Docket No. 50-255

MAR 2 1973

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

Change No. 4
License No. DPR-20

Gentlemen:

On December 8, 1972, we changed the Interim Special Technical Specifications (Appendix B) of Provisional Operating License No. DPR-20 (Amendment No. 4) for the Palisades Plant. This change authorized a temporary power increase up to 85% of full power (1870 MWt) for an operating period not to exceed 750 effective full power hours (about 6 weeks of operation at 85% of full power). In our December 8, 1972, letter we also stated that we were continuing to review the methods used to determine the fuel rod gap thermal conductance and that as a result of our review an increase in the linear power density might be allowed at some time in the future. On December 11, 1972, Combustion Engineering, your nuclear steam supplier, submitted a topical report, CENPD-73, which provided responses to our questions with respect to fuel rod cladding creep collapse predictions. You submitted information to us on December 18, 1972, January 3, 1973, January 16, 1973, and February 12, 1973, which we have used in our reconsideration of gap conductance. In your January 3, 1973, submittal you proposed changes to the Technical Specifications that would permit operation of the Palisades Plant at full power (2200 MWt) for a period not to exceed the first 6000 MWd/Te of operation. This proposal has been designated Change No. 4.

The recent submittals supplement information you sent to us on October 12, November 13, November 21, and December 8, 1972, present your analyses justifying the power increase, evaluate operation of the primary coolant system at 1800 psia, and address concerns related to densified fuel and collapsed fuel cladding. Based on our evaluation, we conclude that the local axial flux peaking augmentation factors for Palisades should remain as given in our December 8, 1972 letter to you. The consequences of anticipated transients and accidents at the reduced primary system pressure

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are no worse than those reported in the FSAR. We agree that collapse of the fuel rods is not expected to occur prior to a burnup value of 6000 MWd/Te for operation of the primary system at 1800 psia. We have used very conservative values for the fuel rod gap thermal conductance for densified fuel in analyzing the consequences of the loss-of-coolant and other accidents. As stated in our December 8, 1972, letter, our continuing review of methods for determining fuel rod gap thermal conductance at various operating conditions may provide a basis for consideration of revised allowable gap conductance values and a corresponding increase in allowable maximum linear power density levels in the future.

Based upon our evaluation, we approve continued operation of Palisades at 85% of full power (1870 MWt), for an operating period not to exceed 1500 effective full power hours after December 8, 1973. This increase is subject to the enclosed revisions to your Operating License.

We conclude that the approved change does not involve significant hazard considerations not described or implicit in the Final Safety Analysis Report and that there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 40.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License No. DPR-20 are hereby changed as set forth in revised pages which are enclosed.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Revised Tech Spec Pages

cc w/encl:
J. L. Bacon, Esq.
Consumers Power Company
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Mr. Paul A. Perry, Secretary
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"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)	$\leq$ 106.5% of Rated Power	< 45% of Rated Power	$\leq$ 25% of Rated Power
2.	Low Primary Coolant Flow(2)	$\geq$ 95% of Primary Coolant Flow With 4 Pumps Operating	$\geq$ 71% of Primary Coolant Flow With 4 Pumps Operating	$\geq$ 46% of Primary Coolant Flow With 4 Pumps Operating
3.	High Pressurizer Pressure	$\leq$ 2400 Psia	$\leq$ 2400 Psia	$\leq$ 2400 Psia
4.	Thermal Margin/Low Pressure(2)(3)	$P_T \geq$ 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	At the Center Line of Feed-Water Ring (6'-0" Below Normal Water Level)	At the Center Line of Feed-Water Ring (6'-0" Below Normal Water Level)	At the Center Line of Feed-Water Ring (6'-0" Below Normal Water Level)
6.	Low Steam Generator Pressure(2)	$\geq$ 500 Psia	$\geq$ 500 Psia	$\geq$ 500 Psia
7.	Containment High Pressure	$\leq$ 5 Psig	$\leq$ 5 Psig	$\leq$ 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 ^oF. Minimum trip setting shall be 1750 psia for two- and three-pump combinations. For four-pump operation, the minimum trip setting shall be 1650 psia for nominal operating pressures less than 1900 psia; and 1750 psia for nominal operating pressures 1900 psia and greater. 4

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

respectively, define the limiting values of primary coolant pressure, reactor inlet temperature, and reactor power level in order that the thermal criteria given in Reference 6 are not exceeded. Figure 3-31 of the FSAR shows the lines of constant DNBR and forms the bases of the thermal limit curves shown on Figures 2-1, 2-2 and 2-3.

To avoid the possibility of DNBR resulting from local flow oscillations, a conservative limit to prevent flow instabilities has been established. A thermal margin trip (TM trip) will occur before the flow instability limit is reached; thus, DNBR resulting from flow oscillations is prevented.[7]

The corollary thermal and hydraulic design bases for the settings are set forth in Section 3 of the FSAR. For four primary coolant pump operation at nominal operating pressures of 1900 psia or greater, the low-pressure trip (LP) of 1750 psia causes a reactor trip in the unlikely event of a loss-of-coolant accident. Similarly, for four primary coolant pump operation at nominal operating pressures of less than 1900 psia, the low-pressure trip of 1650 psia causes a reactor trip in the unlikely event of a loss-of-coolant accident. Operation with a nominal operating pressure of less than 1900 psia will be in accordance with Figure 2-4. 4

A TM trip is initiated by a continuously computed function of reactor inlet temperature, reactor outlet temperature and pressure. The generated function represents a measure of the combination of pressure, temperature and power to flow ratio (as indicated by reactor temperature rise) at which a shutdown should occur to prevent violating the DNBR and void fraction limits. The formula for the TM trip (P_T) set point, $P_T = AT_h - BT_c - C$, is based on reactor inlet temperature (T_c) and reactor outlet temperature (T_h), which defines the minimum allowable pressure for operation and is continuously compared to the

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

measured pressurizer pressure. Conservative values for these constants (A, B and C) will be determined during the power test program at power levels below those where TM protection would be required. These settings will be such that, including instrumentation errors, operation in violation of the thermal limits as shown on Figure 2-3 will result in a reactor trip. At no time will the calculated trip pressure (P_T) be less than the minimum allowable pressure for the applicable reactor condition shown on Figure 2-3.

For operations with either two or three primary coolant pumps, the high-power level trip will be reduced such that operations in violation of the core safety limits as shown on Figure 2-1 or 2-2, respectively, are not possible. These reduced high-power level trip settings and the TM trip setting will be reviewed after the power test program has been completed and the results evaluated. In order to compensate for maximum temperature and pressure measurement errors, the values chosen will be such that a TM trip will always occur before the thermal limits are exceeded. Accordingly, the maximum error assumed is -138 psi[8] in accident analysis. The trip pressure (P_T) calculated by the TM function should, therefore, be at least 139 psi above the pressure shown in Figure 2-3 for a given combination of reactor inlet temperature and pressure. For four-pump operation, the trip pressure (P_T) is continuously indicated and can be compared to the thermal limit curve to check for proper adjustment in operation of the TM trip circuit. The safety setting limits used when operating at either 45% of rated power for three-pump operation or at 25% of rated power for two-pump operation assure that with the secondary coolant system safety valves set at 1000 psia, and with a fixed power level, the T_c cannot be above the limiting T_c temperature. Therefore, with a reactor trip occurring at 1750 psia (which sets a minimum pressure condition that cannot be less than an acceptable pressure level),

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

it is impossible for the core DNBR and/or void fraction safety limits to be violated. These lower three- and two-pump power levels will be set by the high-power level - low-primary coolant flow switch.

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

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

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

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

8. Low Power Physics Testing

   For low power physics tests, certain tests will require the reactor to be critical at low temperature ($\geq 260^{\circ}F$) and low pressure (≥ 415 psia). For these certain tests only, the thermal margin/low pressure, and low steam generator pressure trips may be bypassed in order that reactor

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power can be increased for improved data acquisition. Special operating precautions will be in effect during these tests in accordance with approved written testing procedures. At reactor power levels below 10^{-1}% of rated power, the thermal margin/low-pressure trip is not required to prevent fuel rod thermal limits from being exceeded. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown, should a steam line break occur during these tests.

References

(1) FSAR, Section 14.1

(2) FSAR, Section 7.2.3.2.

(3) FSAR, Section 7.2.3.3.

(4) FSAR, Section 14.7.4.

(5) FSAR, Section 3.3.3.

(6) FSAR, Section 3.3.3.5.

(7) FSAR, Section 3.3.3.6.

(8) FSAR, Section 14.14.3.

(9) FSAR, Section 14.13.3.

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

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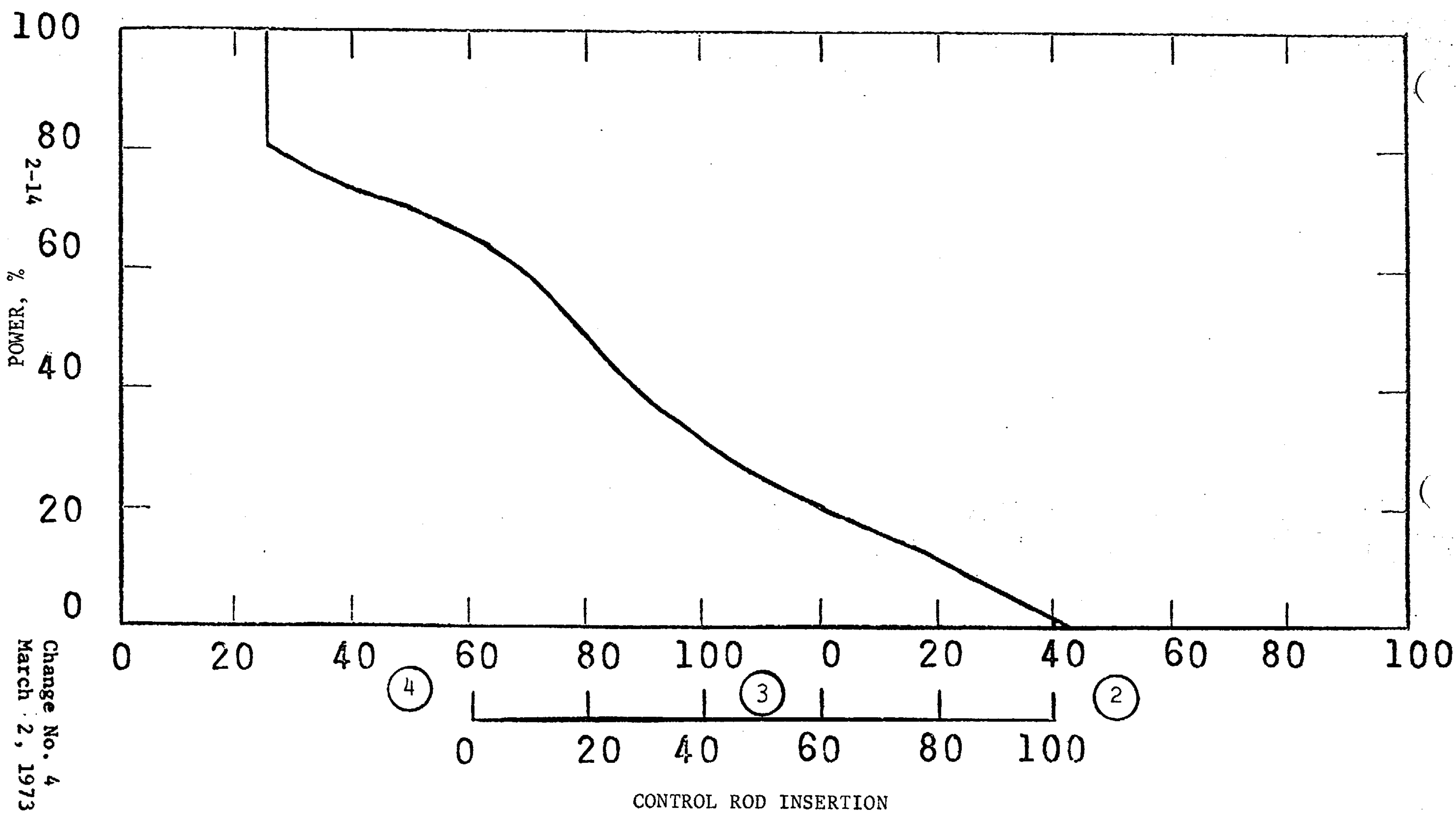
FIGURE 2-4

PALISADES PLANT
CONTROL ROD INSERTION LIMITS
ALLOWED POWER LEVEL (% OF 2200 Mwt) VS CONTROL ROD INSERTION (%) BY ROD GROUP
FOR OPERATION BELOW 1900 psia



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"(1) Added to this is the contribution from N^{16} whose equilibrium radioactivity in the primary coolant is estimated to be 121 uCi/cc. Semi-infinite cloud geometry and uniform mixing of radioactivity in the containment atmosphere was assumed. N^{16} equilibrium exists in containment atmosphere due to its short"

I. Changes (Contd)

B. Change Section 3.16 to read as follows:

"3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS

Applicability

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

Objective

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

Specifications

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

Basis

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

b. Pressurizer Low Pressure - The pressurizer low-pressure safety injection signal is a diverse signal to the high containment pressure safety injection signal. The settings include an uncertainty of -22 psia and are the settings used in the loss-of-coolant accident analysis.[3] 4

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

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

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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 Psig (+0.75 (-0.25	
2. Pressurizer Low Pressure	Safety Injection	≥1550 Psia[(1)] for Nominal Operating Pressures <1900 Psia ≥1593 Psia[(2)] for Nominal Operating Pressures ≥1900 Psia	4
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	$\leq 2.2 \times 10^5$ Cpm	

(1) May be bypassed below 1600 psia and is automatically reinstated above 1600 psia.
(2) May be bypassed below 1700 psia and is automatically reinstated above 1700 psia.
(3) May be bypassed below 550 psia and is automatically reinstated above 550 psia." 4

APPENDIX B

TO

PROVISIONAL OPERATING LICENSE DPR-20

INTERIM SPECIAL TECHNICAL SPECIFICATIONS

FOR THE

PALISADES PLANT

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

Date of Issuance: March 2, 1973

INTERIM SPECIAL TECHNICAL SPECIFICATIONS

FOR OPERATION OF THE PALISADES PLANT

DOCKET NO. 50-255

1. The maximum steady state core power level shall not exceed 1870 MWt.

2. The primary to secondary leakage in a steam generator shall not exceed 0.3 gpm for any period greater than 24 consecutive hours.

3. For the first 1500 effective full power hours of operation after December 8, 1972:

   a. The peak linear power with appropriate consideration of normal flux peaking, flux peaking augmentation factors, measurement-calculational uncertainty (10%), engineering factor (3%), increase in linear heat rate due to axial fuel densification (1.75%) and power measurement uncertainty (2%), shall not exceed 10.1 kW/ft.

   b. Flux peaking augmentation factors used will be assumed to vary linearly from 1.0 at the bottom of the core to 1.16 at the top of the core.

   c. For power operation above a power level of 70% incore detector alarms generated by the data logger shall be set, based on the latest power distribution obtained, such that the peak linear power including appropriate consideration of flux peaking augmentation factors does not exceed 10.5 kW/ft at the alarm set point. If four or more coincident alarms are received, the validity of the alarms shall be immediately determined and, if valid, power shall be immediately decreased below alarm set point and a power distribution map obtained. If a power distribution is not obtained within 24 hours of the alarm condition power shall be reduced to 70%.

   d. The incore detector alarm set points shall be established based on the latest power distribution maps, normalized to 1870 MWt, and shall not exceed the latest power distribution map, normalized to 1870 MWt by more than 20%.

   e. Power distributions shall be evaluated every week or more often as required by plant operations.

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f. Primary coolant gross radioactivity shall be measured at least five times per week and after each significant operating event which could affect fuel-clad integrity.

Primary coolant gross gamma radioactivity shall be monitored continuously by the fission product monitor. If the fission product monitor is not operating, the primary coolant gross radioactivity shall be measured at least once per day.

g. Secondary coolant gas radioactivity shall be monitored continuously by the air ejector gas monitor.

Secondary coolant gross radioactivity shall be measured at least twice per week. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured at least once per day to evaluate steam generator leak tightness.

h. A monthly report of primary and secondary activity measurements, effluent discharge radioactivity levels and core average fuel burnup shall be made to the Directorate of Licensing.

i. Rates of power increase shall not exceed 10% power/hour.

j. During the initial power increase to 85% power primary coolant gross radioactivity shall be monitored:

(1) Continuously by the fission product monitor and daily by grab sample analysis.

(2) If the fission product monitor is not operating, grab sample analysis shall be performed each shift.

4. Nominal primary system operating pressure shall not exceed 1800 psia.

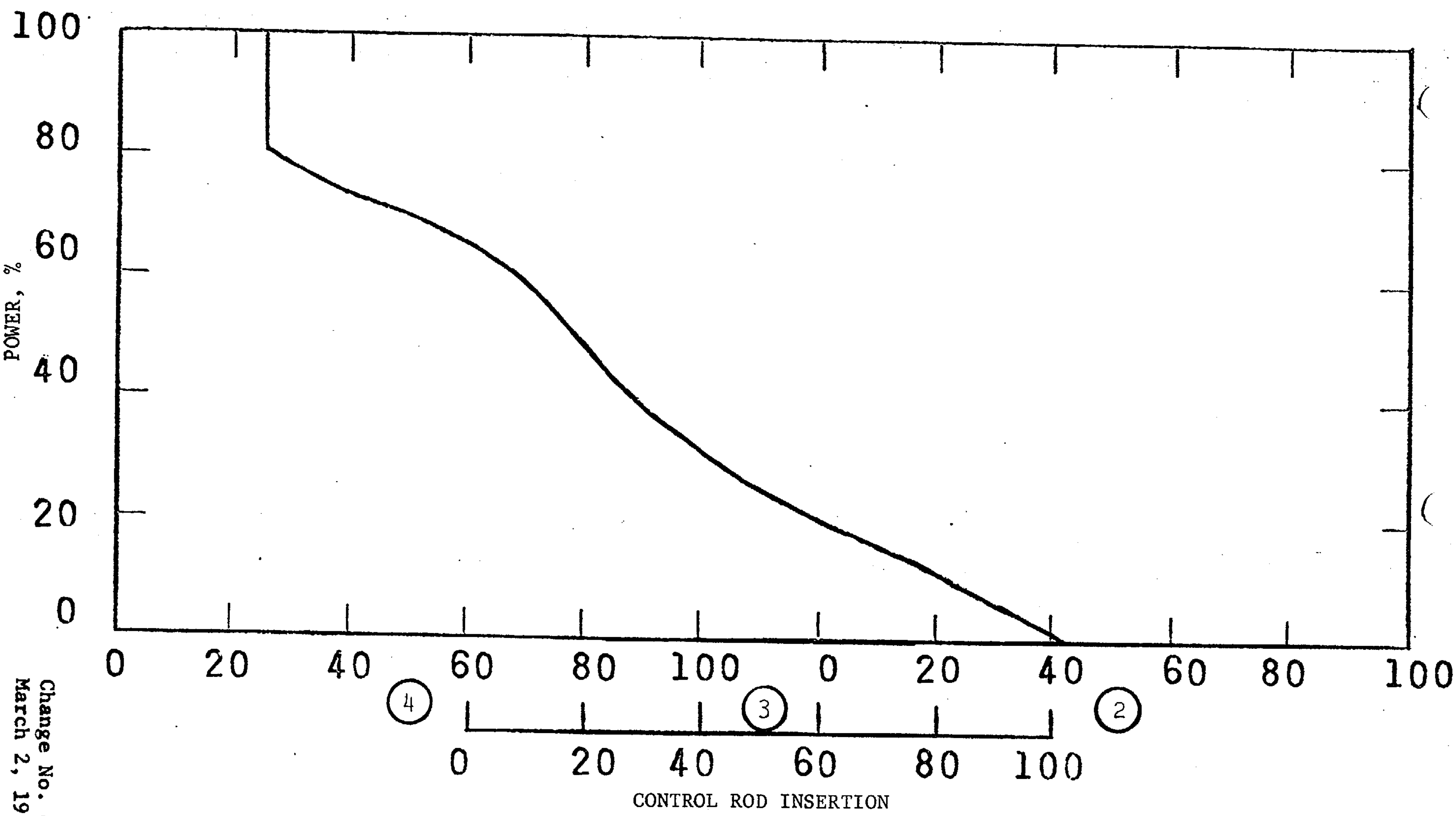
5. If at the end of the 1500 effective full power hour period additional information has not been submitted and approved by the Directorate of Licensing, the power level will be reduced immediately to 1320 MWt.

6. The data logger can be inoperable for two hours. If at the end of two hours, it is not available, the power level shall not exceed 70% of full power.

7. Control rod insertion limits will be in accordance with Figure B-1.

FIGURE B-1

PALISADES PLANT
CONTROL ROD INSERTION LIMITS
ALLOWED POWER LEVEL (% OF 2200 Mwt) VS CONTROL ROD INSERTION (%) BY ROD GROUP



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