ENC calculational methods, that the consequences of these transients and accidents are acceptable. Table 4.0-2 summarize the results of the analyses. Table 4.0-3 lists those incidents which were not reanalyzed and provides the reasons for not requiring a reanalysis.

The Loss-of-Coolant Accident is discussed separately in a later section. A brief discussion of the potentially more severe accidents follows.

TABLE 4.0-1

TRANSIENTS AND ACCIDENTS REANALYZED

Uncontrolled Rod Withdrawal

Control Rod Drop

Loss of Coolant Flow

Locked Rotor

Excessive Feedwater

Excessive Load

Loss of Load

Loss of Feedwater

Steam Line Break

Control Rod Ejection

Loss of Coolant Accident

TABLE 4.0-2

SUMMARY OF RESULTS

Transient	Maximum Power Level (Mwt)	Maximum Core Average Heat Flux (Btu/hr·ft^2)	Maximum Pressurizer Pressure (psia)	Maximum Primary-Secondary ΔP (psid)	MDNBR¶
Initial Conditions* For Transients	2580.6	169,600	2010	1238	1.75
Uncontrolled Rod Withdrawal					
Rod Withdrawal @ 1.4×10^{-4} $\Delta\rho$/sec from 102% Power	2838	183,050	2103	1290	1.52
Rod Withdrawal @ 1.0×10^{-5} $\Delta\rho$/sec from 102% Power	2833	182,970	2161	1319	1.45
Rod Withdrawal @ 6.0×10^{-4} $\Delta\rho$/sec from 52% Power	3188	143,910	2113	1169	2.00
Rod Withdrawal @ 6.0×10^{-5} $\Delta\rho$/sec from 52% Power	1942	124,110	2133	1221	1.89
Control Rod Drop	2196	165,240++	**	1238	1.35
Loss of Coolant Flow					
Four Pump Coastdown	2629	169,600	2073	1240	1.39
Locked Rotor	2650	169,600	2080	1250	1.27
Excessive Feedwater Flow Incidents					
Reduction in Feedwater Enthalpy	2590	169,910	2019	1272	1.75
Increased Feedwater Flow from 52% Power	1484	97,500	2036	1240	3.00

TABLE 4.0-2 (Continued)

Transient	Maximum Power Level (Mwt)	Maximum Core Average Heat Flux (Btu/hr·ft^2)	Maximum Pressurizer Pressure (psia)	Maximum Primary-Secondary ΔP (psid)	MDNBR¶
Excessive Load					
From 102% Power	2870	178,780	**	1287	1.74
From Hot Standby	258	17,075	**	1363	3.60†
Loss of Load	2838	176,415	2394	1388	1.39
Loss of Feedwater	2673	172,905	2162	1238	1.65
Steam Line Break					
From 102% Power	464	30,960††	**	***	1.30†
From Hot Standby	694	45,530	**	***	1.41†
Uncontrolled Withdrawal of an Individual Control Rod	2841	182,515	2125	1297	1.44
Control Rod Ejection Incident	399,740	¶¶	2260	†††	†††

* Initial conditions are for 102% of rated power (including measured error and control board allowances).
** Pressure decreases from initial value (2060 ± 50 psia).
*** The criteria on primary secondary ΔP is not applicable for steam line breaks.
† Calculated using the modified Barnett CHF correlation.
†† Maximum heat flux after return to power.
††† Not applicable for control rod ejection incident.
¶ Does not include rod bow penalty.
¶¶ Average enthalpy of hottest fuel pellet < 247 cal/gm.

TABLE 4.0-3

TRANSIENTS AND ACCIDENTS NOT REANALYZED

INCIDENT	REASON NOT REANALYZED
Boron Dilution	At startup or refueling the FSAR analysis is still bounding. At power, the incident is bounded by the Rod Withdrawal incident.
Steam Generator Tube Rupture	The FSAR analysis, done at 2650 MWt, is bounding
Turbine Generator Overspeed	The FSAR analysis is still valid since it is not affected by the power increase.
Fuel Handling Accident	A bounding analysis was performed in connection with the spent fuel pool storage expansion approved by us in a license amendment issued on June 30, 1977
Idle Loop Startup	Startup of the reactor is not permitted with less than 4 pumps in operation.
Malpositioning of Part-Length Control Rod Group	Operation of the reactor is permitted only with the part-length control rods completely withdrawn from the core.

4.1 Control Rod Drop Indicent

This analysis was performed for the maximum (-0.12 $\Delta\rho$) and minimum (-0.04 $\Delta\rho$) expected dropped rod worths at both beginning and end of cycle conditions. The lowest MDNBR of 1.35 occurred for the maximum rod worth at BOC conditions. This is above the 1.30 DNBR limit. The staff finds this acceptable.

4.2 Loss of Coolant Flow

A four pump coastdown was analyzed assuming loss of power to all four reactor coolant pumps from operation at 102% power without turbine generator assistance. Beginning of cycle kinetic coefficients were used and a 0.8 multiplier was applied to the BOC Doppler coefficient for conservation. A MDNBR of 1.39 and a peak system pressure of 2073 psia were reached. This pressure is well below the Technical Specification limit of 2750 psia. The staff concludes thatthis is acceptable.

4.3 Locked Rotor

This accident assumes the instantaneous seizure of a reactor coolant pump shaft due to mechanical failure. Beginning of cycle, full power conditions are used. A MDNBR of 1.27 and a maximum system pressure of 2080 psia were predicted. A MDNBR ratio less than 1.30 is acceptable for this accident because of its low probability of occurrence. The maximum predicated pressure of 2080 psia is less than the Technical Specification limit of 2750 psia and we therefore conclude that the results of the analysis of the locked rotor accident are acceptable.

4.4 Steam Line Break

This accident was reanalyzed for two loop (four pump) operation at both full power and zero power conditions. Both analyses assumed a shutdown margin of -2.0% $\Delta\rho$. For the 102% power case a MDNBR of 1.30 and peak power of 464 MWt are predicted. For the zero power case a MDNBR of 1.41 and peak power of 693 MWt are predicted.

The steam break accident was also analyzed for less than four operating primary coolant pumps (PCPs) since limited operation (12 hours) is authorized at partial power following the trip of one or two PCPs. The most severe case was determined to be for 1 loop operation at zero power conditions. The two active pumps were assumed to be located in the ruptured loop. The analysis was performed to determine the required shutdown margin to just prevent return to power even though some return to power would be allowed for this accident. It was shown that a shutdown margin of 3.75% $\Delta\rho$ would be required. The Technical Specifications have been modified to reflect this requirement (see Section 7.0)

The staff concludes that these analyses are acceptable and that the results show that the core will be protected during operation at the higher power level.

4.5 Control Rod Ejection

This accident was reanalyzed for both the hot zero power and hot full power conditions, with and without heat transfer from the fuel to the moderator. The most limiting hot zero power case was at beginning of cycle with no heat transfer. A fuel pellet enthalpy of 247 cal/gm was predicted. For hot full power, the end of cycle, no heat transfer case resulted in the most limiting fuel pellet enthalpy of 200 cal/gm. The results of these analyses are less severe than those previously analyzed for cycle 1 operation and continue to meet the acceptance criteria of less than 280 cal/gm and are therefore acceptable.

5.0 ECCS PERFORMANCE EVALUATION

The licensee submitted an ECCS performance analysis, for cycle 2 operation at 2530 MWt, which used the ENC WREM-II PWR Evaluation Model. This model has been reviewed and accepted by the staff.

The cycle 2 increased power analysis included a spectrum of seven large pipe breaks performed for the ENC type E fuel operating at a peak linear heat generation rate (PLHGR) of 14.68 kw/ft. Additional conservatism has been provided by this analysis, since the core will be limited to operation at a PLHGR of 14.12 kw/ft. A spectrum of three large pipe breaks also was performed for the CE type D fuel since it was shown to be the more limiting fuel type by previous analyses. The D fuel was analyzed at a PLHGR of 14.12 kw/ft. The staff concludes that the break spectrums included in the analysis are acceptable.

The most limiting pipe break was determined to be a double-ended guillotine located in the pump discharge side of the cold leg, with a discharge coefficient of 0.6. The results of the calculations are summarized in Table 5.0 - 1.

The licensee also submitted a sensitivity study of changes in peak clad temperature (PCT) with fuel burnup. It was shown that PCT decreases with fuel burnup. The results listed in Table 5.0 - 1 are for beginning of life conditions.

TABLE 5.0 - 1

RESULTS OF ECCS CALCULATIONS FOR CYCLE 2 AT 2530 MW_t

Fuel Type	PLHGR	Peak Clad Temperature	Local Clad Oxidation	Hydrogen Generation
E	14.68 kw/ft	2179°F	<12%	<<1.0%
D	14.12 kw/ft	2152°F	<13%	<<1.0%

As indicated in Table 5.0 - 1, the predicted values of peak clad temperature, local clad oxidation, and hydrogen generation are below their respective limits of 2200°F, 17 percent and 1 percent as specified in 10 CFR 50.46 (b).

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 additional plugging should it become necessary. If any future plugging results in an additional 500 plugged tubes, the LOCA analysis would be resubmitted for our approval prior to resumption of operation.

Based on our review, we conclude that the Palisades Plant can operate at an increased power of 2530 MWt for the remainder of cycle 2 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 limit of 14.12 kw/ft is not exceeded.

6.0 PHYSICS TESTS

The licensee has described the power increase test program to be conducted at Palisades. (References 9 and 11)

The power distribution and moderator temperature coefficient will be measured. The power level then will be increased slowly (5% per 8 hour shift) from the present licensed level of 2200 MWt to approximately 95% of 2530 MWt. At that point the power distribution, moderator temperature coefficient and power coefficient will be measured. The power level will then be increased to 2530 MWt at a rate of 5% per 8 hours. A final power map will be taken.

Test results and comparison with predictions and acceptance limits will be reported to NRC within 90 days of completion of the above tests.

The staff concludes that the licensee's plan for confirmatory testing and documentation is acceptable.

7.0 TECHNICAL SPECIFICATIONS

The results of the steady state and transient safety analyses, performed for the increased power level of 2530 MWt, have been used to define Limiting Conditions of Operation (LCO) and Limiting Safety System Setpoints (LSSS). The licensee proposed the following modifications to the Technical Specifications for operation at 2530 MWt:

Sec. 1.1 (p1-1) Rated Power is changed to 2530 MWt

Sec. 2.1 (p2-2) 4-Pump Operation, hot channel factors have been modified for 2530 MWt operation.

Sec 2.3 (Table 2.3-1) Reactor Protective System Trip Setting Limits have been modified for 2530 MWt operation.

Sec. 2.1 (Figs. 2-1,-2,-3) Reactor Core Safety Limits for 2, 3 and 4 Pump Operation have been modified for 2530 MWt operation.

Sec. 3.1.1 (p3-1) The minimum 4 pump reactor vessel flow has been modified for 2530 MWt operation. Also a formula to adjust the thermal margin trip limits to accommodate flow reductions due to possible future steam generator tube plugging has been provided.

Sec. 3.1.1 (p3-1a, Fig. 3-0) Reactor Inlet Pressure VS Operating Pressure for a given mass flow rate has been modified for 2530 MWt operation.

Sec. 3.1.7 (p3-25) The minimum number of operable secondary system safety valves has been increased for 2530 MWt operation.

Sec. 3.10.1 (p3-58) The Shutdown Margin Requirements have been modified to be consistent with the steam line break analysis for 2530 MWt operation.

Sec. 3.10.2 (p3-58) The Individual Rod Worth limits have been modified to be consistent with the rod ejection incident for 2530 MWt operation.

Sec. 3.10.3 (p3-58, 59, Fig 3-9) The Power Distribution Limits have been modified for 2530 MWt operation. PLHGR is consistent with the LOCA analysis and an axial power distribution term has been provided which replaces the upper to lower half of the core power ratio.

Sec. 3.10.5 (Fig 3-6) Control Rod Insertion Limits have been modified for 2530 MWt operation.

Sec. 3.16,4.1 (p3-72, Table 3.16.1, Table 4.1.3) The turbine runback feature has been eliminated for a dropped control rod as a result of the analysis of this incident for 2530 MWt operation. It was necessary to remove the turbine runback feature to assure an MDNBR of greater than 1.30 for a dropped rod at BOC condition.

Other minor modifications have been proposed, such as, updated references and editorial changes.

The staff has reviewed the proposed modifications to the Technical Specifications and concludes that they are acceptable.

8.0 STEAM GENERATOR CONSIDERATIONS

On March 11, 1977, we issued a license amendment to the Palisades POL which permitted an increase in the primary system pressure from 1800 psia to 2100 psia. The basis for our evaluation was an analysis which considered a primary to secondary differential pressure of 1380 psia and the effects of this differential pressure on steam generator tube integrity under normal operating and accident conditions.

The request for a power increase to 2530 MWt does not require an associated increase in primary coolant pressure in that the high power can be safely attained with the currently authorized system pressure of 2100 psia and the reduced allowable peaking factors.

The licensee has performed tests of flow-induced vibrational loads on the steam generator tubes under ultimate design conditions. The results of these tests have been submitted. (Reference 13). We have reviewed the results of these tests and conclude that operation at the proposed increased power level of 2530 MWt does not introduce any unacceptable consequences with regard to steam generator integrity.

9.0 RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

During the Operating License review, the staff analyzed the offsite radiological consequences of accidents at the Palisades plant for a core power level of 2650 MWt. The results were reported in the staff's original SER, dated March 6, 1970, and in the SER Supplement 3, dated June 11, 1971. The staff found that the doses would be less than current dose guidelines for all accidents then postulated, except the LOCA. The staff estimated that the LOCA offsite doses would be less than 10 CFR Part 100 guidelines for core power levels only up to 2200 MWt. The staff required that the licensee have a containment iodine removal system installed at the Palisades plant prior to operation at core power levels greater than 2200 MWt. In connection with his stretch power application, the licensee recently notified us that they would have such a system operable. To confirm the staff's previous analysis of the iodine removal system at Palisades, we independently reanalyzed the LOCA offsite doses and the final design of the containment spray system. Our analysis and conclusions are summarized later in this section.

During the OL review neither the licensee nor the staff determined the offsite doses from a postulated control rod ejection accident. For the current application for 2530 MWt operation, we estimated the rod ejection accident consequences at a core power level of 2650 MWt and found the offsite doses to be well within current dose guidelines. The assumptions made and results obtained in our analysis of this accident are given in Tables 9.0-1 and 9.0-2.

Also during our review of the current licensing action we found that control room habitability under accident conditions was not analyzed in the Palisades FSAR or reviewed by the staff. We will resolve this matter (compliance with the control room operator radiation exposure guidelines of General Design Criterion 19 of 10 CFR Part 50, Appendix A and Standard Review Plan 6.4) with the licensee prior to Cycle 3. This delayed resolution is acceptable because the proposed power increase would increase the potential operator doses only by a small amount (about 15%). We believe that the probability of occurrence of an accident of sufficient magnitude to endanger the control room operators (if they are not already adequately protected) during the period required to resolve this matter with the licensee is very low. For these reasons we will permit continued operation of the Palisades plant and operation at power levels up to 2530 MWt while we resolve the matter of control room habitability with the licensee.

9.1 Loss of Coolant Accident Analysis (LOCA)

We reanalyzed the LOCA offsite doses using information in the Palisades FSAR and Technical Specifications and in the staff's SERs Several key assumptions made in the staff's earlier SERs required modification. For example, the earlier analyses used X/Q values based on Pasquill "F" atmospheric stability and a 2 meter per second wind speed. Detailed meteorological data recorded at the D. C. Cook plant site (30 miles away, also on the shore of Lake Michigan) indicated a more conservative wind speed should be assumed for analysis of accidents at Palisades than was previously assumed. The X/Q values used in our current analysis are significantly higher than those used previously. In addition, our analysis of the final design of the containment spray system resulted in somewhat lower assumed iodine removal effectiveness than was used in the previous staff analysis. Further, the staff now includes leakage from Engineered Safety Features recirculation system components outside containment in its LOCA dose calculations. We estimated that at the previous technical specification limit on this leakage, this source alone could result in significant LOCA doses.

After meeting with the staff to discuss the LOCA dose reanalysis results, the licensee agreed to technical specifications changes for the Palisades plant which would decrease the ESF allowable leak limit from 0.5 gpm to 0.2 gpm and require substituting trace level hydrazine (greater than 50 ppm N_2H_2) for sodium hydroxide (NaOH) in the containment spray system. We conclude that the hydrazine spray solution would achieve a much higher iodine removal efficiency during injection than the NaOH-buffered (pH=7) spray solution. With these two changes to the Palisades plant, we find that the potential LOCA offsite doses would be less than 10 CFR, Part 100 guidelines for power levels up to 2650 MWt. The assumptions made and results obtained in our final LOCA dose estimates are given in Tables 1 and 2.

Although the present systems at the Palisades facility are adequate to insure that 10 CFR Part 100 guidelines for offsite doses are met, the licensee has committed to examine areas where system improvements could be made. Specifically, the applicant will examine the present system of delivering hydrazine to the borated water spray solution by gravity feed and positive injection systems to see if the latter system could enhance performance. In addition, an extension of the injection period will be further examined. With regard to the long-term post-LOCA pH control, the licensee will evaluate alternate methods of chemical addition by which the pH will be maintained during the circulation following a LOCA. The time delay feature controlling the opening of the hydrazine isolation valves affects the assumed time at which hydrazine is injected and thus the offsite doses. The licensee will also address the elimination of the time delay. The licensee will further examine means available for minimizing the potential consequences of passive failures after a LOCA. The licensee will study these items to determine where improvements in the Palisades plant dose mitigating features may be made. The results of his analysis are to be sent to the staff by December 1, 1977.

By letter dated October 25, 1977, the licensee agreed to Technical Specification changes for the Palisades plant reflecting the assumptions used in the staff's recent LOCA dose reanalysis. We reviewed the proposed Specifications and found that they did appropriately reflect the assumptions made in our current analysis.

Based on the foregoing, we believe the Palisades plant may be operated at power levels up to 2530 MWt without threatening the public health and safety.

TABLE 9.0-1

ASSUMPTIONS USED IN NRC STAFF'S ANALYSIS OF OFFSITE DOSES FROM POSTULATED ACCIDENTS AT PALISADES

A. Loss of Coolant Accident

1. Regulatory Guide 1.4 Assumptions plus Standard Review Plan 15.6.5, Appendices A and B review procedures with specific values for key parameters noted below.

2. Exclusion Area Boundary distance - 700 meters, LPZ Boundary distance = 5000 meters.

3. X/Q's from Figures 2(A) and 2(B) in Regulatory Guide 1.4, with a building wake factor of 2.09, where applicable :

Time Period (hrs)	X/Q (sec/cubic meter)	Location
0-2	5.5×10^{-4}	EAB
0-8	6.0×10^{-5}	LPZ
8-24	1.2×10^{-5}	LPZ
24-96	4.1×10^{-6}	LPZ
96-720	8.2×10^{-7}	LPZ

4. Containment leak rate = 0.1%/day

5. Containment spray description:

Time Period (hrs)	Spray Solution	Iodine Removal Rate (hr^{-1})
0 - .01667	Borated water	.42 Elemental/ 1.0 Particulate
.01667 - .25	Borated water + >50 ppm N_2H_2	10 Elemental/ 1.0 Particulate
.25 - 720	Containment Sump water	0 Elemental/ 1.0 Particulate

6. Containment free volume = 1.64 million cubic feet

TABLE 9.0-1 (cont)

7. Volume sprayed by containment spray system - 1.48 million cubic feet.

8. Rate of air exchange between unsprayed and sprayed containment regions = 2 unsprayed region volumes per hour.

9. Engineered Safety Features components outside containment leak rate = 0.2 gpm.

10. Iodine partition factor for the ESF leakage = 10

11. Iodine plateout factor due to hi-radiation trip of ESF cubicles ventilation system if significant leakage occurred: 2.

12. ESF recirculation begins at .33 hours after the LOCA.

13. ESF recirculation solution volume = 46000 cubic feet of water.

B. Rod Ejection Accident

1. Regulatory Guide 1.77 Assumptions and Standard Review Plan 15.4.8, Appendix review procedures, plus the same assumptions used for the LOCA dose analysis except for specific assumptions or differences noted below:

2. Palisades FSAR estimates of failed fuel resulting from a rod ejection accident are still conservative at 2650 MWt because of fuel design and operating limit changes made since the FSAR was issued. The FSAR indicated 0.3% of the fuel pins in the core might suffer clad perforation, 0.1% might suffer centerline melting.

3. Effective core iodine inventory fraction released to containment = 0.055%.

4. Effective core noble gas inventory fraction released to containment = 0.13%.

5. Primary coolant mass = 2.28×10^8 grams, primary coolant volume = 10900 ft^3.

TABLE 9.0-1 (continued)

6. Primary-to-secondary system leak rate = 1 gpm for 24 hours.

7. Effective core iodine inventory fraction released to primary coolant = 0.08%

8. Effective core noble gas inventory fraction released to primary coolant = 0.13%

9. Iodine partition factor between water and steam in the secondary system = 10.

TABLE 9.0-2

SUMMARY OF ESTIMATED OFFSITE DOSES
FOR POSTULATED ACCIDENTS AT PALISADES

Loss-of-Coolant Accident Doses, rem

		Containment Leakage Contribution	ECCS Leakage Contribution	Total
Exclusion Area Boundary	Thyroid	182	98	280
	Total Body	3.2	.25	3.5
Low Population Zone	Thyroid	86	70	156
	Total Body	.8	.1	.9

Rod Ejection Accident Doses, rem

		Containment Leakage Doses	Secondary System Release Doses
Exclusion Area Boundary	Thyroid	1.5	4.1
	Total Body	.01	.17
Low Population Zone	Thyroid	.84	2.2
	Total Body	.002	.0002

10. SUMMARY

Based on our review of the information provided, we conclude that it is acceptable to operate the Palisades Plant at an increased power level of 2530 MWt for the remainder of cycle 2.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 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: November 1, 1977

REFERENCES

1. Letter from R.A. Lamley to J.F. O'Leary dated January 22, 1974.

2. "Palisades Stretch Power LOCA Analyses Using the ENC WREM-Based PWR ECCS Evaluation Model Example Problem", XN-NF-77-9, May 1977.

3. "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt", XN-NF-77-18, July 1977.

4. "Steady-State Thermal Hydraulic and Neutronics Analysis of the Palisades Reactor for Operation at 2530 MWt", XN-NF-77-22, July 15, 1977.

5. "LOCA Analysis for Palisades at 2530 MWt Using the ENC WREM-II PWR ECCS Evaluation Model", XN-NF-77-24, July 1977.

6. "LOCA Analysis for Palisades Type D Fuel at 2530 MWt Using the ENC WREM-II PWR ECCS Evaluation Model", XN-NF-77-24, Supplement 1, August 1977.

7. Letter from D.P. Hoffman to A. Schwencer dated July 28, 1977.

8. Letter from D.P. Hoffman to A. Schwencer dated August 12, 1977.

9. Letter from D.P. Hoffman to A. Schwencer dated September 12, 1977.

10. Letter, from D.P. Hoffman to A. Schwencer dated September 13, 1977.

11. Leter from D.P. Hoffman to A. Schwencer dated September 26, 1977.

12. K. R. Merckx, "Cladding Collapse Calculational Procedure", XN-72-23, November 1972.

13. Letter from D. P. Hoffman to A. Schwencer dated October 11, 1977.

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE DIVISION OF SITE SAFETY AND ENVIRONMENTAL ANALYSIS

SUPPORTING

AMENDMENT NO. 31 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

PALISADES NUCLEAR GENERATING PLANT

DOCKET NO. 50-255

Description of Proposed Action

By letters dated April 12, 1977 and September 26, 1977, the applicant, Consumers Power Company, filed a request with the Nuclear Regulatory Commission (NRC) to amend Provisional Operating License No. DPR-20 to permit operation of the Palisades Plant at an increased power level of 2530 MWt. The proposed action is issuance of such a license amendment.

Need for Power Increase

The Palisades Nuclear Generating Plant is now authorized by the license to operate at 2200 MWt (686 MWe). Sections 7 and 9 of the Draft Addendum to the Final Environmental Statement (NUREG-0125) related to operation of Palisades Nuclear Generating Plant, November 1976 (Draft Addendum), established the need for an additional 100 MWe of capacity (to bring the plant power level up to 2638 MWt or 786 MWe). The Final Addendum to the Final Environmental Statement is presently being prepared which confirms the need for this additional generating capacity considering more current data and agency comments. Thus, the requested power increase to 2530 MWt is needed.

Environmental Impact of the Proposed Action

Section 8.1 of the Draft Addendum contains the NRC Staff's evaluation of the environmental impacts of plant operation at the increased power level of 2638 MWt. The Staff then concluded that operation at that increased power level would produce no additional adverse environmental impacts beyond those predicted and described in the Commission's Final Environmental Statement issued in July 1972. The Final Addendum presently being prepared confirms this finding.

Conclusion and Basis for Negative Declaration

On the basis of the foregoing discussion and the NRC Staff evaluation, it is concluded that there will be no environmental impact attributable to the proposed action other than that predicted and described in the Commission's FES issued in July 1972. Having reached this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Date: November 1, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

AND NEGATIVE DECLARATION

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Provisional Operating License No. DPR-20 issued for Consumers Power Company which revised Technical Specifications to operation of the Palisades Plant, located in Covert Township, Van Buren County, Michigan. The amendment is effective as of the date of issuance.

This amendment authorizes operation of the Palisades Plant at power levels up to 2530 megawatts thermal.

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. Notice of Receipt of Application for Full Term License, which includes a request for a power increase,was published in the FEDERAL REGISTER on December 18, 1974 (39 FR 43753). No request for a hearing or petition for leave to intervene was filed.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action.

For further details with respect to this action, see (1) the application for amendment dated August 12, 1977, as supplemented September 26, 1977, (2) Amendment No. 31 to License No. DPR-20, (3) the Commission's related Safety Evaluation, and (4) the Commission's related Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan 49006. A copy of items (2), (3) and (4) 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 1st day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors