FEB 1 1 1975

Docket No. 50-255

Consumers Power Company
ATTN: Mr. R. C. Youngdahl
Senior Vice President
212 West Michigan Avenue
Jackson, Michigan 49201

Gentlemen:

The Commission has issued the enclosed Amendment No. 12 to Facility License No. DPR-20, for the Palisades Plant. This amendment includes Change No. 16 to the Technical Specifications, and is in response to your request dated April 8, 1974.

This amendment involves many miscellaneous changes to the Technical Specifications. In some cases, the changes we have made are somewhat different from those you have requested for reasons discussed in the attached Safety Evaluation. These items have also been discussed with members of your staff.

No action has been taken on the changes you requested to Section 6, "Administrative Controls". Our letter to you dated October 21, 1974, requested that you submit revised administrative controls in the format and content of the recently developed standard. This will probably result in many additional changes to this section. We have concluded, therefore, that it would not be fruitful to consider these changes at this time.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
~~Directorate of Licensing~~
Division of Reactor Licensing

Enclosures:
1. Amendment No. 12
2. Safety Evaluation
3. Federal Register Notice

Conc (1)

cc w/enclosures: See next page

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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See Previous Concurrence

OFFICE	L:ORB #1	L:ORB #1	OGC	L:OR KRG	
SURNAME	CTrammell:lb	RAPurple	L. Brenner	KRGoller	
DATE	1/3/75	1/ /75	1/ /75	2/11/75	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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BScharf (15)
TJCarter
P?Collins
SVarga
RSchemel
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ACRS (16)

OFFICE➤	L:ORB #1	L:ORB #1	OGC	L:OR		
SURNAME➤	CTrammell:lb	RAPurple		KRGoller		
DATE➤	11/19/74	11/12/74	11/ /74	11/ /74		

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 12
License No. DPR-20

1. The ~~Atomic Energy~~ Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consumers Power Company (the licensee) dated April 8, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-20 is hereby amended to read as follows:

OFFICE➤						
SURNAME➤						
DATE➤						

"B. Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 16."

3. This license amendment is effective as of the date of its issuance.

FOR THE ~~ATOMIC ENERGY~~ NUCLEAR REGULATORY COMMISSION

Karl R. Goller, Assistant Director
for Operating Reactors
~~Directorate of Licensing~~
Division of Reactor Licensing

Attachment:
Change No. 16 to Technical Specifications

Date of Issuance: FEB 11 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 12

CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

Revise Appendix A as follows:

Remove the following pages and insert identically numbered new pages:

2-3	4-5	4-15
2-5	4-6	4-25
3-15	4-7	4-26
3-19	4-8	4-27
3-25	4-9	4-34
3-75	4-10	4-35
4-2	4-11	4-42
4-3	4-12	4-63
4-4	4-14	

Add the following new pages:

4-13a

4-15a

2.2 SAFETY LIMITS - PRIMARY COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on primary coolant system pressure.

Objective

To maintain the integrity of the primary coolant system and to prevent the release of significant amounts of fission product activity to the primary coolant.

Specification

The primary coolant system pressure shall not exceed 2750 psia when there are fuel assemblies in the reactor vessel.

Basis

The primary coolant system[1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The maximum transient pressure allowable in the primary coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient presssure allowable in the primary coolant system piping, valves and fittings under ASA Section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings and capacity of the secondary coolant system safety valves (985-1025 psig),[3] the reactor high-pressure trip (2400 psia) and the primary safety valves (2500-2580 psia)[4] have been established to assure never reaching the primary coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psia (125% of design pressure) to verify the integrity of the primary coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves at 2400 psia and the secondary coolant system steam-dump and bypass valves at 900 psia. 16

References

(1) FSAR, Section 4.
(2) FSAR, Section 4.3.
(3) FSAR, Section 4.3.4.
(4) FSAR, Section 4.3.9.

Table 2.3.1

Reactor Protective System Trip Setting Limits

		Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating	Two Primary Coolant Pumps Operating
1.	High Power Level (1)	≤ 106.5% of Rated Power	≤ 45% of Rated Power	≤ 25% of Rated Power
2.	Low Primary Coolant Flow (2)	≥ 95% of Primary Coolant Flow With 4 Pumps Operating	≥ 71% of Primary Coolant Flow With 4 Pumps Operating	≥ 46% of Primary Coolant Flow With 4 Pumps Operating
3.	High Pressurizer Pressure	≤ 2400 Psia	≤ 2400 Psia	≤ 2400 Psia
4.	Thermal Margin/Low Pressure (2)(3)	P_T ≥ Applicable Limits To Satisfy Figure 2-3	Replaced by High-Power Level Trip and 1750 Psia Minimum Low-Pressure Setting	Replaced by High-Power Level Trip and 1750 Psia Minimum Low-Pressure Setting
5.	Low Steam Generator Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level 16	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level 16	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level 16
6.	Low Steam Generator Pressure (2)	≥500 Psia	≥500 Psia	≥500 Psia
7.	Containment High Pressure	≤ 5 Psig	≤ 5 Psig	≤ 5 Psig

(1) Below 5% rated power, the trip setting may be manually reduced by a factor of 10.

(2) May be bypassed below 10^{-4}% of rated power provided auto bypass removal circuitry is operable. For low power physics tests, thermal margin/low pressure and low steam generator pressure trips may be bypassed until their reset points are reached (approximately 1750 psia and 500 psia, respectively), provided automatic bypass removal circuitry at 10^{-1}% rated power is operable.

(3) T_h and T_c in °F. Minimum trip setting shall be 1750 psia for two- and three-pump combinations. For four-pump operation, the minimum trip setting shall be 1650 psia for nominal operating pressures less than 1900 psia; and 1750 psia for nominal operating pressures 1900 psia and greater.

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3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.3 Minimum Conditions for Criticality

a. Except during low-power physics tests, the reactor shall not be made critical if the primary coolant temperature is below 525°F.

b. In no case shall the reactor be made critical if the primary coolant temperature is below NDTT +120°F.

c. When the primary coolant temperature is below the minimum temperature specified in (a) above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

d. No more than one control rod at a time shall be exercised or withdrawn until after a steam bubble and normal water level are established in the pressurizer.

e. Primary coolant boron concentration shall not be reduced until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all control rods withdrawn.(1) However, the uncertainty of the calculation is such that it is possible that a slightly positive coefficient could exist.

The moderator coefficient at lower temperatures will be less negative or more positive than at operating temperature.(1, 2) It is therefore prudent to restrict the operation of the reactor when primary coolant temperatures are less than normal operating temperature ($\geq$525°F).

Assuming the most pessimistic rods out moderator coefficient, the maximum potential reactivity insertion that could result from depressurizing the coolant from 2100 psia to saturation pressure at 525°F is 0.1% $\Delta\rho$. 16

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(3) and the small integrated $\Delta\rho$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

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3.1 PRIMARY COOLANT SYSTEM (Contd)

The 525°F temperature in the specification corresponds to a saturation pressure of 848 psia, which is below the 985 psig minimum 16
set point of the secondary system relief valves. Therefore, potential primary to secondary leakage at temperatures below 525°F 16
could be contained within the steam generator by closing the steam line isolation valve on the defective steam generator.

The 568°F temperature in the specification corresponds to the average temperature of the primary coolant at rated operating conditions. Therefore, measurements of primary coolant radioactivity concentrations made at room temperature will be density corrected to 568°F.

Measurement of $\bar{E}$ will be performed at least twice annually, and in any event will be performed each time the primary coolant radioactivity concentration changes by 10 μCi/cc from the previous measurement of $\bar{E}$. Calculations require to determine $\bar{E}$ will consist of the following:

1. Quantitative measurement in units of μCi/cc of radionuclides with half-lives longer than 30 minutes making up at least 95% of the total activity in the primary coolant.
2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes. 16
3. A calculation of $\bar{E}$ by appropriate weighting of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

References

(1) FSAR, Table II-1.

(2) FSAR, Section II.1.1.

(3) FSAR, Section 14.15.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.7 Primary and Secondary Safety Valves

Specifications

a. The reactor shall not be made critical unless all three pressurizer safety valves are operable with their lift settings maintained between 2500 psia and 2580 psia (±1%). 16

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

c. Whenever the reactor is in power operation, a minimum of 22 secondary system safety valves shall be operable with their lift settings between 985 psig (±10 psig) and 1025 (±1%) psig. 16

Basis

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

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

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

The total relief capacity of the 24 secondary system safety valves is 11.7×10^6 lb/hr. This is based on a steam flow equivalent to an NSSS power level of 2650 MWt at the nominal 1000 psia valve lift pressure. At the initial rated power of 2200 MWt, a relief capacity of only 9.8×10^6 lb/hr is required to prevent overpressurization of the secondary system on loss-of-load conditions and 22 valves provide relieving capability of 10,705,200 lb/hr.(1)

The ASME Boiler and Pressure Vessel Code, Section III, 1971 edition, paragraph NC-7614.2(a) allows the specified tolerances in the lift pressures of safety valves. 16

Reference

(1) FSAR, Sections 4.3.4, 4.3.7 and 14.12.4.

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Table 3.16.1

Engineered Safety Features System Initiation Instrument Setting Limits

Functional Unit	Channel	Setting Limit
1. High Containment Pressure	a. Safety Injection b. Containment Spray c. Containment Isolation d. Containment Air Cooler DBA Mode	5-5.75 Psig 16
2. Pressurizer Low Pressure	Safety Injection	≥1550 Psia[1] for Nominal Operating Pressures <1900 Psia ≥1593 Psia[2] for Nominal Operating Pressures ≥1900 Psia
3. Containment High Radiation	Containment Isolation	≤20 R/Hr
4. Low Steam Generator Pressure	Steam Line Isolation	≥500 Psia[3]
5. SIRW Low-Level Switches	Recirculation Actuation	≤27-Inch (+0 (-6 Above Tank Bottom
6. Rod Limit Switches (LS-6)	Turbine Cutback	≤5 Inches
7. Power Range Nuclear Instr	Turbine Cutback	a. Time Delay 8 Sec (+1 (-1 b. ≤8% Power
8. Turbine Valve Position Switches	Turbine Cutback	≤70% Rated Power
9. Engineered Safeguards Pump Room Vent - Radiation Monitors	Engineered Safeguards Pump Room Isolation	$\le 2.2 \times 10^5$ Cpm

(1) May be bypassed below 1600 psia and is automatically reinstated above 1600 psia.
(2) May be bypassed below 1700 psia and is automatically reinstated above 1700 psia.
(3) May be bypassed below 550 psia and is automatically reinstated above 550 psia."

4.1 INSTRUMENTATION AND CONTROL (Contd)

Thus, minimum calibration frequencies of once-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. 16

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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Table 4.1.1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM(5) 16

Channel Description	Surveillance Function	Frequency	Surveillance Method	
1. Power Range Safety Channels	a. Check	S	a. Comparison of four power channel readings	
	b. Calibrate(3)	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparator alarms.	
	c. Test	M(2)	c. Internal test signal.	16
2. Wide-range Logarithmic Neutron Monitors	a. Check	S	a. Comparison of both wide-range readings.	
	b. Test	P	b. Internal test signal.	16
3. Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.	
	b. Calibrate	R	b. Known differential pressure applied to sensors.	16
	c. Test	M(2)	c. Bistable trip tester. (1)(4)	16
4. Thermal Margin/Low Pressurizer Pressure	a. Check: (1) Temperature Input	S	a. Check: (1) Comparison of four separate calculated trip pressure setpoint indications.	
	(2) Pressure Input		(2) Comparison of four pressurizer pressure indications. (Same as 5(a) below).	

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Table 4.1.1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. (Contd)	b. Calibrate (1) Temperature Input	R	b. Calibrate: (1) Known resistance substituted for RTD coincident with known pressure input.
	(2) Pressure Input		(2) Part of 5(b), below. 16
	c. Test	M[2]	c. Bistable trip tester.[1]
5. High-pressurizer Pressure	a. Check	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M[2]	c. Bistable trip tester.[1]
6. Steam Generator Level	a. Check	S	a. Comparison of four level indications per generator.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M[2]	c. Bistable trip tester.[1]
7. Steam Generator Pressure	a. Check	S	a. Comparisons of four pressure indications per generator.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M[2]	c. Bistable trip tester.[1]
8. Containment Pressure	a. Calibrate	R	a. Known pressure applied to sensors.
	b. Test	M[2]	b. Simulate pressure switch action.
9. Loss of Load	a. Test	P	a. Manually trip turbine auto stop oil relays.

Table 4.1.1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM (contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
10. Manual Trips	a. Test	P	a. Manually test both circuits.
11. Reactor Protection System Logic Units	a. Test	M(2)	a. Internal test circuits.

Notes: (1) The bistable trip tester injects a signal into the bistable and provides a precision readout of the trip set point.

(2) All monthly tests will be done on only one of four channels at a time to prevent reactor trip.

(3) Calibrate using built-in simulated signals.

(4) Trip setting for operating pump combination only. Settings for other than operating pump combinations must be tested during routine monthly testing performed when shut down and within four hours after resuming operation with a different pump combination if the setting for that combination has not been tested within the previous month. 16

(5) It is not necessary to perform the specified testing during prolonged periods in the refueling shutdown condition. If this occurs, omitted testing will be performed prior to returning the plant to service.

S - Each Shift
D - Daily
M - Monthly
R - Each Refueling Shutdown, But Not To Exceed 16 Months
P - Prior to Each Start-Up if Not Done Previous Week

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Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method	
1. Low-pressure SIS Initiation Channels	a. Check	S	a. Comparison of four separate pressure indications.	
	b. Test(1)	R	b. Signal to meter relay adjusted with test device to verify SIS actuation logic.	16
	c. Test	M(2)	c. Signal to meter relay adjusted with test device.	
2. Low-pressure SIS Signal Block Permissive and Auto Reset	a. Test(1)	R	a. Part of 1(b) above.	16
3. SIS Actuation Relays	a. Test	Q	a. Simulation of SIS 2/4 logic trip using built-in testing system. Both "standby power" and "no standby power" circuits will be tested for left and right channels. Test will verify functioning of initiation circuits of all equipment normally operated by SIS signals.	
	b. Test	R	b. Complete automatic test initiated by same method as Item 1(b) and including all normal automatic operations.	16
4. Containment High-pressure Channels	a. Calibrate	R	a. Known pressure applied to sensors.	16
	b. Test	R	b. Simulation of CHP 2/4 logic trip to verify actuation logic for SIS, containment isolation, and containment spray.	16

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Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. (Contd)	c. Test	$M^{(2)}$	b. Pressure switch operation simulated by opening or shorting terminals. 16
5. Containment High Radiation Channels	a. Check	D	a. Comparison of four separate radiation level indications.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	$M^{(2)}$	c. Remote operated integral radiation check source used to verify instrument operation. 16
	d. Test	R	d. Simulation of CHR 2/4 logic trip with test switch to verify actuation relays, including containment isolation.
6. Manual SIS Initiation	a. Test	R	a. Manual push-button test.
7. Manual Containment Isolation Initiation	a. Test	R	a. Manual push-button test.
	b. Check	R	b. Observe isolation valves closure.
8. Manual Initiation Containment Spray Pumps and Valves	a. Test	R	a. Manual switch operation.
9. DBA Sequencers	a. Test	Q	a. Proper operation will be verified during SIS actuation test of Item 3(a) above.

Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method	
10. Normal Shutdown Sequencers	a. Test	R	a. Simulate normal actuation with test-operate switch and verify equipment starting circuits.	
11. Diesel Start	a. Test	M	a. Manual initiation followed by synchronizing and loading.	16
	b. Test	R	b. Diesel start, load shed, synchronizing, and loading will be verified during Item 3(b) above.	
	c. Test	P	c. Diesel auto start initiating circuits.	
12. SIRW Tank Level Switch Interlocks	a. Test	R	a. Level switches removed from fluid to verify actuation logic.	16
	b. Test	Q	b. Use SIRW tank control switch to verify actuation of valves.	16
13. Safety Injection Tank Level and Pressure Instruments	a. Check	S	a. Verify that level and pressure indication is between independent high high/low alarms for level and pressure.	
	b. Calibrate	R	b. Known pressure and differential pressure applied to pressure and level sensors.	
14. Boric Acid Tank Level Switches	a. Test	R	a. Pump tank below low-level alarm point to verify switch operation.	

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Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Cont'd)

Channel Description	Surveillance Function	Frequency	Surveillance Method	
15. Boric Acid Heat Tracing System	a. Check	D	a. Observe temperature recorders for proper readings	
16. Main Steam Isolation Valve Circuits	a. Check	S	a. Compare four independent pressure indications.	
	b. Test (3)	R	b. Signal to meter relay adjusted with test device to verify MSIV circuit logic.	
17. SIRW Tank Temperature Indication and Alarms	a. Check	M	a. Compare independent temperature readouts.	
	b. Calibrate	R	b. Known resistance applied to indicating loop.	16

Notes: (1) Calibration of the sensors is performed during calibration of Item 5(b), Table 4.1.1.

(2) All monthly tests will be done on only one channel at a time to prevent protection system actuation. 16

(3) Calibration of the sensors is performed during calibration of Item 7(b), Table 4.1.1.

S - Each Shift
D - Daily
M - Monthly
Q - Quarterly
R - Each Refueling Shutdown, But Not to Exceed 16 Months
P - Prior to Each Start-up if Not Done Previous Week
SA - Semiannually 16

FEB 11 1975

Table 4.1.3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Startup Range Neutron Monitors	a. Check	S	a. Comparison of both channel countrate indications when in service.
	b. Test	P	b. Internal test signals. 16
2. Primary Rod Position Indication System	a. Check	S	a. Comparison of output data with secondary RPIS.
	b. Check	M	b. Check of power dependent insertion limits monitoring system.
	c. Calibrate	R	c. Physically measured rod drive position used to verify system accuracy. Check rod position interlocks.
3. Secondary Rod Position Indication System	a. Check	S	a. Comparison of output data with primary RPIS.
	b. Check	M	b. Same as 2 (b) above.
	c. Calibrate	R	c. Same as 2 (c) above, including out of sequence alarm function.
4. Area and Process Monitors	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	M	c. Detector exposed to remote operated radiation check source.

Table 4.1.3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
5. Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
	b. Test	M	b. Battery Check.
6. Environmental Monitors	a. Check	M	a. Operational check.
	b. Calibrate	A	b. Verify airflow indicator.
7. Pressurizer Level Instruments	a. Check	S	a. Comparison of six independent level readings.
	b. Calibrate	R	b. Known differential pressure applied to sensor.
	c. Test	M	c. Signal to meter relay adjusted with test device. 16
8. Control Rod Drive System Interlocks	a. Test	R	a. Verify proper operation of all rod drive control system interlocks, using simulated signals where necessary.
	b. Test	P	b. Same as 8(a) above, if not done within three months. 16
9. Turbine Runback	a. Test	M	a. Check combination nuclear instrumentation and rod drive control system signal with test circuit.
	b. Test	R	b. Insert rod drives below lower electrical limit to verify runback signal initiation.

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Table 4.1.3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method	
10. Flux-ΔT power Comparator	a. Calibrate	R	a. Use simulated signals.	
	b. Test	M	b. Internal test signal.	16
11. Calorimetric Instrumentation	a. Calibrate	R	a. Known differential pressure applied to feedwater flow sensors.	16
12. Containment Building Humidity Dectors	a. Test	R	a. Expose sensor to high humidity atmosphere.	6
13. Interlocks - Isolation Valves on Shutdown Cooling Line	a. Calibrate	R	a. Known pressure applied to sensor.	16
14. Service Water Break Detector in Containment	a. Test	R	a. Known differential pressure applied to sensors.	16
15. Control Room Ventilation	a. Test	R	a. Check damper operation for DBA mode with HS1801 and isolation signal.	
	b. Test	R	b. Check control room for positive pressure.	

S - Each Shift
D - Daily
M - Monthly
A - Annually
R - Each Refueling Shutdown, But Not To Exceed 16 Months
P - Prior to Each Startup, if Not Done Previous Week

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4.2 EQUIPMENT AND SAMPLING TESTS (Contd)

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the purge system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA." 16

The post-incident recirculation systems provide adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Proper hydrogen recombiner operation, after a LOCA, is assured by measuring (and adjusting, if necessary) the amount of electrical power provided to the recombiner unit. The temperature measuring equipment (thermocouple) is provided for convenience in testing and is not considered necessary to assure proper operation.

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Table 4.2.1

Minimum Frequencies for Sampling Tests

		Test	Frequency	FSAR Section Reference	
1.	Reactor Coolant Samples	Quantitative gamma spectral analysis or gross beta gamma radioactivity analysis by internal proportional counter and qualitative gamma spectral analysis.	3 Times/Week(1)	None	16
		Tritium Radioactivity	Weekly	None	
		Chemistry (Cl and O_2)	3 Times/Week	None	
		Radiochemical Analysis for $\bar{E}$ Determination	Semiannual(2)	None	
2.	Reactor Coolant Boron	Boron Concentration	Twice/Week	None	
3.	SIRW Tank Water Sample	Boron Concentration	Monthly	None	
4.	Concentrated Boric Acid Tanks	Boron Concentration	Monthly	None	
5.	SI Tanks	Boron Concentration	Monthly	6.1.2	
6.	Spent Fuel Pool	Boron Concentration	Monthly	9.4	
7.	Secondary Coolant	Iodine Concentration	Weekly(3)	None	
8.	Liquid Radwaste	Radioactivity Analysis	Prior to Release of Each Batch	11.1	
9.	Radioactive Gas Decay Tanks	Radioactivity Analysis	Prior to Release of Each Batch	11.1	
10.	Stack-Gas Monitor Particulate Samples	Iodine 131 and Particulate Radioactivity	Weekly(4)	11.1	

(1) When radioactivity level exceeds 10 percent of limits in Specification 3.1.4, or 3.1.5, the sampling frequency shall be increased to a minimum of once each day.

(2) Redetermine if: (a) the primary coolant radioactivity increases by more than 10 μCi/cc from the previous determination, and (b) upon each start-up only after a two-week equilibrium adjustment period shows a 10 μCi/cc increase from the previous determination in accordance with Specification 3.1.4.

(3) When radioactivity level exceeds 10 percent of limits in Specification 3.1.5, the sampling frequency shall be increased to a minimum of once each day.

(4) When iodine or particulate radioactivity levels exceed 10 percent of limit in Specification 3.9.6 and 3.9.9, the sampling frequency shall be increased to a minimum of once each day.

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Table 4.2.2

Minimum Frequencies for Equipment Tests

		Test	Frequency	FSAR Section Reference
1.	Control Rods	Drop Times of All Full-Length Rods	Each Refueling Shutdown	7.4.1.3
2.	Control Rods	Partial Movement of All Rods (Minimum of 6 In)	Every Two Weeks	7.4.1.3
3.	Pressurizer Safety Valves	Set Point	One Each Refueling Shutdown	7.3.7
4.	Main Steam Safety Valves	Set Point	Five Each Refueling Shutdown	4.3.4
5.	Refueling System Interlocks	Functioning	Prior to Refueling Operations	9.11.3
6.	Service Water System Valve Actuation (SIS-CHP)	Functioning	Each Refueling Operation	9.1.2
7.	Fire Protection Pumps and Power Supply	Functioning	Monthly	9.6.2
8.	Primary System Leakage	Evaluate	Daily	4 Amend 15, Ques 4.3.7
9.	Diesel Fuel Supply	Fuel Inventory	Daily	8.4.1
10.	Critical Headers Service Water System	150 Psig Hydrostatic Test	Every Five Years	9.1.2
11.	Charcoal & Hi Efficiency Filters for Control Room Fuel Storage Building and Containment Purge System (containment post-accident filter).	Charcoal filters checked ≥ 99% efficiency per Freon 112 test (ORNL). HEPA filters checked ≥ 99% efficiency per ANSI N1J1.1-1972	Each Refueling Shutdown and at any time work on filters could alter their integrity.	Amend 14, Ques 14.19-1 6.5.1 9.8.3

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Table 4.2.2 (continued)

Minimum Frequencies for Equipment Tests

12. Hydrogen Recombiners

Each hydrogen recombiner unit shall be demonstrated operable:

a. At least once per 6 months by verifying during a recombiner unit functional test that the minimum heater sheath temperature increases to $\geq$ 700°F* within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

1. Verifying that each of the electrical busses providing recombiner unit power is aligned to receive power from separate diesel generators.

2. Performing a channel calibration of all recombiner instrumentation and control circuits.

3. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.) 16

4. Verifying during a recombiner unit functional test that the heater sheath temperature increases to $\geq$1200°F* within 180 minutes and that the system operates for a least 4 hours.

5. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test immediately following the above required functional test. The resistance to ground for any heater element shall be $\geq$ 1000 ohms.

*As measured by installed or portable temperature measuring instruments.

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4.5 CONTAINMENT TESTS

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the prestressing tendon loads are maintained within specified values.

Specifications

4.5.1 Integrated Leakage Rate Tests

a. Test

(1) Integrated leak rate tests shall be performed prior to initial
plant operations at containment design pressure (P_a) of 55 psig 16
and a test pressure (P_t) of at least 28 psig to establish the
respective measured leak rates, L_{am} and L_{tm}. A minimum test 16
temperature of 50°F will be utilized. The maximum test temperature will be 100°F.

(2) Subsequent leak rate tests shall be performed at the test pressure of about 28 psig. The tests shall be performed without any leak detection surveys or leak repairs immediately prior to or during the test, except as noted below.

(3) Major leak repairs, if necessary to permit the integrated leak rate test, shall be preceded by local leakage measurements. The local leakage differences, as a result of repair, shall be corrected to P_t and added to the final integrated leak rate test result to determine the subsequent retest schedule.

(4) All systems which, under accident conditions, become an extension of the containment shall be vented to the containment atmosphere during integrated leak rate tests. Closure of containment isolation valves is to be accomplished by the normal mode of actuation.

(5) The test duration shall not be less than 24 hours unless test experiences of at least two prior tests provide evidence of the adequacy of shorter test duration. Test accuracy shall be verified by supplementary means, such as measuring the quantity of air required to return to the starting point, or by imposing a known leak rate to demonstrate validity of measurements.

4.5 CONTAINMENT TESTS (Contd)

b. Acceptance Criteria

(1) The maximum allowable leakage rate under DBA conditions, L_a, shall not exceed 0.10 weight percent per 24 hours.

(2) The allowable operational leakage rate, L_{to}, which shall be met prior to resumption of power operation following a test shall not exceed $0.75L_t$ where L_t is defined by 4.5.1.b(3).

(3) The allowable leakage rate, L_t, at the reduced test pressure,
shall not exceed L_a (L_{tm}/L_{am}) unless L_{tm}/L_{am} exceeds 0.7, in
which case L_t shall be equal to L_a $(P_t/P_a)^{1/2}$. The subscript 16
m refers to values of the leakage measured during preoperational
tests. The subscripts a and t refer to tests at calculated
accident pressure and reduced pressure respectively.

c. Corrective Action for Retests

(1) If repairs are necessary to meet the acceptance criterion, the integrated leak rate test need not be repeated provided local leakage rate measurements are made and the leakage rate difference achieved by repairs reduces the overall measured integrated leak rate to a value not in excess of the allowable operational leakage rate, L_{to}.

(2) The reduction in leakage effected by the repair of isolation valves shall be included in the integrated leak rate test results.

d. Frequency

16

(1) After the initial preoperational leakage rate tests, a set
of three integrated leak rate tests shall be performed at
approximately equal intervals during each 10-year service
period. The third test of each set shall be conducted
when the plant is shut down for the 10-year inservice 16
inspections.

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4.5 CONTAINMENT TESTS (Contd)

(2) If any periodic integrated leak rate test fails to meet the acceptance criteria, the test schedule applicable to subsequent integrated leak rate tests will be reviewed and approved by the Commission. 16

e. Report of Test Results

Each integrated leak rate test will be the subject of a summary technical report which will include summaries of local leak detection tests and leak test of the recirculation heat removal systems.

4.5.2 Local Leak Detection Tests

a. Test

(1) Local leak rate tests shall be performed at a pressure of not less than 55 psig.

(2) Acceptable methods of testing are halogen gas detection, soap bubble, pressure decay, or equivalent.

(3) The local leak rate shall be measured for each of the following components:

(a) Containment penetrations that employ resilient seal gaskets, sealant compounds, or bellows.

(b) Air lock and equipment door seals.

(c) Fuel transfer tube.

(d) Isolation valves on the testable fluid systems' lines penetrating the containment.

(e) Other containment components which require leak repair in order to meet the acceptance criterion for any integrated leak rate test.

b. Acceptance Criterion

The total leakage from all penetrations and isolation valves shall not exceed $0.60L_a$. 16

c. Corrective Action

(1) If at any time it is determined that $0.60L_a$ is exceeded, repairs shall be initiated immediately. 16

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4.5 CONTAINMENT TESTS (Contd)

The test pressure of 28 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the preoperational leak rate test at 28 psig. The specification provides relationships for relating in a conservative manner the measured leakage of air at 28 psig to the potential leakage of a steam-air mixture at 55 psig and 283°F. The specification also allows for possible deterioration of the leakage rate between tests by requiring that only 75% of the allowable leakage rates actually be measured. The basis for these deterioration allowances is 10 CFR Part 50, Appendix J which is believed to be conservative and will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

The duration of 24 hours for the integrated leak rate test is established to provide a minimum level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

The frequency of the periodic integrated leak rate test is keyed to the refueling schedule for the reactor because these tests can best be performed during refueling shutdowns. The initial refueling will occur about 36 months after initial criticality was achieved. Subsequent refueling shutdowns are expected to occur at approximately 12-month intervals. The specified frequency is as specified in 10 CFR Part 50, Appendix J which is based on three major considerations. First is the low probability of leaks in the liner because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which is consistent with 0.1% leakage at design basis accident (DBA) conditions; and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value ($0.60L_a$) of the total leakage that is specified as acceptable from penetrations and isolation valves. 16 Third is the tendon stress surveillance program which provides assurance that

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

The basis for specification of a total leakage rate of $0.60L_a$ from penetrations and isolation valves is specified to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. This value allows for possible deterioration in the intervals between tests. The limiting leakage rates from the shutdown cooling system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a DBA. The test pressure (270 psig) achieved either by normal system operation or by hydrostatically testing gives an adequate margin over the highest pressure within the system after a DBA. Similarly, the hydrostatic test pressure for the return lines from the containment to the shutdown cooling system (100 psig) gives an adequate margin over the highest pressure within the lines after a DBA.[5] 16

A shutdown cooling system leakage of 1/2 gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the DBA. The engineered safeguards room ventilation system is equipped with isolation valves which close upon a high radiation signal from a local radiation detector. These monitors shall be set at 2.2×10^5 cpm, which is well below the expected level, following a loss-of-coolant accident (LOCA), even without clad failure. The 1/2-gpm leak rate is sufficiently high to permit prompt detection and to allow for reasonable leakage through the pump seals and valve packings, and yet small enough to be readily handled by the sumps and radioactive waste system. Leakage to the engineered safeguards room sumps will be returned to the containment clean water receiver following an LOCA, via the equipment drain tank and pumps. Additional makeup 16

4.7 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specifications

4.7.1 Diesel Generators

a. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 10 seconds. The signal initiated to start the diesel shall be varied from one test to another to verify all starting circuits are operable. The generation shall be synchronized from the control room, paralleled and loaded to the nameplate rating.

b. A test shall be conducted during each refueling outage to demonstrate the overall automatic operation of the emergency power system. The test shall be initiated by a simulated simultaneous loss of normal and standby power sources and a simulated SIS signal. Proper operations shall be verified by bus load shedding and automatic starting of selected motors and equipment to establish that restoration with emergency power has been accomplished within 30 seconds.

c. Each diesel generator shall be subjected to an inspection, in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service, at least once per 18 months during plant shutdown. The licensee shall utilize his best efforts to conduct additional major diesel generator inspections and overhauls during shutdown periods. 16

d. Diesel generator electric loads shall not be increased beyond the continuous rating of 2500 kW.

e. The fuel transfer pumps shall be verified to be operable each month.

4.7.2 Station Batteries

a. Every month, the voltage of each cell (to the nearest 0.01 volt), the specific gravity and the temperature of a pilot cell in each battery shall be measured and recorded.

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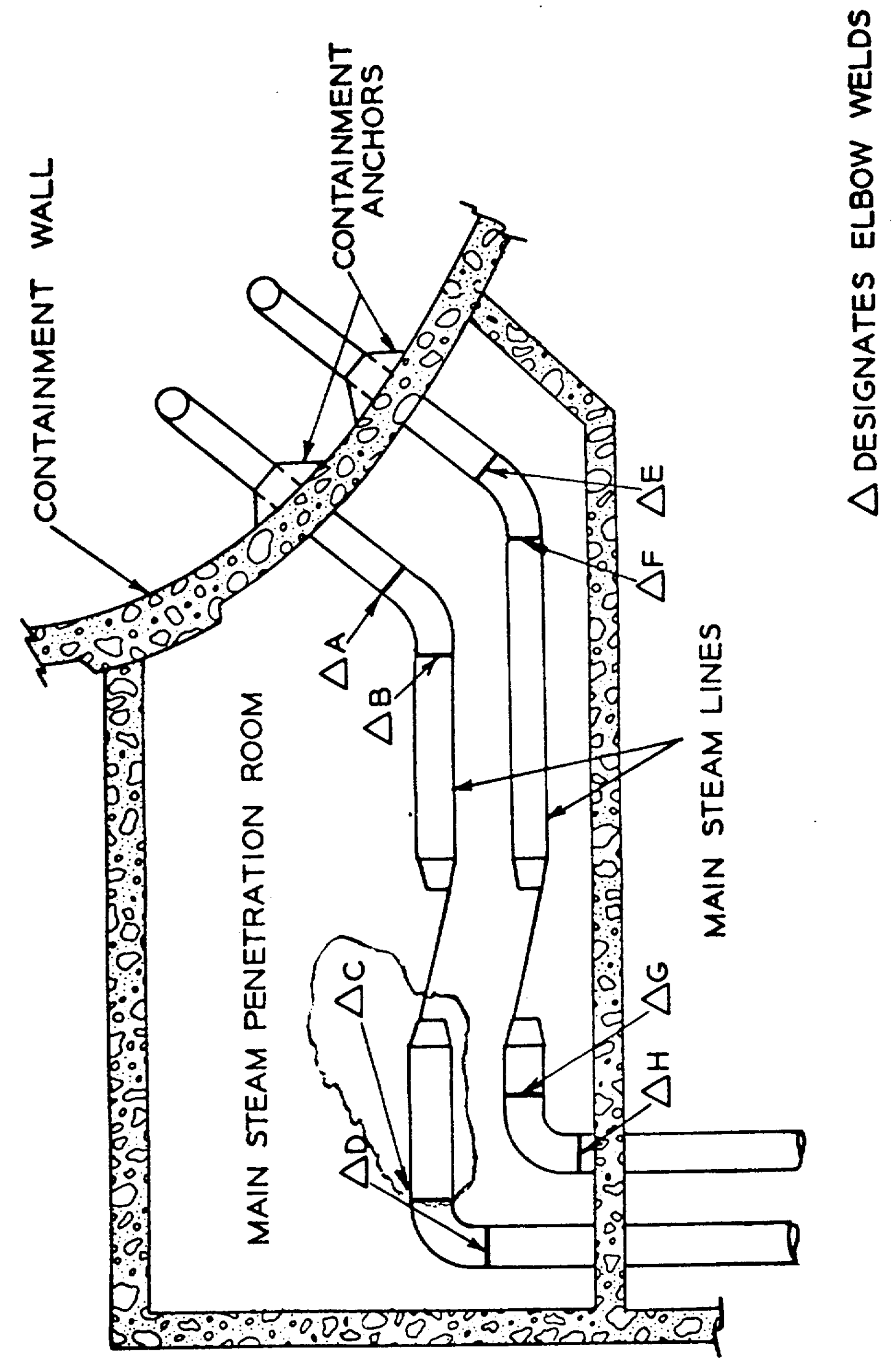

AUGMENTED INSERVICE INSPECTION PROGRAM – MAIN STEAM WELDS	PALISADES TECHNICAL SPECIFICATIONS	FIGURE 4.12 A Change No. 16

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TABLE 1

Page	Section	Summary of Proposed Change
2-3	2.2	Corrects error in settings of secondary coolant system safety valves - psia to psig.
2-5	Table 2.3.1	Changes Low Level Steam Generator Water Level trip setting language to read "not lower than the centerline of feedwater ring", as opposed to "at the centerline" of the feedwater ring.
3-15	3.1.3 (Basis)	Corrects error in language with respect to potential reactivity insertion due to depressurization of the reactor coolant system. The present Basis states "... the maximum potential reactivity insertion that could result from depressurizing the coolant from 525°F to saturation temperature at 2100 psia is 0.1%$\Delta\rho$." The proposed change reverses the position of the temperature and pressure words, and results in a correct statement: "... the maximum potential reactivity insertion that could result from depressurizing the coolant from 2100 psia to saturation pressure at 525°F is 0.1%$\Delta\rho$."
3-19	3.1.4 (Basis)	Changes "psia" to "psig" to correct error. Add "F" to "525°".
4-3	Table 4.1.1 Items 1.c. 2.b. 3.b.	Changes the language of Surveillance Method to specify the surveillance method only. The items to be verified are defined in Section 1.3, Instrumentation Surveillance, and need not be repeated here.
4-4	Table 4.1.1 Item 4.b.(2)	Changes the language of the Surveillance Method for the pressure input to the thermal margin/pressurizer pressure channel to clarify that this surveillance is part of the surveillance for the high pressurizer pressure channel.
4-6 4-7 4-9	Table 4.1.2 Items 1.b. 1.c. 2.a. 3.b. 4.c. 5.c. 16.b. 17.b.	Adds three new footnotes: to indicate and clarify that the low-pressure SIS initiation channels and the low-pressure SIS block permissive channels are calibrated in Table 4.1.1, Item 5(b); to indicate in footnote format that monthly tests of the low-pressure SIS initiation channels, the containment high pressure channels, and containment high radiation channels are tested one channel at a time to prevent protection system actuation, instead of stating this under Surveillance Method for these channels; and to indicate and clarify that calibration of the main steam isolation valve circuits is governed by Table 4.1.1, Item 7(b). Other changes clarify the relationship between

Page	Section	Change
		the trip channels for low pressure SIS initiation, SIS actuation relays and the main steam isolation circuits, and their associated sensor channels. The changes add clarity to the test table, and break the testing into more manageable units. In addition, the changes utilize the definition of Instrumentation Surveillance contained in Section 1.3, and therefore eliminate repeating the functions that are verified under Surveillance Method for individual channels.
4-10 4-11 4-12	Table 4.1.3 Items 1.b. 7.c. 14.a.	Changes the Surveillance Method to delete material contained in the definition of Instrumentation Surveillance, Section 1.3.
4-11	Table 4.1.3 Item 8.b.	Clarifies the language. Since this surveillance is required prior to each startup, the phrase "and plant is shutdown" is not needed.
4-12	Table 4.1.3 Items 10.b. 11.a.	Makes minor word changes.
4-12	Table 4.1.3 Item 12.a.	Permits in-place testing of the containment humidity detectors.
4-63	Figure 4.12A	Corrects drafting error in figure.

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

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 12 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

(CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS)

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

Introduction

By letter dated April 8, 1974, Consumers Power Company requested changes to the Technical Specifications appended to Provisional Operating License No. DPR-20 for the Palisades Plant. The proposed changes involve numerous editorial corrections to clarify the intent of several specifications and to correct typographical errors; a modified specification for reactor coolant flow testing; revised acceptance criteria for containment periodic testing; revised diesel generator surveillance; addition of allowable tolerances for safety valve settings; modified procedures for testing reactor coolant samples and air filters; new surveillance requirements for the hydrogen recombiners; modified effluent release specifications for low level gaseous waste; a revised setting for initiation of engineered safeguards on high containment pressure; elimination of surveillance requirements for the Reactor Protection System during prolonged refueling shutdown periods; modifications to the surveillance requirements for the containment high pressure channels; a revised method for testing the Safety Injection Refueling Water Tank Level Switch interlock; a different surveillance test for the interlock on the shutdown cooling system isolation valves; and modifications to the administrative controls including reporting requirements.

Discussion

The proposed changes represent a multiplicity of changes that, according to the licensee, have been shown to be necessary by the operating experience they have gained by several years' usage of the Technical Specifications. The proposed changes involving potential safety considerations are discussed individually below.

Evaluation

1. The proposed change to the Basis of Section 3.1.4 (Maximum Primary Coolant Activity) would delete the specific parenthetic reference to the Table of Isotopes, Sixth Edition, March 1968, with respect to the determination of beta and gamma disintegration energies of nuclides in the primary coolant. The proposed change involves the Basis for Section 3.1.4 and is not a Technical Specification. Nevertheless, the deletion of this reference in the Basis would improve clarity because it implies that this is the only suitable source for

this information, which was not intended. When the Tech Specs were initially issued in November 1971, this was the most complete source of decay energies and schemes. With the passage of time, this document will become progressively outdated. What was, and is, intended is that the licensee utilize the best available sources, including the Table of Isotopes where appropriate. In addition, there is no specific reference to this document (or any other) for such determinations in Tech Specs currently being issued. We agree that the proposed revision would improve clarity, and should be made.

2. The proposed change to Section 3.1.7 would add a ± 1% tolerance to the lift settings of (a) the primary safety valves and (b) secondary safety valves with a lift setting greater than 1000 psig, and a ± 10 psi tolerance to those secondary safety valves with lift settings below 1000 psig. These tolerances are permitted by the ASME Boiler and Pressure Vessel Code, Section III, 1971 edition, para. NC-7614.2(a), and recognize the practical impossibility of achieving exact lift settings for such valves. The nominal lift settings remain unchanged, and thus the proposal represents no change in a limiting condition for operation except for the code-allowable tolerances cited above. Operation in the proposed manner will not result in any adverse effects since these tolerances are small, and the loss-of-load incident analysis (FSAR, Section 14.12) demonstrates that there is adequate margin between the maximum allowable pressure for the reactor coolant system and steam generators and the transient pressure resulting from a loss-of-load. The addition of these tolerances will not significantly alter these results. This change is acceptable.

3. The requested changes to Section 3.9, Effluent Release, involving modified effluent release specifications for low-level gaseous waste, have been superseded by Amendment No. 6 (issued August 30, 1974) which transmitted Change No. 10 to the Tech Specs. Accordingly, no action is necessary on this item.

4. The change proposed to the high containment pressure setting for safeguards initiation (Table 3.16.1) eliminates the ambiguity of the present specification (≤5 ± $\overset{.75}{.25}$ psig), and assures that reactor trip on high containment pressure (≤5 psig) will occur at or prior to safeguards initiation on high containment pressure. As proposed, the setting for high containment pressure for safeguards initiation is 5.0 - 5.75 psig which represents no change from the upper limit previously approved, and eliminates the overlap at the lower limit, which could result in safeguards initiation prior to reactor scram, an undesirable sequence. This change is acceptable.

5. Changes are proposed to Table 4.1.1 (Minimum Frequencies for Checks, Calibrations, and Testing of the Reactor Protective System.)

   (a) A new "Note 5" is proposed which eliminates the surveillance requirements during prolonged periods in the refueling shutdown condition but provides for performing any omitted surveillance prior to returning the plant to service. The Reactor Protection System (RPS) is not required when in the refueling shutdown condition. The Startup Range Neutron Monitors are not part of the RPS and are required to be operable below 10^{-4}% of rated power (Table 3.17.4); surveillance of these channels in accordance with Table 4.1.3 is continued in the refueling mode. It was never the intention of the original Tech Specs to require surveillance testing of the RPS during prolonged periods in the refueling shutdown condition. Since any omitted surveillance to demonstrate operability will be performed prior to the need for the RPS, the proposed change is acceptable , and is consistent with Tech Specs currently being issued.

   (b) A new "Note 4" is proposed with respect to testing of the reactor coolant flow circuit to specify more clearly the extent of the monthly test. The present specification requires a monthly test of the four reactor coolant flow channels one at a time. The trip settings are verified by a bistable test device. The channel trip settings, however, are dependent upon the position of a manual pump selector switch. The manual selector switch must be set for the operating pump combination (2, 3, or 4 pumps), and the trip settings are verified for that combination. The existing surveillance requirement does not explicitly state whether <u>all</u> trip settings should be verified, or just the trip setting for the operating combination. The proposed Note 4 states that it is not necessary to verify the low flow trip settings for pump combinations other than the operating combination, since to do so would require changing the manual switch which could cause a reactor trip on high flux or low flow depending on operating conditions. In view of the fact that the licensee will conduct (a) the monthly test for all pump combinations if shutdown and (b) the test for a new pump combination within four hours if the setting has not been tested within the previous month, we conclude that the proposed change is acceptable.

6. Changes are proposed to Table 4.1.2 (Minimum Frequencies for Checks, Calibrations, and Testing of Engineered Safety Feature Instrumentation Controls).

   (a) The proposed change to the containment high-pressure channel surveillance is intended to be editorial in nature, and consists of the deletion of words that the licensee states are already included in the definition of "channel calibration" in Section 1.3. The present surveillance method states: "known pressure applied to sensors and CHP actuation logic for SIS, containment isolation and containment spray verified." The proposed change would state: "known pressure applied to sensors."

Following discussions with the licensee, he has agreed that the definition of "channel calibration" does not include logic tests described above. This conclusion is based on IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Generating Stations", which states in Section 2. (Definitions) that a channel loses its identity where single action signals are combined. In this context, then, channel calibration does not include logic tests which combine outputs from several channels.

Based on the above, specific reference to logic testing should be retained on a refueling interval frequency, consistent with such testing for other channels which initiate safety injection, containment spray, and containment isolation. The proposed change, as modified, is acceptable.

(b) The proposal to delete the requirement for quarterly starting of the diesels during the quarterly test of the SIS actuation relays is acceptable because the automatic diesel start is not initiated by these relays. The diesels are tested monthly by manual initiation, which includes the automatic starting circuits on a rotating basis (Tech Spec 4.7.1), as well as prior to plant start-up. This results in all diesel start circuits being tested quarterly without this quarterly specification.

(c) A change to the method of testing the Safety Injection Refueling Water Tank Level Switch Interlock is requested (Table 4.1.2, Item 12). The present test method requires that the tank level switches be removed to simulate low tank liquid level and to verify actuation of the associated valves. Since the actuation logic involves coincidence of 2/4 level switches, this involves excessive valve cycling to verify all logic paths. The licensee proposes to test the circuit in two stages: (1) remove the level switches in order to verify actuation logic, and (2) use the tank test switch to verify valve actuation.

This change is acceptable because the tank test switch is in series with the level switch and the new procedure is therefore equivalent to the present method. Repeated valve cycling is not necessary to demonstrate correct performance.

(d) The licensee proposes surveillance requirements for the electrically-operated hydrogen recombiner units added during the 1974 shutdown. These consist of operability checks semi-annually and verification of proper heatup on a refueling outage frequency. While these proposed surveillance intervals are generally consistent with those specified in Tech Specs currently being issued, the proposed acceptance criteria are not consistent with those of plants with identical equipment, and lack specificity. Therefore, these proposed surveillance requirements should be modified to add the necessary requirements and to be consistent with the regulatory position established by Tech Specs being currently issued for newer plants. With these modifications, the proposed surveillance requirements for the hydrogen recombiners are acceptable.

7. The proposed change to Table 4.1.3 (Item 13) concerning the interlock for the isolation valves on the shutdown cooling system would revise the surveillance method to reflect a design change made to this system. Previously, the interlock consisted of a torque switch on the valve motor operator to prevent valve operation when high differential pressure across the disc was present. The surveillance method for this design consisted of applying a differential pressure of 300 psi across the disc and adjusting the torque switch to actuate under this condition. The design change replaced the torque switch on the motor operator with a pressure switch on the piping to sense high pressure and prevent motor operation. Therefore, the licensee proposes a revised surveillance method of applying known pressure to the pressure switch to verify proper interlocking function, which is acceptable.

8. The licensee's proposal to conduct quantitative gamma spectral analysis of reactor coolant samples (Table 4.2.1, Minimum Frequencies for Sampling Tests) will provide a more complete analysis of individual isotope content and reactor coolant radioactivity levels and is acceptable. This added test method will allow the licensee a choice of methods - the one above or gross beta-gamma counting combined with qualitative gamma spectral analysis.

9. Table 4.2.2 (Minimum Frequencies for Equipment Tests) presently requires an in-place DOP test for HEPA filters with an acceptance criterion of greater than 99% removal efficiency for 0.3 micron particles. These filters must meet this acceptance criterion when purchased, prior to installation.

Demonstrating this efficiency for 0.3 micron particles requires laboratory-type equipment. The test is performed by the manufacturer or laboratory under conditions which cannot be practically achieved using field testing techniques. The intent of the field in-place test in this specification is to verify proper installation. This requires verification that the filters have not been damaged by the installation process or time (no tears or rips), and that they have been properly installed (minimal bypass flow). Thus, the field test is an installation verification, rather than a filter unit performance test.

The licensee has stated, and we concur, that the particle size of 0.3 micron cannot be practically achieved with field testing techniques and requests a change to allow the test to be conducted with a mean particle size of approximately 0.7 micron, which, he states, is achievable. The proposed mean particle size is consistent with ANSI N101.1-1972, "Efficiency Testing of Air Cleaning Systems Containing Devices for the Removal of Particles", which requires that in-place testing be performed using an aerosol with an average particle diameter "of the order of 0.5 micron."

The proposed 0.7 micron particle size is "of the order of 0.5 micron" specified by the industry standard for in-place testing. The test results for the installed filter bank would not be altered by the proposed increased particle size since the in-place test is designed to detect gross openings in the filters caused by damage or improper installation. Filter performance itself would be unaffected by the proposed change since these units are tested for 99% efficiency with 0.3 micron particles prior to delivery to the licensee.

Following discussions between the licensee and us, the licensee agrees that testing HEPA filters in accordance with ANSI N101.1-1972 would provide a test method equivalent to the requested change.

Based on the above considerations, we conclude that the proposed change, as modified by the staff to require testing in accordance with this standard, is acceptable, and provides adequate assurance that the HEPA filters are performing as required. This test method is consistent with Regulatory Guide 1.52 for in-place testing of HEPA filters.

10. The licensee has proposed revised acceptance criteria for periodic containment leak rate testing which are identical to the requirements of 10 CFR Part 50, Appendix J.

The present acceptance criterion for the integrated leak rate test at reduced pressure requires that leakage be less than 0.0386% per day and was formulated prior to publication of this Appendix. The proposed acceptance criterion for this test would be 0.0535% per day. The difference between these values is caused by a temperature factor which is not contained in Appendix J, and which has since been demonstrated to be needlessly conservative and not supportable on a theoretical basis. This conclusion is based on the results of testing performed by Battelle Pacific Northwest Laboratories [1] and Idaho Nuclear Corporation [2], which indicate that pressure testing using ambient air (~70°F) results in measured leak rates <u>greater</u> than those measured using heated air or steam-air mixtures. Since the effect of the present temperature factor is to <u>reduce</u> the allowable leak rate at ambient conditions, it represents conservatism not supported by these test results. The present acceptance criterion for maximum leakage from all penetrations and valves also contains this temperature factor and is equal to 0.043% per day. The licensee has proposed to revise this to 0.060% per day for the same reasons discussed above. This is also consistent with the requirements of Appendix J.

(1) BNWL-1475, "Leakage Rate Tests on CSE Containment Vessel With Heated Air and Steam-Air Atmospheres", Battelle Pacific Northwest Laboratory

(2) IN-1399, "Final Results of the Carolina-Virginia Tube Reactor Containment Leakage Test", Idaho Nuclear Corporation

Based on the above considerations and the fact that the maximum allowable leakage rate under design basis accident conditions remains unchanged, we conclude that the proposed change is acceptable.

Following discussions with the licensee, Section 4.5.1.d concerning test frequency should be amended to eliminate an incorrect reference and to correctly reflect current Appendix J requirements for test schedules in the event that the acceptance criteria are not met for a periodic test.

11. A change is requested to Section 4.7.1.c to clarify the diesel generator inspection surveillance requirement. At present, the specification states that each diesel generator be subjected to a thorough inspection at least annually, following the recommendations of the manufacturer for this class of service. Since, according to the licensee, the manufacturer's recommendations are based on actual engine run times with no annual requirement, the specification is in apparent conflict with the manufacturer's recommendations.

    The licensee has proposed to perform diesel generator maintenance strictly in accordance with the manufacturer's recommendations, and has proposed to delete the annual inspection requirement. This proposal would resolve the conflict. However, there are two undesirable facets to this proposal. Considering the limited operation of these units, the inspection interval would be extended well beyond the annual inspection frequency. Second, the required inspections could fall during the operating phase between refuelings. Some maintenance, both corrective and preventative, is expected in this period but major inspections which can be anticipated should be conducted during plant shutdown in order to keep the diesel generator availability as high as possible.

    Accordingly, the licensee's proposal should be modified to require that a diesel generator inspection be performed [in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service] at least once per 18 months during plant shutdown. We have also added a statement requiring the licensee to utilize his best efforts to schedule major diesel generator inspections and overhauls during plant outages. The revised specification would require that an appropriate inspection, dependent on engine hours or otherwise, be developed and performed at least once per 18 months. The increased inspection interval is balanced by the added requirement that the plant be shut down for the inspection. This will result in higher diesel generator availability during plant operation, and a neglibible difference in reliability considering the limited number of running hours for diesel generators used for this class of standby service (on which most of the recommended inspections are based). We have concluded, therefore, that the proposed change, as modified by the staff, is acceptable. This inspection requirement and interval is consistent with Tech Specs currently being issued.

12. No action should be taken at this time with respect to the proposed changes to Section 6, "Administrative Controls", since further changes to this section are currently being contemplated by the licensee.

13. The balance of the requested changes are editorial in nature-corrections of errors, improvement of language, or rearrangement of text for more logical presentation. These are summarized in Table 1. These requested changes do not involve safety considerations, and are acceptable.

Conclusion

We have concluded, based on the reasons discussed above, that the authorization of this change does not involve a significant hazards consideration. We also conclude that 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 and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 11, 1975

UNITED STATES Nuclear Regulatory ~~ATOMIC ENERGY~~ COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory ~~Atomic Energy~~ Commission (the Commission) has issued Amendment No. 12 to Provisional Operating License No. DPR-20 issued to Consumers Power Company which revised Technical Specifications for operation of the Palisades Plant, located in Covert Township, Van Buren County, Michigan. The amendment is effective as of its date of issuance.

This amendment involves: editorial corrections to clarify the intent of several specifications and to correct typographical errors; a modified specification for reactor coolant flow testing; revised acceptance criteria for containment leak testing; revised diesel generator surveillance; the addition of allowable tolerances for safety valve settings; modified procedures for testing reactor coolant samples and air filters; new surveillance requirements for the hydrogen recombiners; a revised setting for initiation of engineered safeguards on high containment pressure; elimination of surveillance requirements for the Reactor Protection System during prolonged refueling shutdown periods; modifications to the surveillance requirements for the containment high pressure channels; a revised method for testing the Safety Injection Refueling Water Tank Level Switch interlock and a different surveillance test for the interlock on the shutdown cooling system isolation valves.

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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.

For further details with respect to this action, see (1) the application for amendment dated April 8, 1974, (2) Amendment No. 12 to License No. DPR-20, with any attachments, 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, NW., Washington, D.C., and at the Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory ~~Atomic Energy~~ Commission, Washington, D.C. 20555 ~~20545~~, Attention: ~~Deputy~~ Director, Division of Reactor ~~for Reactor Projects, Directorate of~~ Licensing ~~- Regulation.~~

Dated at Bethesda, Maryland, this FEB 11 1975

FOR THE Nuclear Regulatory ~~ATOMIC ENERGY~~ COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
~~Directorate of Licensing~~
Division of Reactor Licensing

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- ☐ Full Report
- ☐ Information
- ☐ See Me
- ☐ Allotment Symbol
- ☐ Approval/Signature
- ☐ Comment/Concurrence
- ☐ Necessary action
- ☐ Note and Return
- ☐ Per Conversation
- ☐ Read & Destroy
- ☐ Recommendation
- ☐ Handle Directly
- ☐ Immediate Action
- ☐
- ☐
- ☐ Answer or Acknowledge Before ____________
- ☐ Prepare Reply for the Signature of ____________

REMARKS:

Charles:

I guess I can't blame them too much for this position. Charge ahead with S.E.

10/18

FROM			
Name	Div./Off./Br.	Date	Telephone

NOTE TO: R. PURPLE
THRU: D. KARTALIA

SUBJECT: PRELIMINARY DETERMINATION FOR CHANGES TO PALISADES TECH. SPECS.

In this instance, OGC is unable to make a preliminary determination regarding prenoticing of the numerous changes to the Palisades tech. specs. on the basis of the information presented in the preliminary determination package. We recognize that developing the necessary background analysis prior to the staff's safety evaluation is especially difficult and burdensome where the applicant is proposing numerous changes. Nevertheless, an analysis of each item sufficient to show the basis for the preliminary determination in light of RP Operation Procedure 601 is necessary for meaningful OGC review.

It is our understanding that in this instance the staff's safety evaluation will be ready very shortly. Therefore, the most practical course appears to be that of withholding OGC concurrence pending review of the staff's final recommended noticing determination issued coincident with the safety evaluation.

Lawrence Brenner

Lawrence Brenner
Attorney, OGC

PRELIMINARY DETERMINATION

NOTICING OF PROPOSED LICENSING AMENDMENT

Licensee: Consumers Power Company - Palisades

Request for: Changes to Technical Specifications, Appendix A, to Provisional Operating License No. DPR-20

Request Date: April 8, 1974

Proposed Action: ( ) Pre-notice Recommended

(x) Post-notice Recommended

( ) Determination delayed pending completion of Safety Evaluation

Basis for Decision: This change involves numerous editorial changes, corrections, and clarifications that, according to the licensee, have been brought to light through several years usage. Most of the changes appear to improve clarity or update the existing Tech Specs. Because of the varied nature of the changes, they are briefly summarized below:

1. Added safety valve tolerances (±1%).
2. Clarification involving release of low level gaseous waste following maintenance and pressure tests of tanks and piping.
3. Clarification of setpoint for containment high pressure (ECCS actuation) to eliminate possible overlap from the reactor trip setting.
4. Elimination of the requirement for surveillance testing of the Reactor Protective System when extended refueling shutdowns are involved.
5. Clarification of plant conditions under which low reactor coolant flow circuit can be tested.

(see back for additional items)

CONCURRENCES:

1. CTrammell/ABurger CMTrammell R. R. Burger
2. RAPurple RAPurple 9/30/74
3. K. R. Goller K.R.G. 10/1/74
4. Office of ~~General Counsel~~