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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 21
License No. DPR-20

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

   A. The applications for amendment by Consumers Power Company (the licensee) dated July 9, 1975, and January 30 and April 5, 1976, as supplemented and amended, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 29, 1976

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered revised pages:

ii	3-29
iii	3-33, 3-59, 3-60, 3-62, 3-63
2-5	3-65
3-1	3-66, 4-9
3-1a	5-3
3-3	6-26

Add the following new pages:

3-29a	3-86
3-66a	3-87
3-84	4-70
3-85	

Remove Fig. 2-4 (page following page 2-13). No replacement.

Revise Appendices B and C as follows:

Remove entire contents of both appendices including title pages.

TABLE OF CONTENTS (Contd)

TABLE OF CONTENTS (Cont'd).

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(4) (Continuous operation not permitted)	≤ 25% of Rated Power(4) (Continuous operation not permitted)
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	1950 ± 20 psia	1950 ± 20 psia	1950 ± 20 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	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level	Not Lower Than the Center Line of Feed-Water Ring Which Is Located 6'-0" Below Normal Water Level
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.

(4) Operation with two or three pumps is permitted to provide a limited time for repair/pump restart, to provide for an orderly shutdown, or to provide for the conduct of reactor internals noise monitoring test measurements (a maximum of 12 hours each time this test is conducted).

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 PRIMARY COOLANT SYSTEM

Applicability

Applies to the operable status of the primary coolant system.

Objective

To specify certain conditions of the primary coolant system which must be met to assure safe reactor operation.

Specifications

3.1.1 Operable Components

a. At least one primary coolant pump or one shutdown cooling pump shall be in operation whenever a change is being made in the boron concentration of the primary coolant.

b. Four primary coolant pumps shall be in operation whenever the reactor is operated continually above 5% of rated power (exception to this specification is permitted as described in Table 2.3.1, Item 1).

c. The minimum flow for various power levels shall be as shown in Table 2.3.1. The measured 'Four Primary Coolant Pumps Operating' reactor coolant vessel flow (as determined by reactor coolant pumps differential pressure and pump performance curves) shall be 124.0×10^6 lb/h or greater, when corrected to 532^oF.

In the event the measured flow is less than that required above, the limits specified in 3.19.1 shall be reduced by 2% for each 1% of reactor flow deficiency.

Continuous operation at power shall be limited to four pump operation. Following loss of a pump, thermal power shall be reduced as specified in Table 2.3.1 and appropriate corrective action implemented. With one or more pumps out of service, return the pumps to service (return to four pump operation) or be in hot standby (or below) within 24 hours. Start-up (above hot standby) with less than four pumps is not permitted.

d. Both steam generators shall be capable of performing their heat transfer function whenever the average temperature of the primary coolant is above 325^oF.

e. Maximum primary system pressure differentials shall not exceed the following:

(1) Maximum steam generator operating transient differential of 1530 psi.

(2) Hydrostatic tests shall be conducted in accordance with applicable paragraphs of Section XI ASME Boiler & Pressure Vessel Code (1974). Such tests shall be conducted with sufficient pressure on the secondary side of the steam generators to restrict primary to secondary pressure differential to a maximum of 1380 psi. Maximum hydrostatic test pressure shall not exceed 1.1 Po plus 50 psi where Po is nominal operating pressure.

(3) Primary side leak tests shall be conducted at normal operating pressure. The temperature shall be consistent with applicable fracture toughness criteria for ferritic materials and shall be selected such that the differential pressure across the steam generator tubes is not greater than 1380 psi.

(4) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum temperature of 100°F is required. Only 10 cycles are permitted.

(5) Maximum secondary leak test pressure shall not exceed 1000 psia. A minimum temperature of 100°F is required.

(6) In performing the tests identified in 3.1.1.e(4) and 3.1.1.e(5), above, the secondary pressure shall not exceed the primary pressure by more than 350 psi.

f. Nominal primary system operating pressure shall be 1800 psia.

g. The reactor coolant temperature at the inlet to the reactor vessel shall be no greater than 525°F (indicated) during steady state operation above 80% of rated power.

The maximum steam generator operating transient differential pressure is expected to occur during a loss of load transient. The loss of load accident, initiated at a nominal reactor coolant system pressure of 2100 psia and a nominal high pressurizer pressure trip of 2400 psia is analyzed in Section 14.12 of the FSAR. Results of this analysis indicate that the maximum steam generator differential pressure is 1530 psi (assuming that reactor control is in the automatic mode, and that steam dump, bypass and pressurizer relief valves function). This pressure differential is approximately 11% greater than that allowed during normal operation, so that substantial safety margin exists between this pressure differential and the pressure differential required for tube rupture.

Secondary side hydrostatic and leak testing requirements are consistent with ASME BPV Section XI (1971). The differential maintains stresses in the steam generator tube walls within code allowable stresses.

The minimum temperature of 100°F for pressurizing the steam generator secondary side is set by the NDTT of the manway cover of +40°F.

The ECCS analysis has been conducted at a vessel flow of 124.0×10^6 lb/h, and the primary system flow areas and loss coefficients used in the analysis were forced to agree with this flow. The DNB analysis (assuming 122% margin to overpower) has also been performed at this flow with a 3% uncertainty. The ECCS limits associated with this flow rate, which may be more restrictive than the DNB limits, are specified in Section 3.19.1.

In the event the measured flow is less than the required flow, a decrease in allowable thermal limits is required. This decrease in thermal limits, at twice the percentage by which flow is decreased, conservatively maintains the power to flow ratio and provides adequate margin for transients and accidents.

References

(1) FSAR, Sections 6.1.2.2, 14.3.2.

(2) FSAR, Section 4.3.7.

3.3 EMERGENCY CORE COOLING SYSTEM

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core in either emergency or normal shutdown situations.

Specifications

Safety Injection and Shutdown Cooling Systems

3.3.1 The reactor shall not be made critical, except for low-temperature physics tests, unless all of the following conditions are met:

a. The SIRW tank contains not less than 250,000 gallons of water with a boron concentration of at least 1720 ppm at a temperature not less than 40°F.

b. All four safety injection tanks are operable and pressurized to at least 200 psig with a tank liquid level of at least 186 inches (55.5%) and a maximum level of 198 inches (59%) with a boron concentration of at least 1720 ppm.

c. One low-pressure safety injection pump is operable on each bus.

d. One high-pressure safety injection pump is operable on each bus.

e. Both shutdown heat exchangers and both component cooling heat exchangers are operable.

f. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the primary coolant system.

g. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable.

h. The Low Pressure Safety Injection Flow Control Valve CV-3006 shall be opened and disabled (by isolating the air supply) to prevent spurious closure.

i. The Safety Injection bottle motor-operated isolation valves shall be opened with the electric power supply to the valve motor disconnected.

j. The Safety Injection miniflow valves CV-3027 and 3056 shall be open with HS-3027 and 3056 positioned to maintain them open.

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

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

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

b. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

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

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

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

References

(1) FSAR, Section 9.10.3.

(2) FSAR, Section 6.1.

3.10.3 Power Distribution Limits

a. If the quadrant to core average power tilt exceeds 15%, except for physics tests, then:

(1) The hot channel factors shall promptly be demonstrated to be less than design values $F_q^N = 3.62$ $F_{\Delta H}^N = 1.94$ or,

(2) Immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

b. If the power in a quadrant exceeds core average by 10% for a period of 24 hours or if the power in a quadrant exceeds core average by 20% at any time, immediate action shall be initiated to reduce reactor power below 50% until the situation is remedied.

c. If the power in a quadrant exceeds the core average by 15%, and if the hot channel factors cannot be demonstrated promptly to be within design limits, then the overpower trip set point shall be reduced to 80% and the thermal margin low-pressure trip set point (P_{trip}) shall be increased by 400 psi.

d. If the power in a quadrant exceeds core average by 5% for a period of 30 days, immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

e. If the ratio of the power in the upper half to the lower half of the core is not within the range of 0.5 to 1.8 as indicated by the in-core and out-of-core detectors, then immediate action shall be taken to reduce the power below 75% of rated power until the situation is remedied.

f. The part-length control rods will be completely withdrawn from the core (except for rod exercises and physics tests).

3.10.4 Misaligned or Inoperable Control Rod or Part-Length Rod

a. A control rod or a part-length rod is considered misaligned if it is out of position from the remainder of the bank by more than 8 inches.

b. A control rod is considered inoperable if it cannot be moved by its operator or if it cannot be tripped. A part-length rod is considered inoperable if it is not fully withdrawn from the core and cannot be moved by its operator. If more than one control rod or part-length rod becomes misaligned or inoperable, the reactor shall be placed in the hot shutdown condition within 12 hours.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

c. If a control rod or a part-length rod is misaligned, hot channel factors must promptly be shown to be within design limits or reactor power shall be reduced to 75% or less of rated power within 2 hours. In addition, shutdown margin and individual rod worth limits must be met. Individual rod worth calculations will consider the effects of xenon redistribution and reduced fuel burnup in the region of the misaligned control rod or part-length rod.

3.10.5 Regulating Group Insertion Limits

a. To implement the limits on shutdown margin, individual rod worth and hot channel factors, the limits on control rod regulating group insertion shall be established as shown on Figure 3-6. These limits may be revised during fuel cycle life based on physics calculations and physics data obtained during plant start-up and subsequent operation.

b. The sequence of withdrawal of the regulating groups shall be 1, 2, 3, 4.

c. An overlap of control banks in excess of 40% shall not be permitted.

d. If the reactor is subcritical, the rod position at which criticality could be achieved if the control rods were withdrawn in normal sequence shall not be lower than the insertion limit for zero power shown on Figure 3-6.

3.10.6 Shutdown Rod Limits

a. All shutdown rods shall be withdrawn before any regulating rods are withdrawn.

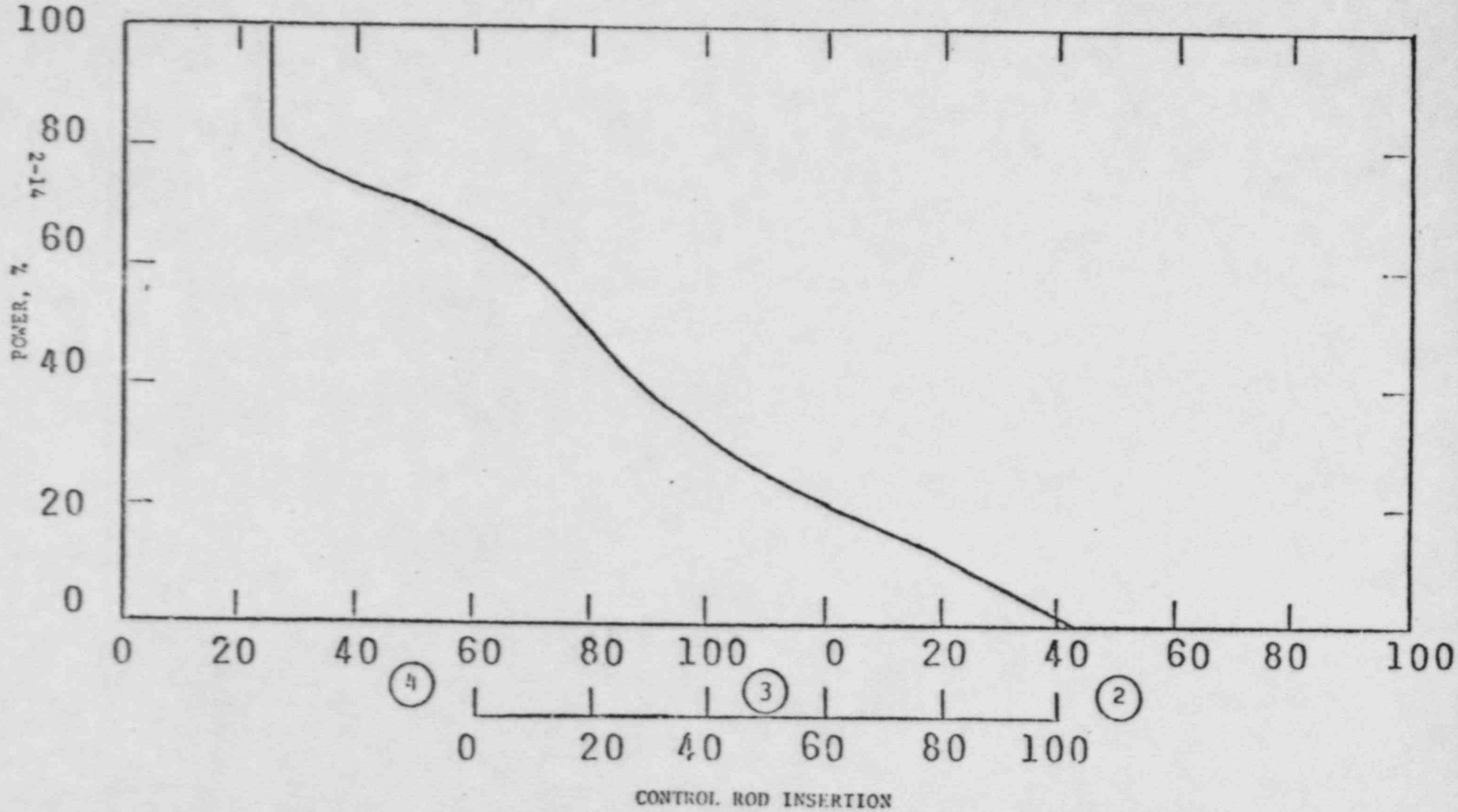
b. The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer.

c. The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted.

3.10.7 Low Power Physics Testing

Sections 3.10.1.a, 3.10.2.b, 3.10.4.b, 3.10.5 and 3.10.6 may be deviated from during low power physics testing and CRDM exercises if necessary to perform a test but only for the time necessary to perform the test.

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



2-14

Fig. 3-6
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value; core outlet thermocouples; and in-core detectors.(3) These values will form the basis for the calculation of peaking factors. Calibration of the out-of-core detectors will take into account the local and total power distribution.

Studies have been performed to show that with this ratio (0.5 - 1.8), total peaking factors are not exceeded. Tests will be performed during start-up testing to verify this relationship.

The insertion of part-length rods into the core, except for rod exercises or physics tests, is not permitted since it has been demonstrated on other CE plants that that design power distribution envelopes can, under some circumstances, be violated by using part-length rods. Further information may justify their use. Part-length rod insertion is permitted for physics tests, since resulting power distributions are closely monitored under test conditions. Part length rod insertion for rod exercises (approximately 6") is permitted since this amount of insertion has an insignificant effect on power distribution.

For a control rod misaligned up to 8 inches from the remainder of the banks, hot channel factors will be well within design limits. If a control rod is misaligned by more than 8 inches, the maximum reactor power will be reduced so that hot channel factors, shutdown margin and ejected rod worth limits are met. If in-core detectors are not available to measure power distribution and rod misalignments >8 inches exist, then reactor power must not exceed 75% of rated power to insure that hot channel conditions are met.

For a "dropped" control rod during operation at high power, a turbine runback will automatically decrease the maximum power to 70% of rated power.(4) Continued operation with that rod fully inserted will only be permitted if the hot channel factors, shutdown margin and ejected rod worths are satisfied.

In the event a withdrawn control rod cannot be tripped, shutdown margin requirements will be maintained by increasing the boron concentration by an amount equivalent in reactivity to that control rod.

The deviations permitted by Spec. 3.10.7 are required in order that the control rod worth values used in the reactor physics calculations, the plant safety analysis, and the Technical Specifications can be verified. These deviations will only be in effect for the time period required for the test being performed. The testing interval during

3.11 IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the in-core instrumentation system.

Objective

To specify the functional and operability requirements of the in-core instrumentation system.

Specification

a. Sufficient in-core instrumentation shall be operable whenever the reactor is operating at or above 50% rated power (65% of rated power if no dropped or misaligned rods are present) in order to: (1) Assist in the calibration of the out-of-core detectors, and (2) check gross core power distribution. As a minimum, 50% of the incore detectors and not less than 10 individual detectors per quadrant, which shall include 2 detectors at each of the four axial levels, shall be operable.

b. For power operation above 85% of rated power, in-core detector alarms generated by the data logger shall be set, based on the latest power distribution obtained, such that the peak linear power does not exceed the limit specified in Section 3.19. 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 conditions, power shall be reduced to 85% of rated power.

c. The in-core detector alarm set points shall be established based on the latest power distribution maps, normalized to the kW/ft limit defined in Section 3.19.

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

e. 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 85% of rated power.

f. If the data logger for the in-cores is not in operation for more than two hours and reactor power is at or above 50% of rated power (65% of rated power if no dropped or misaligned rods are present), readings shall be taken and logged on a minimum of 10 individual detectors per quadrant (to include at least 50% of the total number of detectors in a 10-hour period) at least each two hours thereafter or the reactor power level shall be reduced to less than 50% of rated power (65% of rated power if no dropped or misaligned rods are present). If readings indicate a local power level equal to or greater than the alarm setpoint, the action specified in 3.11.b shall be taken.

Bases

A system of 45 in-core flux detector and thermocouple assemblies and a data display, alarm and record functions has been provided.[1] The out-of-core nuclear instrumentation calibration includes:

a. Calibration (axial and azimuthal) of the split detectors at initial reactor start-up and during the power escalation program.

b. A comparison check with the in-core instrumentation in the event abnormal readings are observed on the out-of-core detectors during operation.

c. Calibration check during subsequent reactor start-ups.

d. Confirm that readings from the out-of-core split detectors are as expected and that the ratio of the top and bottom detector readings is acceptable.

Core power distribution verification includes:

a. Measurement at initial reactor start-up to check that power distribution is consistent with calculations.

b. Subsequent checks during operation to insure that power distribution is consistent with calculations.

c. Indication of power distribution in the event that abnormal situations occur during reactor operation.

If the data logger for the in-core readout is not in operation for more than two hours, power will be reduced to provide margin between the actual peak linear heat generation rates and the limit and the in-core readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the manpower available, the reactor power will be reduced further to minimize the probability of exceeding the peaking factors. The time interval of two hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the data logger is returned to service.

Reference - (1) FSAR, Section 7.4.2.4.

3.19 Linear Heat Generation Rate Limit Associated With LOCA Considerations

Applicability

Applies to fuel linear heat generation rates.

Objective

To delineate the requirements regarding fuel linear heat generation rates associated with a postulated Loss of Coolant Accident.

Specification

3.19.1 The linear heat generation rate with appropriate consideration of normal flux peaking, measurement-calculational uncertainty, engineering factor, increase in linear heat rate due to axial fuel densification, power measurement uncertainty, and flux peaking augmentation shall not exceed 14.19 kW/ft.

The measurement-calculational uncertainty shall be 10%, the engineering factor shall be 3%, the increase in linear heat rate due to axial densification shall be 1.75% (as applied to hot dimensions), the power measurement uncertainty shall be 2%, and the flux peaking augmentation factor shall be as given in Figure 3-7 for uncollapsed fuel and Figure 3-8 for collapsed fuel. Augmentation factors for pressurized densification resistant ENC fuel and pressurized high density CE fuel shall be 1.0.

Bases

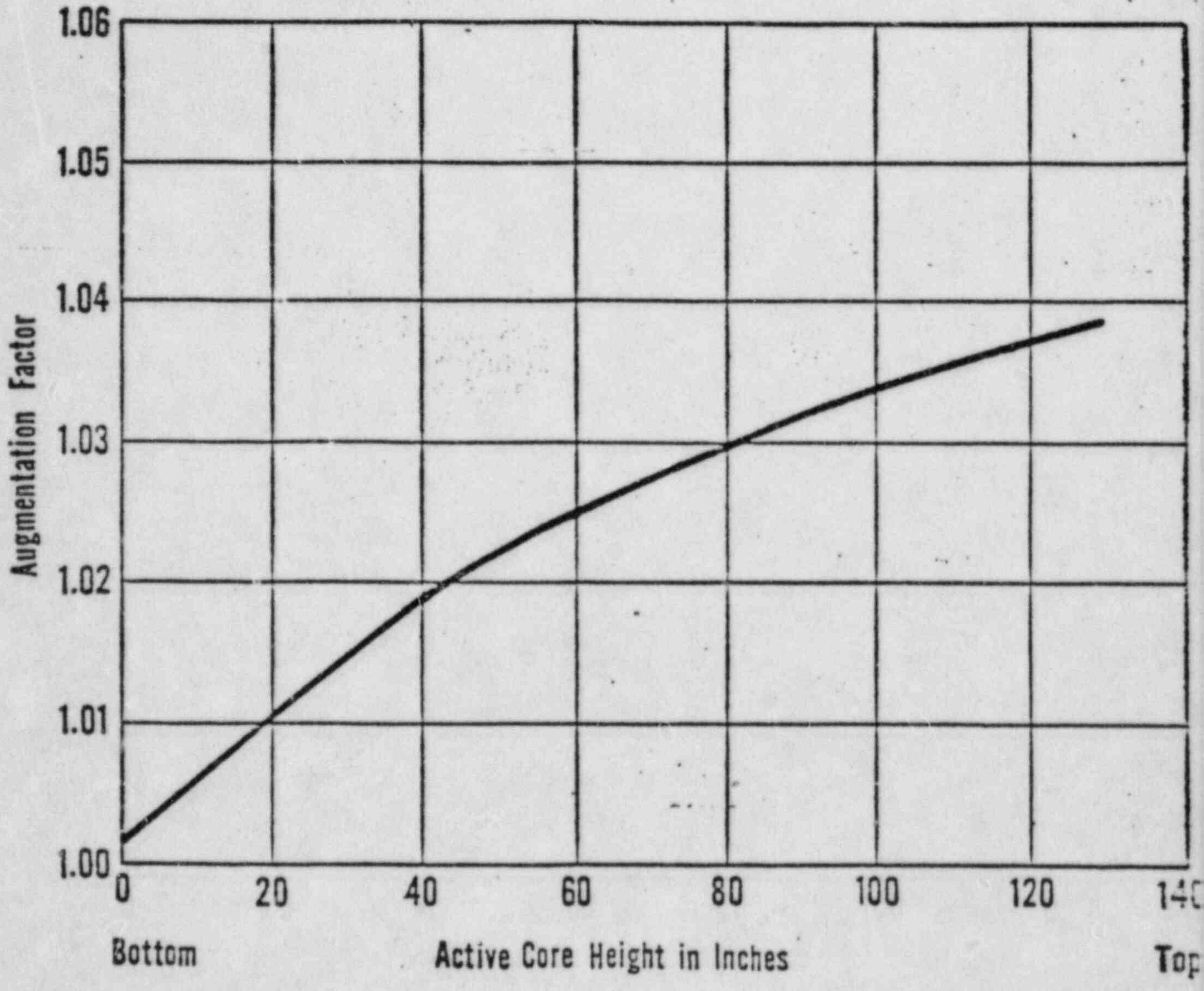
To maintain the integrity of the fuel cladding under the conditions of a postulated Loss of Coolant Accident (LOCA), the Emergency Core Cooling Systems (ECCS) must satisfy certain criteria set forth by the US Nuclear Regulatory Commission in

10 CFR 50.46(b). These criteria assure that under LOCA conditions the temperature and oxidation of the cladding will be controlled such that the uranium dioxide pellets will be maintained in a coolable geometry. These criteria are summarized below:

1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation, including effects of cladding thinning and rupture.
3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The computer codes which predict cladding temperature and oxidation under LOCA conditions must be approved by the US Nuclear Regulatory Commission in accordance with Appendix K of 10 CFR 50. The results of these computer code calculations depend on ECCS performance characteristics and fuel design. Analyses performed with approved codes and techniques for each fuel design type taking into consideration anticipated operating conditions will be kept on file at the plant and at the General Office. These analyses provide safety limits given in terms of allowable linear heat generation rates in kW/ft for each fuel design type.

Appropriate factors for measurement-calculation uncertainty, engineering factor and shortening of the fuel pellet stack are specified to insure that linear heat generation rate limits are not exceeded during steady state operation.



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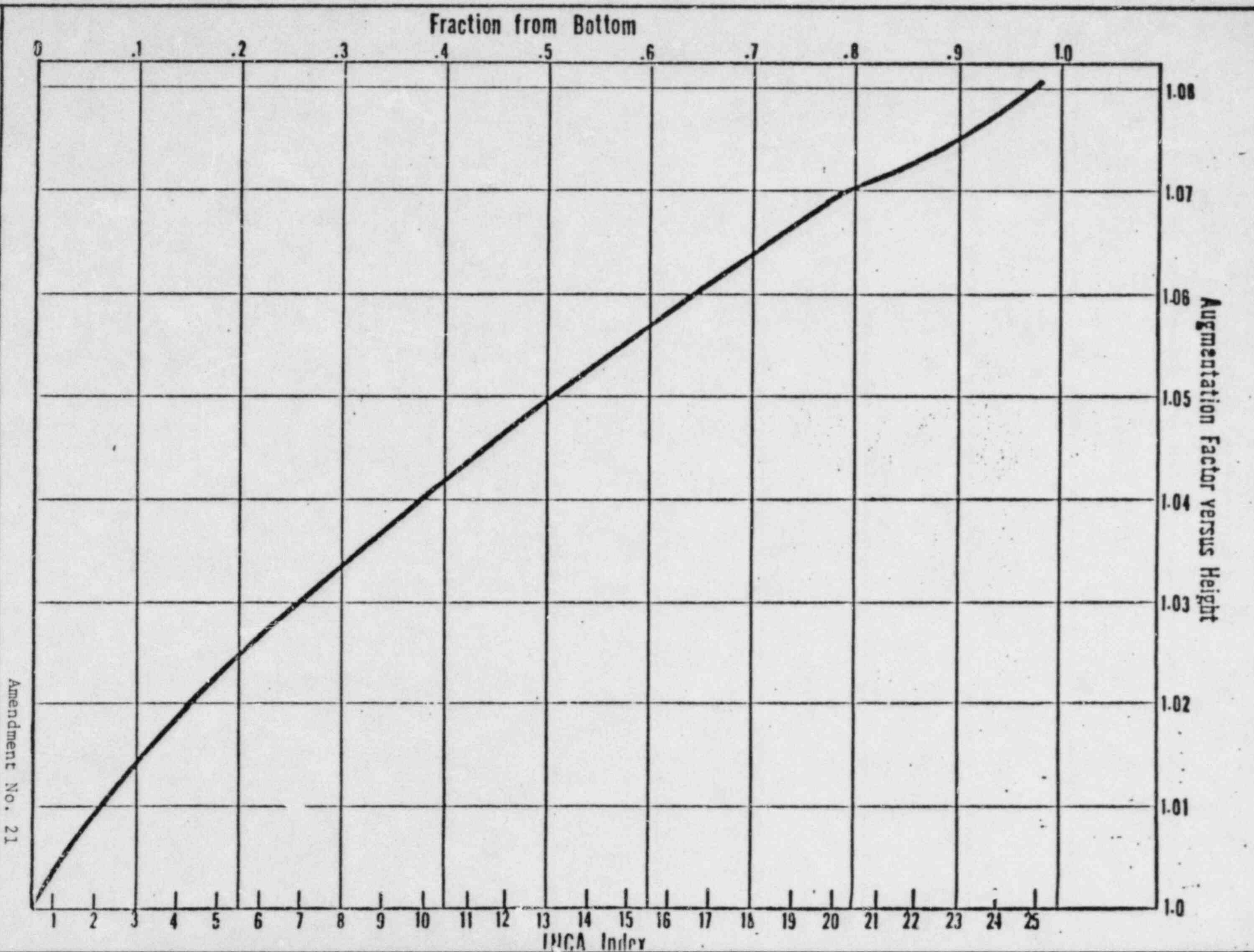
Augmentation Factor Versus Height Monte Carlo Computation for Palisades	Palisades Technical Specifications	Figure 3-7



Augmentation Factor
[From Enclosure to P-CE-4059 10/10/75]

Palisades
Technical Specifications

Figure
3-8

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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.
18. Low Pressure Safety Injection Flow Control Valve CV-3006	a. Check	P	a. Observe valve is open with air supply isolated.
19. Safety Injection Bottle Isolation Valves	a. Check	P	a. Ensure each valve open by observing valve position indication and valve itself. Then lock open breakers (at MCC-9) and control power (key switch in control room).
20. Safety Injection miniflow valves CV-3027, 3056	a. Check	P	a. Verify valves open and HS-3027 and 3056 positioned to maintain them open.

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.

(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

4.15 Primary System Flow Measurement

Applicability

Applies to the measurement of primary system flow rate with four primary coolant pumps in operation.

Objective

To provide assurance that the primary system flow rate is equal to or above the flow rate required in 3.1.1(c).

Specification

After each refueling outage, or after plugging 10 or more steam generator tubes, a primary system flow measurement shall be made with four primary coolant pumps in operation before the reactor is made critical.

Basis

This surveillance program assures that the reactor coolant flow is consistent with that assumed as the basis for Specification 3.1.1(c).

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) (Contd)

5.3.2 Reactor Core and Control

a. The reactor core shall approximate a right circular cylinder with an equivalent diameter of about 136 inches and an active height of about 132 inches.

b. The reactor core shall consist of approximately 43,000 Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall be grouped into 204 assemblies.

A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution.

c. The fully loaded core shall contain approximately 211,000 pounds UO_2 and approximately 56,000 pounds of Zircaloy-4.

Poison may be placed in the fuel bundles for long-term reactivity control.

d. The core excess reactivity shall be controlled by a combination of boric acid chemical shim, cruciform control rods, and mechanically fixed boron rods where required. Forty-five control rods shall be distributed throughout the core as shown in Figure 3-5 of the FSAR. Four of these control rods may consist of part-length absorbers.

5.3.3 Emergency Core Cooling System

An emergency core cooling system shall be installed consisting of various subsystems each with internal redundancy. These subsystems shall include four safety injection tanks, three high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in Section 6 of the FSAR.

5.4 FUEL STORAGE

5.4.1 New Fuel Storage

a. Unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less

6.9.3.3. Special Reports

a. Special reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable referenced specification:

Area	Specification Reference	
Prestressing, Anchorage, Liner and Penetration Tests	4.5.4 4.5.5	90 Days After Completion of the Test*
Primary System Surveillance Evaluation and Review	4.3	Five Years

*A test is considered to be complete after all associated mechanical, chemical, etc., tests have been completed.

b. Bimonthly status reports on the program to improve the reliability of the paths to prevent post-LOCA boron precipitation shall be submitted to the Division of Operating Reactors until completed.

c. The results of the fuel surveillance program (CPCo letters dated March 20 and April 8, 1976) shall be reported to NRC prior to the re-use of Cycle 2 fuel for Cycle 3.

6.10 RECORD RETENTION

(Records not previously required to be retained shall be retained as required below commencing with the effective date of Technical Specification Change No. 20. A system for efficient record retrieval shall be in effect not later than June 1976.)

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Reportable Occurrences.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.