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Docket No. 50-255

SEP 10 1979

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

Dear Mr. Bixel:

The Commission has issued the enclosed Amendment No. 51 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your request dated January 3, 1978, as supported by information transmitted by letters dated March 8, 1977, June 24, 1977, and November 28, 1977.

This amendment authorizes changes that will enhance low temperature overpressure protection and increase assurance that the reactor vessel will not be subjected to pressure transients which could exceed the limits established in accordance with Appendix G of 10 CFR Part 50.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

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

Enclosures:
1. Amendment No 51 to DPR-20
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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

September 10, 1979

Docket No. 50-255

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

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This amendment authorizes changes that will enhance low temperature overpressure protection and increase assurance that the reactor vessel will not be subjected to pressure transients which could exceed the limits established in accordance with Appendix G of 10 CFR Part 50.

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cc w/enclosures:
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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. 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. <u>Technical Specifications</u>

   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.

PALISADES PLANT
TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

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 $\leq$ 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.

3.1 PRIMARY COOLANT SYSTEM (Contd)

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

PRIMARY COOLANT SYSTEM (Contd)

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:

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

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

The restrictions on starting a Reactor Coolant Pump with one or more PCS cold legs $\leq$ 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 $\leq$ 250°F, two power operated relief valves (PORVs) with a lift setting of $\leq$ 400 psia, or a reactor coolant system vent of $\geq$ 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 $\geq$ 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 $\geq$ 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 $\leq$ 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 $\leq$ 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.

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.

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

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

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

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.

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.

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.

4.6 SAFETY INJECTI(AND CONTAINMENT SPRAY SYSTEMS (STS

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 51 TO LICENSE NO. DPR-20
CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET NO. 50-255

1.0 Introduction

By letter dated January 3, 1978, Consumers Power Company (CPC) requested changes to the Technical Specifications for the Palisades Plant. The proposed changes would establish requirements related to prevention of low temperature overpressurization events. Supporting information was submitted by letters dated March 8, 1977, June 24, 1977, and November 28, 1977. The CPC submittals are in response to NRC requests related to the generic issue of PWR overpressure protection.

2.0 Background

The history of the generic low temperature overpressure protection issue is described in NUREG-0138 (Reference 1). Briefly, a series of over thirty incidents had occurred in pressurized water reactors (PWRs) since 1972 in which the Appendix G pressure-temperature limits had been exceeded at temperatures less than normal operating temperature.

These incidents consisted of two varieties of pressure transients: a mass input type from charging pumps, safety injection pumps, or safety injection accumulators, and an energy input type caused by thermal feedback when a reactor coolant pump (RCP) sweeps cooler primary system water through a steam generator with a hot secondary side. These incidents usually occurred in a water solid system during startup or shutdown operations.

Pressure transients which could occur at normal operating temperature, approximately 570°F, are mitigated in most plants by large code safety valves located on the pressurizer. These are mechanical valves which open against a spring pressure of about 2400 psia. The code safety valves are quite simple, having no electrical components, and as such are considered passive, failure free components. These code safety valves are tested in accordance with ASME Code, Section XI requirements.

Prior to the introduction of an overpressure protection system (OPS), pressure transients initiated while operating at lower temperatures were not protected against and there were no pressure relief devices in the reactor coolant system to prevent these transients from exceeding the Appendix G pressure-temperature limits. Nuclear reactors such as Palisades, which have a pressure limit in excess of 2500 psia at 570°F, have only a 700 psia limit at 200°F. The code safety valves with settings in the 2400 psia range would not be able to relieve a pressure transient at low reactor coolant system (RCS) temperature without the limits of 10 CFR Part 50 Appendix G being violated by a large amount.

The Appendix G pressure limit drops off rapidly at lower temperatures because the reactor vessel material and welds have significantly less toughness at lower temperatures and are therefore more susceptible to flaw induced failure. In addition, factors such as copper content in welds and neutron fluence levels affect the material toughness and contribute to the reduction in safety margin to vessel failure at low temperature conditions.

In a series of meetings and through correspondence with PWR vendors and licensees, the staff developed a set of criteria, which if adhered to, would produce an acceptable OPS. These criteria are:

a. Operator Action - No credit can be taken for operator action until ten minutes after the operator is aware, through an action alarm, that a pressure transient is in progress.

b. Single Failure Criterion - The low temperature overpressure protection system should be designed to protect the reactor vessel given a single failure in addition to the event that initiated the pressure transient.

c. Testability - The system must be testable on a periodic basis on a schedule consistent with the frequency that the system is relied upon for low temperature overpressure protection.

d. Seismic and IEEE 270 Criteria - The system should meet both seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a failure mode that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

In addition to the four formally stated criteria mentioned above, a number of additional criteria were established in the process of the staff review of generic submittals from the various vendors and in the exchange of information between the staff and the licensees.

Foremost among these was the requirement that the licensees show protection for the limiting mass addition transient regardless of the administrative procedures proposed to eliminate that potential scenario. Each licensee, therefore, was required to analyze the effects of the single pump start which would produce the most limiting mass addition transient and most severely challenge the Appendix G limits.

For the worst case energy addition transient the licensees were allowed to limit the severity of the transient in their analyses by assuming a maximum ΔT across the steam generator. By maximum ΔT we mean the maximum difference in the temperature between the primary loop coolant and the secondary loop water in the steam generator. For this case and for other scenarios the licensees were required to develop Technical Specifications which delineated the actions required to limit the severity of these scenarios and also provide justification for their action.

Another criterion for the design of the OPS was that the electrical instrumentation and control system provide a variety of alarms to alert the operator to 1) properly enable the low temperature OPS at the proper temperature during cooldown, and 2) indicate if a pressure transient was occurring. Additionally the electrical system had to provide positive assurance that the isolation valve upstream of each PORV was open when the system was enabled by wiring its position into the enable alarm. The enable alarm would not be permitted to clear until the OPS mode selector switch for each PORV system was placed in the low pressure setpoint position and the isolation valve was opened.

The Combustion Engineering Owner's Group, comprised of five utilities, submitted a generic overpressurization protection report prepared by Combustion Engineering (CE) (Reference 2). The generic report provided information on RCS response to postulated pressure transients that occur at low temperatures during heatup and cooldown, and provided a general description of design modifications which could be used to prevent overpressurization of CE designed Nuclear Steam Supply Systems (NSSS). In Reference 3 CPC pointed out the difference between the Palisades plant and the 2560 MWt class of plants covered in the generic report in terms of equipment and operating procedures. The staff, in conjunction with its review of the CE generic report, requested that CPC commit to a schedule for implementing a permanent or interim version of the OPS by December 31, 1977, and requested additional information related to the application of the generic aspect of the OPS as pertinent to the Palisades plant (Reference 4). In References 4 through 8 the licensee submitted additional information to the staff on equipment and procedural improvements as well as a schedule for implementation of the proposed system.

The system proposed by CPC for Palisades incorporates a defense in depth concept for overpressure protection, utilizing operator training, administrative procedures, Technical Specifications, and hardware improvements to meet the criteria established by the staff. The objective of the OPS is, first, to ensure that pressure transients while operating at low RCS temperatures become and remain unlikely events, and second, to mitigate the consequences of a pressure transient should one occur. The proposed mitigating system includes sensors, actuating mechanisms, and valves to prevent a RCS pressure transient from exceeding the pressure-temperature limits included in the Palisades Technical Specifications as required by Appendix G to Chapter 10, Code of Federal Regulations, Part 50 (10 CFR 50).

The Palisades final OPS was installed during the 1978 refueling. This conformed to the staff's requirement to install a final version of OPS at least by the first refueling outage after December 31, 1977.

This Safety Evaluation Report presents the results of the staff's review of the installed OPS, administrative controls and proposed changes to the Technical Specifications.

3.0 Low Temperature Overpressure Protection System

The licensee identified pressurizer power operated relief valves (PORV's) and the shutdown cooling system (SDC) safety valves as being capable of providing pressure relief during low temperature operations.

The PORV's are located on the pressurizer and are normally available for overpressure protection during normal plant operations. These valves usually have a single pressure setpoint just below the opening pressure of the mechanical code safety valves and are designed to relieve small pressure transients without requiring the code safety valves to lift. The licensee proposed to provide the PORV's with a low pressure setpoint to which they could be switched as the plant cooled down. If a pressure transient would occur at these lower temperatures and the lower setpoint had been selected, there would then be a pathway to relieve system pressure.

The PORV's are significantly more complicated than the code safety valves since the PORV's require electrical circuitry to sense pressure, transmit a signal to the valve, and actuate the solenoid to open the valve. Thus it is desirable to insure redundancy and separability in the circuitry to preclude a single failure from disabling the entire OPS system.

The SDC safety valves, in addition to the PORV's, are also available for overpressure protection during low temperature operation because the SDC system is aligned and operational. The Palisades SDC system does not have autoclosure and would, therefore, not be isolated because of a pressure transient. There is no electrical circuitry associated with the SDC safety valves and they are considered passive, failure free components.

3.1 System Description and Evaluation

3.1.1 OPS Functioning

Acceptable performance of OPS depends on the proper functioning and adequate relief capacity of the two pressurizer PORV's and a SDC safety relief valve. The two PORV OPS trains are enabled at 300°F during a cooldown and have a setpoint of 415 psia. The SDC system is placed in operation at 225°F. Although the SDC system has two safety relief valves, only one falls into the range of pressures expected during an overpressure transient. These SDC valves have setpoints of 300 psia and 500 psia and are on the suction and discharge sides of the SDC system. The maximum transient pressure for an overpressure event would be less than 500 psia. The NSSS vendor and the licensee demonstrated that with two PORV's and the SDC safety relief valve functioning all postulated mass and energy addition transients could be mitigated. If one PORV is assumed to fail, administrative procedures must be relied upon to limit the severity of the limiting transients in both the energy and mass addition cases to insure that Appendix G limits are not violated.

3.1.2 Energy Addition Transients

The temperature difference across the steam generators must be maintained less than 70°F to prevent a RCP start energy addition transient from violating Appendix G limits. This is accomplished by the careful method by which the plant is cooled down.

Energy Incorporated performed the overpressurization analyses for CPC using the RETRAN computer code, a modified version of RELAP 4, Mod 03, Update 95. RETRAN provides the capability to analyze light water reactor plant transients. Energy Incorporated used a number of CE overpressurization analysis results for comparison with the RETRAN results. In all cases the RETRAN results agreed with the CE results or were more conservative. The following conservatisms were inherent in these RETRAN analyses:

a. The RCS was assumed to be water solid.

b. Metal masses did not act as heat sinks.

c. Pump starts were assumed to be instantaneous.

With one PORV disabled the analysis showed that one PORV and the SDC safety relief valve would limit the maximum pressure to 450 psia for a ΔT of 85°F. For a ΔT of 70°F, a single PORV can relieve the pressure transient which results with a maximum pressure of 470 psia. The initial conditions for these analyses were a primary temperature of 120°F, primary pressure of 270 psia, and secondary temperature of 190°F for the 70°F ΔT case and 205°F for the 85°F ΔT case. The corresponding Appendix G limit at 120°F is 475 psia.

We conclude that the licensee and vendor have demonstrated that the OPS can protect the RCS from exceeding Appendix G limits for an energy addition transient even with the additional single failure of a PORV.

3.1.3 Mass Addition Transients

Protection from the effects of the limiting mass addition transient was afforded by the licensee by assuring that components of the ECCS system would be disabled by procedure and Technical Specification during cooldown. This is accomplished at a pressurizer pressure of 1400 psia by valving out the Safety Injection Tanks and physically removing the control system fuses for the high pressure safety injection pumps. This provides assurance that Appendix G limits will not be violated should a single PORV fail prior to or during a mass addition transient. As noted previously, the overpressurization transients were analyzed using the RETRAN computer code along with appropriately conservative assumptions.

The staff guidance to the licensee for analyzing the mass addition transient was to show that Appendix G limits were not violated assuming that the safety injection pump which could produce the worst case transient inadvertantly started, regardless of administrative procedures calling for disabling the pumps at various stages. For the Palisades plant the worst pump start would be a HPSI pump. The licensee demonstrated that a single HPSI pump start would produce a peak pressure of ∿460 psia assuming the failure of a single PORV and the SDC safety relief valve. The PORV opens at 415 psia, the OPS setpoint, and the pressure continues to rise until the OPS relief rate equals the HPSI input rate at 460 psia. The initial conditions for this transient were a RCS temperature of 120°F and a pressure of 270 psia. The Appendix G limit for a RCS temperature of 120°F is 475 psia.

We conclude that the licensee has demonstrated that the OPS will prevent overpressurization of the RCS due to mass addition transients, assuming the single failure of a PORV.

3.1.4 System Electrical Design

The low temperature overpressure protection system is comprised of two redundant and independent channels designed to comply with the above criteria. Each channel will automatically provide a relief path from the RCS pressurizer to the relief tank should a pressure in excess of 400 psi coexist with a temperature below 250°F. Each channel consists of a power operated relief valve (PORV), a PORV isolation valve, related instrumentation and controls, and other ancillary equipment. The low temperature overpressure protection system is manually enabled by the operator any time the RCS temperature is below 300°F. If, at any time, the RCS temperature is below 300°F amd either overpressure protection system channel is not enabled, a control room annynciator alarm, "NO PCS PROTECTION," will be activated (a single annunciator alarm is provided for both channels). Each channel is enabled by closing a key-operated switch in the control room and by opening the associated isolation valve (also with a switch in the control room). Once enabled, the PORVs will open automatically in response to pressure transients. Annunciators are also provided to alert the operator of an approaching high pressure condition ("PCS PRESSURE 375 PSI") and of pressure grater than 430 psig coincident with temperature less than 250°F ("PORV OPEN"). Single annunciators are provided for both channels. In addition to the above annunciators, control room indicator lights in each channel will inform the operator that (1) the PORV isolation valve is open, (2) the channel is enabled, and (3) the PORV is open. The "PORV OPEN" light will remain 'on' until it is reset by the operator.

The overpressure protection system design as submitted by CPC and described above is in accordance with the criteria in Section II, and therefore is acceptable to the staff.

3.1.5 Inadvertent Operation of SIS Components

The staff position with regard to inadvertent operation of safety injection system (SIS) components during cold shutdown operations requires the deenergization of SIS pumps and closure of safety injection header/discharge valves.

The licensee has agreed to valve out the safety injection tanks and to remove control power from the high pressure safety injection pumps when the RCS pressure is reduced to 1400 psi during cooldown. We find this acceptable.

3.1.6 Testability

The OPS components will be tested at appropriate intervals. A channel functional test will be performed on the electrical circuitry during each refueling. The PORV's are pilot-operated and therefore cannot be tested unless a differential pressure exists across the valve. Also, these valves are physically inaccessible during operation and cooldown. CPC proposes, therefore, to test the valves in accordance with the applicable requirements of ASME Code Section XI, Subsection IWV. We find this acceptable.

4.0 Procedures and Technical Specifications

One cornerstone of the Palisades OPS is the use of operating procedures and Technical Specifications to limit the probability of initiating pressure transients at low temperatures (<250°F) and to insure the enabling, disabling, and proper functioning of the OPS. Procedures and Technical Specifications described and submitted by the licensee are described in the following two sections.

4.1 Procedures

The licensee will make extensive use of operating procedures to provide a large measure of the administrative protection against overpressure transients. Among these operating procedures for low temperature operating conditions are the following:

a. When RCS temperature, pressure, and other operating conditions permit, a pressurizer steam volume of about 60% of the pressurizer volume will be maintained.

b. The licensee will conduct the Palisades plant cooldown in such a manner that the ΔT across the steam generator will be no more than 70°F. This will insure that a RCP start would not be capable of overpressurizing the RCS. The cooldown from hot conditions will be conducted by steaming to the condenser via the turbine bypass valve. Cooling of the primary side continues to 325°F at which point the low head safety injection pumps are realigned for the SDC mode. The RCP's continue to run until the cooldown reaches 160°F to 180°F at which time the system goes solid and the RCP's are stopped. At this stage the secondary side is also in the 160°F to 180°F range Cooldown via SDC continues to 120°F at which time the primary side is reduced to atmospheric pressure. Therefore the ΔT across the steam generator will not be greater than 70°F during the cooldown.

c. ECCS component testing will be conducted with a steam bubble or with the reactor vessel head removed. Operational testing of the Safety Injection and CVCS components (i.e., pumps, valves, automatic signals, etc.) will be accomplished with a non-solid RCS.

We conclude that these operating procedures significantly contribute to plant protection from low temperature overpressure transients and are acceptable. We also conclude that the licensee's method of insuring a maximum ΔT is acceptable.

4.2 Technical Specifications

The following are suggested Technical Specifications submitted by the licensee for implementation as part of the Palisades OPS.

a. The OPS will be enabled prior to reaching 250°F during cooldown.

b. ECCS components which could cause overpressure transients and which are not necessary during low temperature operations will be disabled in stages.

c. OPS components will be tested at appropriate intervals. For Palisades a channel functional test will be conducted during each refueling. Valve functioning will be tested in accordance with ASME Code, Section XI, Subsection IWV.

In addition to the Technical Specifications submitted by the licensee, a Technical Specification concerning inoperability of OPS is required as follows: Both PORV's must be operable whenever the RCS temperature is less than the minimum pressurization temperature, except one PORV may be inoperable for seven days. If these conditions are not met, the primary system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours. We have revised the Technical Specifications to include these requirements and to conform the Technical Specifications to the staff's model Standard Technical Specifications. We have discussed these revisions with representatives of CPC and they have informed us that they find our revisions acceptable.

We conclude that the Technical Specifications submitted by the licensee and as modified by the staff, will provide assurance that pressure transients at low temperatures will be unlikely and that the system will function to prevent overpressure transients from exceeding Appendix G limits. We further conclude that the Technical Specifications as modified by the staff meet the criteria established by the staff and are acceptable.

5.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types, an increase in total amounts of effluents or an increase in power level and therefore will not result in any significant environmental impact. Having made this determination, we have concluded, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

6.0 Conclusions

The system presented by the Consumers Power Company to provide protection for the Palisades plant from low temperature overpressure transients provides assurance that these transients will be unlikely events and that, should they occur, the plant will be protected.

We conclude, therefore, that the Palisades OPS meets the criteria established by the staff for overpressure protection and is acceptable as a low temperature overpressure protection system. We further conclude that the suggested Technical Specifications submitted with the OPS design as modified by us are in consonance with the OPS criteria and are therefore acceptable.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: September 10, 1979

7.0 REFERENCES

1. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff," NUREG-0138, November 1976.

2. Combustion Engineering Owner's Group, "Generic Report Overpressure Protection for Operating CE NSSS".

3. Letter, Sewell to Director, Nuclear Reactor Regulation, Docket No. 50-255, December 6, 1976.

4. Letter, Schwencer to Sewell, Docket No. 50-255, January 10, 1977.

5. Letter, Hoffman to Schwencer, Docket No. 50-255, March 8, 1977.

6. Letter, Hoffman to Schwencer, Docket No. 50-255, June 24, 1977, forwarding "Palisades Plant Overpressurization Analysis", prepared by Energy Incorporated.

7. Letter, Hoffman to Schwencer, Docket No. 50-255, November 28, 1977.

8. Letter, Hoffman to Schwencer, Docket No. 50-255, January 3, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 51 to Provisional Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised Technical Specifications for operation of the Palisades Plant (the facility) located in Covert Township, Van Buren County, Michigan. The amendment is effective as of its date of issuance.

The amendment authorizes changes that will enhance low temperature overpressure protection and increase assurance that the reactor vessel will not be subjected to pressure transients which could exceed the limits established in accordance with Appendix G of 10 CFR Part 50.

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. Prior public notice of this amendment was not required since the amendment did not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated Jnauary 3, 1978, and supporting information transmitted by letters dated March 8, 1977, June 24, 1977, and November 28, 1977, (2) Amendment No. 51 to License No. DPR-20, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan 49006. A copy of items (2) and (3) 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 10th day of September, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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