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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
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 November 1, 1991, 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;

   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 the license amendment and Paragraph 2.C.2 of Facility Operating License No. DPR-20 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 143 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

for L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 27, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-2	1-2
2-4	2-4
2-9	2-9
3-3	3-3
3-3a	3-3a
3-29	3-29
3-31	3-31
3-66b	3-66b
3-67	3-67
3-105	3-105
3-107	3-107
3-111	3-111

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 2% of rated power, not including decay heat and primary system temperature and pressure shall be in the range of 260°F to 5[illegible]°F and 415 psia to 2150 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions therefore in these Technical Specifications.

Shutdown Boron Concentrations

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sufficient to provide $K_{eff} \leq 0.98$ with all control rods in the core and the highest worth control rod fully withdrawn.

Refueling Boron Concentration

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of coolant at least 1720 ppm (corresponding to a shutdown margin of at least 5% $\Delta\rho$ with all control rods withdrawn).

Quadrant Power Tilt

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

Assembly Radial Peaking Factor - F_r^A

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

Total Radial Peaking Factor - F_r^T

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

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM

Applicability

This specification applies to reactor trip settings and bypasses for instrument channels.

Objective

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1.

The TM/LP trip system monitors core power, reactor coolant maximum inlet temperature, (T_{in}), core coolant system pressure and axial shape index. The low pressure trip limit (P_{var}) is calculated using the following equation.

$$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9493$$

where:

$$
\begin{aligned}
QR_1 &= 0.412(Q) + 0.588 && Q \le 1.0 \qquad Q = \frac{\text{core power}}{\text{rated power}} \\
&= Q && Q > 1.0 \\
ASI &= 0 \text{ when } Q < 0.0625 \\
QA &= -0.720(ASI) + 1.028 \text{ when } && -0.628 \le ASI < -0.100 \\
&= -0.333(ASI) + 1.067 \text{ when } && -0.100 \le ASI < +0.200 \\
&= +0.375(ASI) + 0.925 \text{ when } && +0.200 \le ASI \le +0.565
\end{aligned}
$$

The calculated limit (P_{var}) is then compared to a fixed low pressure trip limit (p_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured reactor coolant pressure (P) and a trip signal is generated when P is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting $P_{trip} + \Delta P$.

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

Basis (Contd)

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 730 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 shutdown before the initiation of the safety injection system and containment spray.[10]

8. Low Power Physics Testing - For low power physics tests, certain tests will require the reactor to be critical at low temperature (≥260°F) and low pressure (≥415 psia). For these certain tests only, the thermal margin/low pressure, primary coolant flow 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 and low flow trip are 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) EMF-91-176, Table 15.0.7-1
(2) deleted
(3) Updated FSAR, Section 7.2.3.3.
(4) EMF-91-176, Section 15.0.7.1
(5) XN-NF-86-91(P)
(6) deleted
(7) deleted
(8) ANF-90-078, Section 15.1.5
(9) ANF-87-150(NP), Volume 2, Section 15.2.7
(10) Updated FSAR, Section 7.2.3.9.
(11) ANF-90-078, Section 15.2.1

Basis (Cont'd)

measurement; ±0.06 for ASI measurement; ±50 psi for pressurizer pressure; ±7°F for inlet temperature; and 3% measurement and 3% bypass for core flow. In addition, transient biases were included in the derivation of the following equation for limiting reactor inlet temperature:

$$T_{Inlet} \le 542.99 + .0580(P-2060) + 0.00001(P-2060)**2 + 1.125(W-138) - .0205(W-138)**2$$

The limits of validity of this equation are:

$1800 \le$ pressure ≤ 2200 psia
$100.0 \times 10^6 \le$ Vessel Flow $\le 150 \times 10^6$ lb/h
ASI as shown in Figure 3.0

With measured primary coolant system flow rates > 150 M lbm/hr, limiting the maximum allowed inlet temperature to the T_{Inlet} LCO at 150 M lbm/hr increases the margin to DNB for higher PCS flow rates.

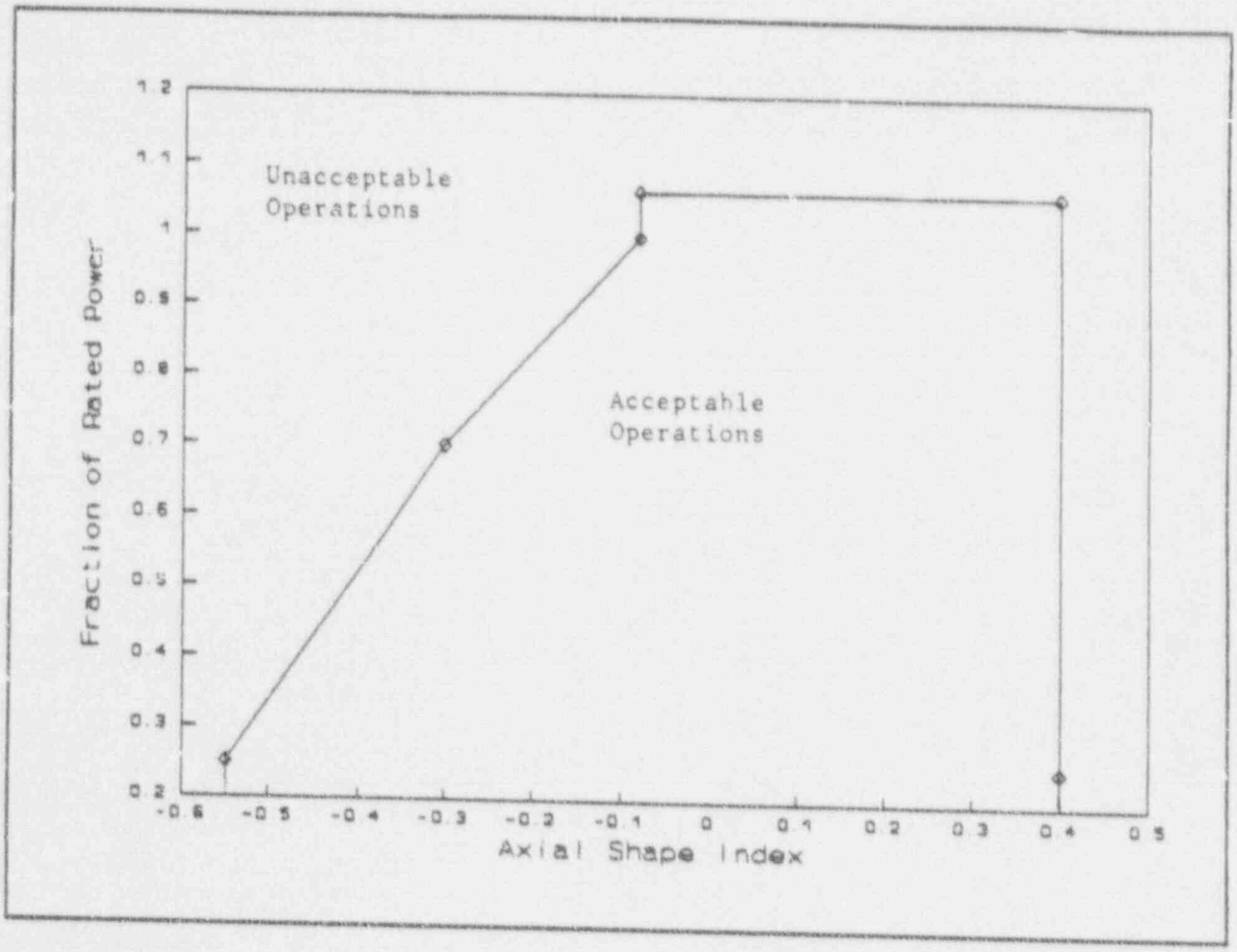
The Axial Shape Index alarm channel is being used to monitor the ASI to ensure that the assumed axial power profiles used in the development of the inlet temperature LCO bound measured axial power profiles. The signal representing core power (Q) is the auctioneered higher of the neutron flux power and the Delta-T po er. The measured ASI calculated from the excore detector signals and adjusted for shape annealing (Y_I) and the core power constitute an ordered pair (Q,Y_I). An alarm signal is activated before the ordered pair exceed the boundaries specified in Figure 3.0.

The requirement that the steam generator temperature be ≤ the PCS temperature when forced circulation is initiated in the PCS ensures that an energy addition caused by heat transferred from the secondary system to the PCS will not occur. This requirement applies only to the initiation of forced circulation (the start of the first primary coolant pump) when the PCS cold leg temperature is < 430°F. However, analysis (Reference 6) shows that under limited conditions when the Shutdown Cooling System is isolated from the PCS, forced circulation may be initiated when the steam generator temperature is higher than the PCS cold leg temperature.

References

(1) Updated FSAR, Section 14.3.2.
(2) Updated FSAR, Section 4.3.7.
(3) Deleted
(4) EMF-91-176 Section 15.0.7.1
(5) ANF-90-078
(6) Consumers Power Company Engineering Analysis EA-A-NL-89-14-1

ASI Limit for T_{inlet} function



Break Points:
-0.550, 0.250
-0.300, 0.700
-0.080, 1.000
-0.080, 1.065
+0.400, 1.065
+0.400, 0.250

FIGURE 3-0

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 2500 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 174 inches and a maximum level of 200 inches with a boron concentration of at least 1720 ppm but not more than 2500 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 cooling 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 opened with HS-3027 and 3056 positions to maintain them open.

3.3 EMERGENCY CORE COOLING SYSTEM (Continued)

c. If Specification a. and b. cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown condition within 12 hours, and cold shutdown within the next 24 hours.

Basis

The normal procedure for starting the reactor is, first, to heat the primary coolant to near operating temperature by running the primary coolant pumps. The reactor is then made critical by withdrawing control rods and diluting boron in the primary coolant.[1] With this mode of start-up, the energy stored in the primary coolant during the approach to criticality is substantially equal to that during power operation and, therefore, all engineered safety features and auxiliary cooling systems are required to be fully operable. During low-temperature physics tests, there is a negligible amount of stored energy in the primary coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards' systems are not required.

The SIRW tank contains a minimum of 250,000 gallons of water containing a minimum of 1720 ppm boron and a maximum of 2500 ppm. This is sufficient boron concentration to provide a 5% shutdown margin with all control rods withdrawn and a new core at a temperature of 60°F.

Heating steam is provided to maintain the tank above 40°F to prevent freezing. The 1.43% boron (2500 ppm) solution will not precipitate out above 32°F. The source of steam during normal plant operation is extraction steam line in the turbine cycle.

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 174-inch level corresponds to a volume of 1040 ft^3 and the maximum 200-inch level corresponds to a volume of 1176 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the

3.11.2 EXCORE POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

Basis (Contd)

Surveillance requirements ensure that the instruments are calibrated to agree with the incore measurements and that the target AO is based on the current operating conditions. Updating the Excore Monitoring APL ensures that the core LHR limits are protected within the ±0.05 band on AO. The APL considers LOCA based LHR limits, and factors are included to account for changes in radial power shape and LHR limits over the calibration interval.

The APL is determined from the following:

$$APL = \left[\frac{LHR(Z)_{TS}}{LHR(Z)_{Max} \times V(Z) \times 1.02} \right]_{Min} \times \text{Rated Power}^{(2)}$$

Where:

(1) $LHR(Z)_{TS}$ is the limiting LHR vs Core Height (from Section 3.23.1),

(2) $LHR(Z)_{Max}$ is the measured peak LHR including uncertainties vs Core Height,

(3) V(Z) is the function (shown in Figure 3.11-1),

(4) The factor of 1.02 is an allowance for the effects of upburn,

(5) The quantity in brackets is the minimum value for the entire core at any elevation (excluding the top and bottom 10% of core) considering limits for peak rods. If the quantity in brackets is greater than one, the APL shall be the rated power level.

References

(1) XN-NF-80-47

(2) EMF-91-177

(next page is 3-66d)

3.12 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to the moderator temperature coefficient of reactivity for the core.

Objective

To specify a limit for the positive moderator coefficient.

Specifications

The moderator temperature coefficient (MTC) shall be less positive than $+0.5 \times 10^{-4}\ \Delta\rho/^{\circ}F$ at $\leq 2\%$ of rated power.

Bases

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the safety analysis[1] remain valid.

Reference

(1) EMF-91-176, Section 15.0.5

POWER DISTRIBUTION LIMITS

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

Basis (Contd)

The time interval of 2 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 monitoring systems are returned to service.

To ensure that the design margin of safety is maintained, the determination of both the incore alarm setpoints and the APL takes into account a measurement uncertainty factor of 1.10, an engineering uncertainty factor of 1.03, a thermal power measurement uncertainty factor of 1.02 and allowance for quadrant tilt.

References

(1) EMF-91-177

(2) (Deleted)

(3) (Deleted)

(4) XN-NF-80-47

(next page is 3-107)

TABLE 3.23-1

LINEAR HEAT RATE LIMITS

	No. of Fuel Rods Assembly	
	208	216
Peak Rod	15.28 KW/ft	15.28 kW/ft

TABLE 3.23-2

RADIAL PEAKING FACTOR LIMITS, F_r

Peaking Factor	No. of Fuel Rods in Assembly		
	208	216 (Reload M and earlier)	216
Assembyl F_r^A	1.48	1.57	1.66
Peak Rod F_r^T	1.92	1.92	1.92

POWER DISTRIBUTION LIMITS

3.23.2 RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

The radial peaking factors F_r^A, and F_r^T shall be less than or equal to the value in Table 3.23-2 times the following quantity. The quantity is [1.0 + 0.3 (1 - P)] for P ≥ .5 and the quantity is 1.15 for P < .5. P is the core thermal power in fraction of rated power.

APPLICABILITY: Power operation above 25% of rated power.

ACTION:

1. For P < 50% of rated with any radial peaking factor exceeding its limit, be in at least hot shutdown within 6 hours.

2. For P ≥ 50% of rated with any radial peaking factor exceeding its limit, reduce thermal power within 6 hours to less than the lowest value of:

$$[1 - 3.33(\frac{F_r}{F_L} - 1)] \times \text{Rated Power}$$

Where F_r is the measured value of either F_r^A, or F_r^T and F_L is the corresponding limit from Table 3.23-2.

Basis

The limitations on F_r^A, and F_r^T are provided to ensure that assumptions used in the analysis for establishing DNB margin, LHR and the thermal margin/low-pressure and variable high-power trip set points remain valid during operation. Data from the incore detectors are used for determining the measured radial peaking factors. The periodic surveillance requirements for determining the measured radial peaking factors provide assurance that they remain within prescribed limits. Determining the measured radial peaking factors after each fuel loading prior to exceeding 50% of rated power provides additional assurance that the core is properly loaded.

The LOCA analysis supports the radial peaking factor limits in Table 3.23-2.