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. 51
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 January 3, 1978, as supported by information transmitted by letters dated March 8, 1977, June 24, 1977, and November 28, 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 3.B of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 51 , 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

Dennis L. Ziemann

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 10, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 51

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A by removing the pages described below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
i	i
3-1a	3-1a
3-2	3-2*
3-3	3-3
---	3-25a
3-29a	3-29a*
3-30	3-30
3-33	3-33
4-1	4-1
4-2	4-2
---	4-2a*
4-39	4-39

*There were no changes made to the provisions contained on these pages. The Technical Specifications have merely been repositioned.

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PALISADES PLANT
TECHNICAL SPECIFICATIONS

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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.

h. A reactor coolant pump shall not be started with one or more of the PCS cold leg temperatures ≤ 250°F unless 1) the pressurizer water volume is less than 700 cubic feet or 2) the secondary water temperature of each steam generator is less than 70°F above each of the PCS cold leg temperatures.

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Basis

When primary coolant boron concentration is being changed, the process must be uniform throughout the primary coolant system volume to prevent stratification of primary coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one shutdown cooling or one primary coolant pump is in operation.[1] The shutdown cooling pump will circulate the primary system volume in less than 60 minutes when operated at rated capacity. The pressurizer volume is relatively inactive, therefore will tend to have a boron concentration higher than rest of the primay coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the primary system during the addition of boron.[2]
Both steam generators are required to be operable whenever the temperature of the primary coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

Calculations have been performed to demonstrate that a pressure differential of 1380 psi can be withstood by a tube uniformily thinned to 36% of its original nominal wall thickness (64% degradation), while maintaining:

(1) A factor of safety of three between the actual pressure differential and the pressure differential required to cause bursting.
(2) Stresses within the yield stress for Inconel 600 at operating temperature.
(3) Acceptable stresses during accident conditions.

The maximum transient steam generator differential pressure is expected to occur during the loss of load accident. The loss of load accident inititated 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

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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:

$1850 \leq$ Pressure ≤ 2250 Psia

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

The restrictions on starting a Reactor Coolant Pump with one or more PCS cold legs ≤ 250°F are provided to prevent PCS pressure transients, caused by energy additions from the secondary system, which would exceed the limits of Appendix G to 10 CFR Part 50. The PCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 70°F above each of the PCS cold leg temperatures.(5)

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.

(5) "Palisades Plant Overpressurization Analysis," June, 1977, and "Palisades Plant Primary Coolant System Overpressurization Subsystem Description," October, 1977.

3.1.8 Overpressure Protection Systems Specifications

a. When the temperature of one or more of the primary coolant system cold legs is ≤ 250°F, two power operated relief valves (PORVs) with a lift setting of ≤ 400 psia, or a reactor coolant system vent of ≥ 1.3 square inches shall be operable except as specified below:

   (1) With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize and vent the PCS through a ≥ 1.3 square inch vent(s) within the next 8 hours; maintain the PCS in a vented condition until both PORVs have been restored to operable status.

   (2) With both PORVs inoperable, depressurize and vent the PCS through a ≥ 1.3 square inch vent(s) within 8 hours; maintain the PCS in a vented condition until both PORVs have been restored to operable status.

b. In the event either the PORVs or the PCS vent(s) are used to mitigate a PCS pressure transient, a Special Report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

Basis

The OPERABILITY of two PORVs or an PCS vent opening of greater than 1.4 square inches ensures that the PCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the PCS cold legs are ≤ 250°F. Either PORV has adequate relieving capability to protect the PCS from overpressurization when the transient is limited to either (1) the start of an idle PCP with the secondary water temperature of the steam generator ≤ 70°F above the PCS cold leg temperatures or (2) the start of a HPSI pump and its injection into a water solid PCS.[1]

References

(1) "Palisades Plant Overpressurization Analysis," June, 1977, and "Palisades Plant Primary Coolant System Overpressurization Subsystem Description," October, 1977.

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3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

3.3.2 During power operation, the requirements of 3.3.1 may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the requirements of 3.3.1 within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of 3.3.1 are not met within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

a. One safety injection tank may be inoperable for a period of no more than one hour.

b. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours. The other low-pressure safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

c. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours. The other high-pressure safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

d. One shutdown heat exchanger and one component cooling water heat exchanger may be inoperable for a period of no more than 24 hours.

e. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

f. Any valve, interlock or pipe associated with the safety injection and shutdown cooling system and which is not covered under 3.3.2e above but, which is required to function during accident conditions, may be inoperable for a period of no more than 24 hours. Prior to initiating repairs, all valves and interlocks in the system that provide the duplicate function shall be tested to demonstrate operability.

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g. A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the PCS cold legs is $\leq$ 250°F.

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

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that 25% of their combined discharge rate is lost from the primary coolant system out the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR Figures 14.17.9 to 14.17.13. These demonstrate that the maximum fuel clad temperatures that could occur over the break size spectrum are well below the melting temperature of zirconium (3300°F).

Malfunction of the Low Pressure Safety Injection Flow control valve could defeat the Low Pressure Injection feature of the ECCS; therefore, it is disabled in the 'open' mode (by isolating the air supply) during plant operation. This action assures that it will not block flow during Safety Injection.

The inadvertent closing of any one of the Safety Injection bottle isolation valves in conjunction with a LOCA has not been analyzed. To provide assurance that this will not occur, these valves are electrically locked open by a key switch in the control room. In addition, prior to critical the valves are checked open, and then the 480 volt breakers at MCC 9 are opened. Thus, a failure of a breaker and a switch are required for any of the valves to close.

The limitation for a maximum of one high pressure safety injection pump to be operable, and the Surveillance Requirement to verify all high pressure safety injection pumps except the required operable pump to be inoperable below 250°F, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

References

(1) FSAR, Section 9.10.3.

(2) FSAR, Section 6.1.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance requirements shall be applicable during the reactor operating conditions associated with individual Limiting Conditions for Operation unless otherwise stated in an individual surveillance requirement.

4.0.2 Unless otherwise specified, each surveillance requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. A total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.1 INSTRUMENTATION AND CONTROL

Applicability

Applies to the reactor protective system and other critical instrumentation and controls.

Objective

To specify the minimum frequency and type of surveillance to be applied to critical plant instrumentation and controls.

Specifications

Calibration, testing, and checking of instrument channels, reactor protective system and engineered safeguards system logic channels and miscellaneous instrument systems and controls shall be performed as specified in 4.1.1 and in Tables 4.1.1 to 4.1.3.

4.1.1 Overpressure Protection Systems

a. Each PORV shall be demonstrated operable by:

1. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

2. Performance of a channel calibration on the PORV actuation channel at least once per 18 months.

3. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4. Testing in accordance with the inservice inspection requirements for ASME Section XI, Section IWV Category C valves.

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b. The PCS vent(s) shall be verified to be open at least once per 12 hours when the vent(s) is being used for overpressure protection except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

Basis

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems when the plant is in operation, a checking frequency of once-per-shift is deemed adequate for reactor and steam system instrumentation. Calibrations are performed to insure the presentation and acquisition of accurate information.

The power range safety channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at each refueling shutdown interval.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures. Thus, minimum calibration frequencies of one-per-day for the power range safety channels, and once each refueling shutdown for the process system channels, are considered adequate.

The minimum testing frequency for those instrument channels connected to the reactor protective system is based on an estimated average unsafe failure rate of 1.14×10^{-5} failure/hour per channel. This estimation is based on limited operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

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For the specified one-month test interval, the average unprotected time is 360 hours in case of a failure occurring between test intervals, thus the probability of failure of one channel between test intervals is $360 \times 1.14 \times 10^{-5}$ or 4.1×10^{-3}. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two-out-of-four channels is $(4.1 \times 10^{-3})^3 = 6.9 \times 10^{-8}$. This represents the fraction of time in which each four-channel system would have one operable and three inoperable channels and equals $6.9 \times 10^{-8} \times 8760$ hours per year, or 2.16 seconds/year.

These estimates are conservative and may be considered upper limits. Testing intervals will be adjusted as appropriate based on the accumulation of specific operating history.

The testing frequency of the process instrumentation is considered adequate (based on experience at other conventional and nuclear plants on Consumers Power Company's system) to maintain the status of the instruments so as to assure safe operation. As the reactor protection system is not required when the plant is in a refueling shutdown condition, routine testing is not required.

Those instruments which are similar to the reactor protective system instruments are tested at a similar frequency and on the same basis.

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4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Applicability

Applies to the safety injection system, the containment spray system, chemical injection system and the containment cooling system tests.

Objective

To verify that the subject systems will respond promptly and perform their intended functions, if required.

Specifications

4.6.1 Safety Injection System

a. System tests shall be performed at each reactor refueling interval. A test safety injection signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this test.

b. The system test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing (ie, the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

c. All high pressure safety injection pumps except those otherwise required to be operable shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the PCS cold legs is $\leq$ 250°F by verifying that the control system fuses and their fuse holders for the HPSI pumps (P66A, P66B and P66C) have been removed from the circuit.

4.6.2 Containment Spray System

a. System tests shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. At least every five years the spray nozzles shall be verified to be open.

c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.6.3 Pumps

a. The safety injection pumps, shutdown cooling pumps, and containment spray pumps shall be started at intervals not to exceed three months. Alternate manual starting between control room console and the C-33 panel shall be practiced in the test program.

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