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

YANKEE ATOMIC ELECTRIC COMPANY

DOCKET NO. 50-29

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. DPR-3

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

   A. The applications for amendment by Yankee Atomic Electric Company (the licensee) dated July 14, 1975 [as supplemented October 10, October 28, November 7 (Proprietary Information appended), November 21, and November 26, 1975]; July 8, 1975 (as supplemented November 24, and November 26, 1975); and September 23, 1975, 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 applications, 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; and

   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.

2. Accordingly, Facility License No. DPR-3 is hereby amended by revising Paragraph 3.A.(2) and adding Paragraph 3.A.(3) as follows:

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"(2) Technical Specifications

The Technical Specifications contained in Appendix A, 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. 1 2 6.

"(3) Yankee shall not operate the facility with Core XII beyond low power physics testing until the modifications to preclude single failures in the ECCS (as described in the staff Safety Evaluation issued with Amendment No. 2 1) have been completed.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Karl Goller

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

Attachment:
Change No. 1 2 6 to the
Technical Specifications

Date of Issuance: DEC 4 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 21

CHANGE NO. 126 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-3

DOCKET NO. 50-29

Revise Appendix A as follows:

Remove Pages	Insert New Pages
2	2
---	2a
5	5
---	5a
---	5b (Figure 8-1)
---	5c (Figure 8-2)
---	5d (Figure 8-3)
---	5e (Figure 8-4)
7a	7a
---	7b
509-2	509-2

OFFICE ➤						
SURNAME ➤						
DATE ➤						

214 Nuclear Instrumentation System
215 Radiation Monitoring System
218 Fuel Handling System
224 Compressed Air Systems
231 Vapor Containment
232 Radiation Shielding
235 Architectural Features

Physical arrangements of structures and equipment will be as described in Section 200 of the license application. Mechanical equipment and systems will be interconnected as shown in the Fundamental Flow Diagram included in that section.

Electrical equipment and systems which provide station auxiliary power supply will be as described in Section 226 of the license application and will be interconnected as shown in the 2400 volt one-line diagram and the 480 volt one-line diagram, sheets 1, 2 and 3, included in that section.

The ventilation system for the control room area, radiochemical laboratory, decontamination cubicle, fuel transfer pit house, and other potentially contaminated portions of the Turbine Generator Service, Primary Auxiliary, and Waste Disposal Buildings shall be in accordance with the description contained in Section 228 of Part B of the license application.

3. The Performance Analysis for the current reload core, "Yankee Nuclear Power Station Core XII Performance Analysis, July 9, 1975" (as supplemented October 10 and November 26, 1975), is incorporated as part of these technical specifications. The analysis presented in the FSAR for Core XI forms the basis for the reference core performance analysis. 126

C. PERFORMANCE SPECIFICATIONS

Calculated values of operating variables such as pressures, temperatures, flows, heat fluxes, reactivity coefficients and on-site radiation levels under steady state and transient conditions which are stated in the sections of the license application listed in Paragraph B, above, are considered to be performance specifications of the reactor and are incorporated by reference herein. Yankee shall not operate the facility under circumstances where there is a substantial variance between the foregoing performance specifications and the corresponding values determined by operation of the facility.

The performance and function of the systems described in the following sections of the license application shall be substantially as described; however, the details of individual components and their arrangement as described in each of these sections may be altered by Yankee at its own discretion provided that such an alteration would not violate some other provision of these Technical Specifications:

206 Component Cooling System
208 Sampling System
211 Vent and Drain System, Primary Plant
216 Vapor Container Atmosphere Control Systems
219 Main and Auxiliary Steam System
220 Condensate and Feed Water System
221 Circulating Water System
222 Water Supply System

(9) The Commission shall be immediately notified should an unexplained reactivity change greater than 0.8% $\Delta K/K$ take place at any time subsequent to the first week of full power operation. This reporting requirement shall be in effect only when the boron concentration in the primary system exceeds 80 ppm and within one week after a reduction to a boron concentration of less than 80 ppm.

(10) During a reactor startup in which core reactivity or control rod positions for criticality are not established, a plot of inverse multiplication rate (or count rate) versus rod position should be made.

b. Power Level

(1) With all four main coolant loops providing normal flow to the core, the reactor power level is limited to 600 MW thermal.

(2) With only three main coolant loops providing normal flow to the core, the reactor power level is limited to 450 MW thermal.

(3) The reactor will be scrammed automatically by a high neutron flux level signal, set at not more than 108% of rated power for each condition as defined in (1) and (2) above.

(4) Except for operation of the reactor at plant power levels not exceeding 15 MW electric, the reactor shall not be operated with less than three main coolant loops providing normal flow to the core.

(5) Whenever there is a sustained outage of one of the 115 kV lines because of maintenance or fault condition, the reactor power level shall be reduced to a level consistent with three loop operation as defined in (2) above.

c. Thermal

(1) During steady state power operation, the peak linear heat rate shall not exceed the limits shown in Figure 8-1. With these limits, if full power cannot be attained, the allowable fraction of full power shall be calculated as follows:

$$\text{Allowable Fraction of Full Power} = \frac{\text{Limiting LHGR}}{\text{Peak Full Power LHGR}}$$

where the limiting LHGR is obtained from Figure 8-1. The peak full power LHGR shall include the following: 1 2

a) Heat flux power peaking factor, F_q^N measured using core instrumentation at a power $\geq$ 10%;

b) Effect of inserting the control bank from its position at the time of measurement to its insertion limit (F_I) as shown in Figure 8-2. The rod insertion limit is shown in Figure 8-3;

c) Effect of xenon redistribution (1.10);

d) Flux peaking augmentation factor (power spike) using Figure 8-4;

e) Shortened stack height factor (1.009);

f) Meas_rement uncertainty (1.05);

g) Power level uncertainty (1.03);

h) Heat flux engineering factor, F_q^E, (1.04);

i) Core average linear heat generation rate at full power (4.34 kw/ft).

These factors are multiplicative and items (a) and (d) shall be c_osen at a core height so as to maximize their product. When operating at constant power, all rods out, with equilibrium xenon, power peaking in the Yankee Rowe core decreases monotonically as a function of cycle burnup. This has been verified by both calculation and measurement on Yankee cores and is in accord with the expected behavior in a core that does not contain burnable poison. The all-rods-out power peaking measured at any time in core life thus provides an upper bound on ARO power peaking for the remainder of that cycle. Therefore, the measured power peaking shall be checked every 1000 equivalent full power hours and the latest measured value shall be used in the computation. The only effects which can increase peaking beyond this value would be control rod insertion and xenon transients and these are accounted for in items (b) and (c).

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Figure 8-1

Core XII Allowable Peak LHGR Versus Exposure

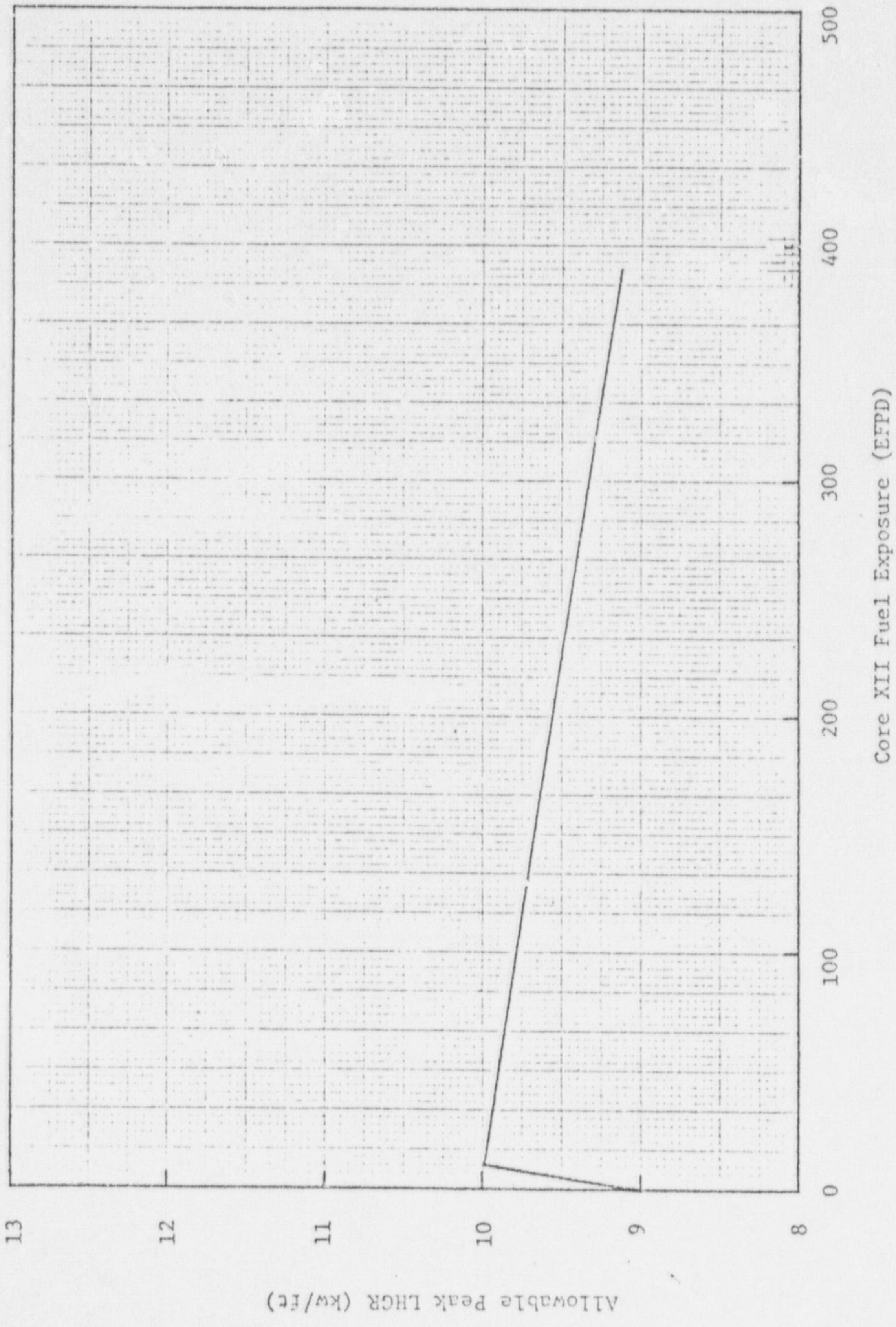
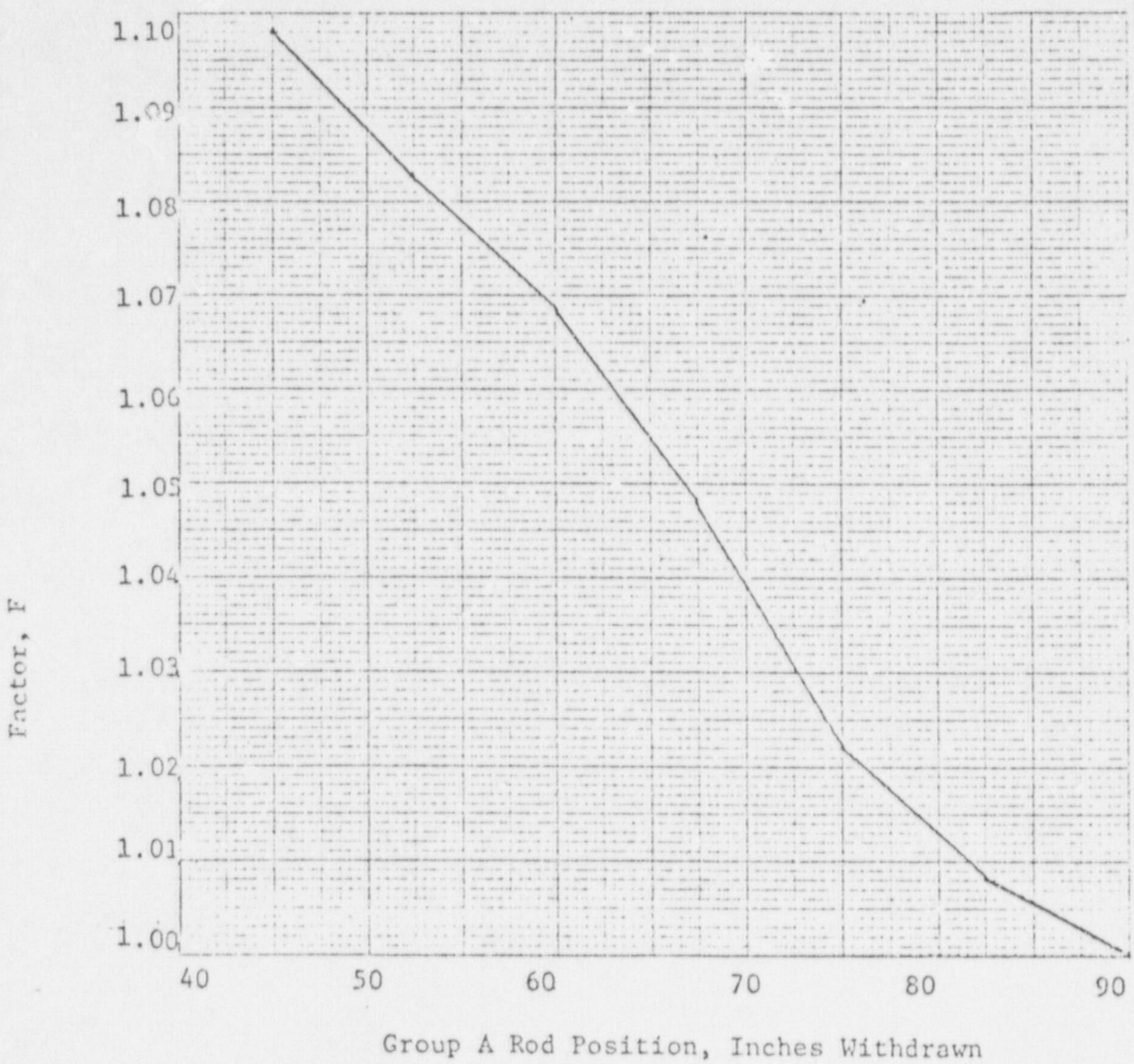


Figure 8-2

Factor F as a Function of Rod Insertion



$$F_I = \frac{F \text{ @ Limit}}{F \text{ @ Measurement}}$$

Figure 8-3

Rod Insertion Limit vs. Allowable Power

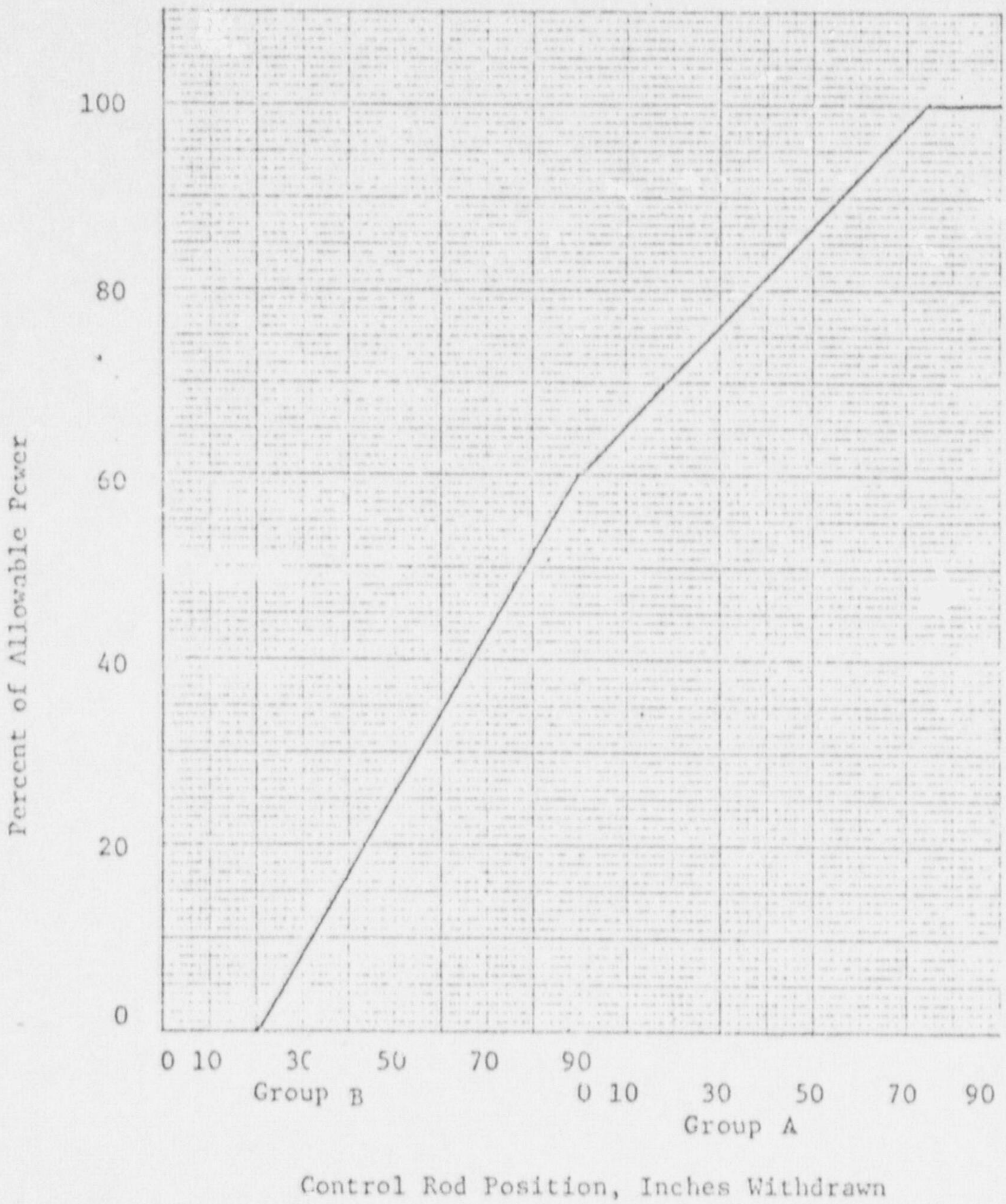
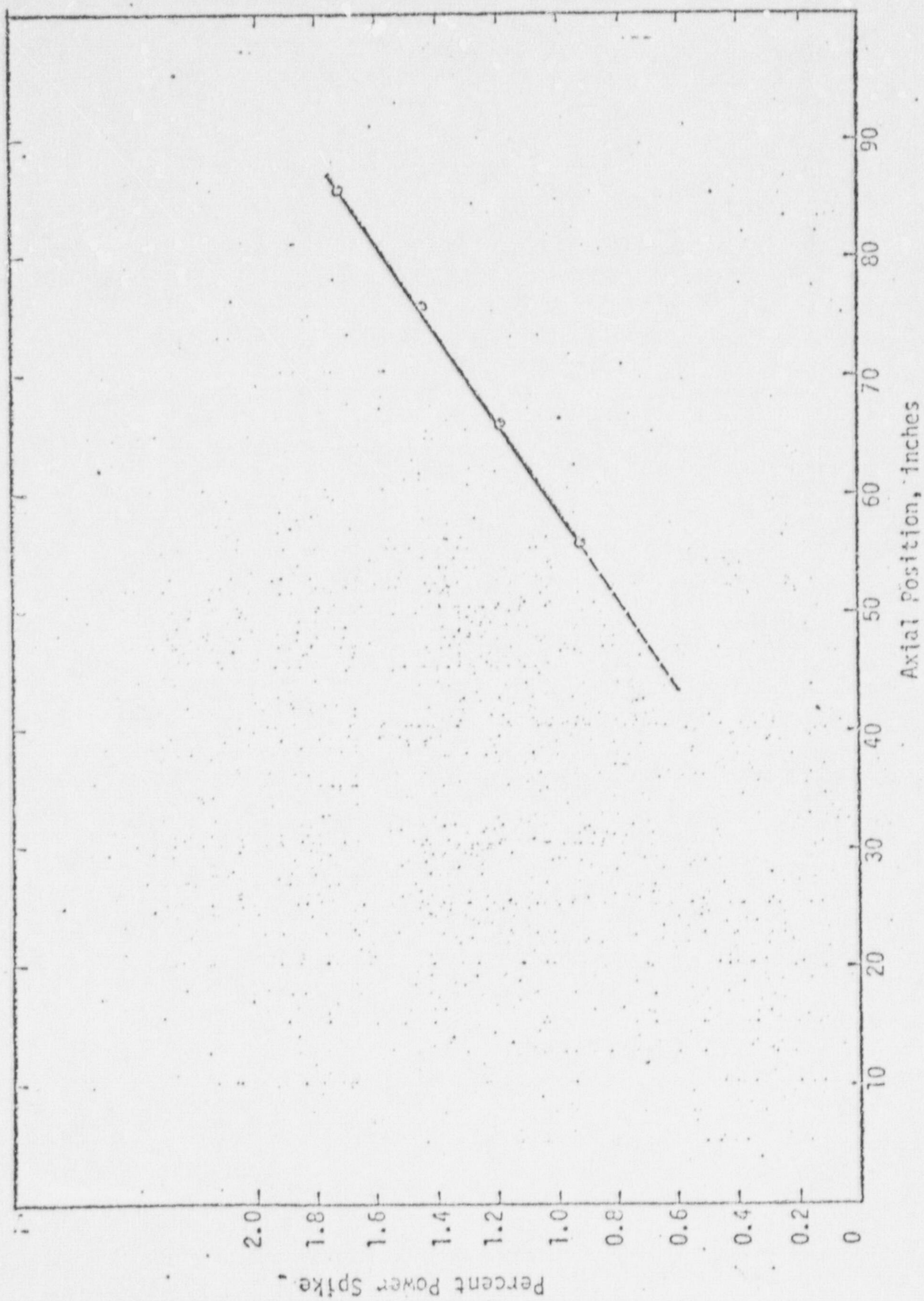


Figure 8-4

Flux Peaking Augmentation
Factor due to Gaps in
Non-Collapsed Fuel



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b. An unexplained increase of 45 cps in the air particulate monitor reading shall require bringing the plant to a hot standby condition to permit primary system inspection.

If both methods for determining main coolant leakage become inoperative and an hourly schedule of local sampling of vapor container air cannot be maintained, the plant will be brought to a hot standby condition.

c. Other Plant Protection

(1) The reactor shall be scrammed automatically, above 15 MW electric, when the turbine is tripped for any reason. The turbine shall be protected by all usual protective trips including high thrust bearing temperature, low bearing oil pressure, low condenser vacuum and overspeed.

(2) The reactor shall be scrammed automatically, above 15 MW electric when the generator is tripped for any reason. The generator shall be protected by all usual protective trips including overcurrent, differential and loss of field.

(3) Automatic initiation of the safety injection system, pumps and valves, shall be set to operate at a main coolant pressure not less than 1700 psig or a containment pressure not greater than 5 psig.

(4) The Low Pressure Safety Injection Accumulator shall have a minimum usable water volume of 700 ft^3 and a minimum nitrogen overpressure of 410 psig.

(5) The Safety Injection System which includes the Charging Pumps shall be maintained in readiness to inject borated water into the reactor at all times when the main coolant pressure is 1,000 psig or higher. Charging system readiness requires that two fixed speed pumps be available. Since the pumps are high maintenance items, a fixed speed charging pump may be out of service for eight hours before a pressure reduction to less than 1000 psig is required.

The following manually-controlled, electrically operated valves shall be in the position indicated during normal power operation.

Valve No.	Normal Position	
PU-MOV-541	Open	
PU-MOV-542	Open	
PU-MOV-543	Closed	
PU-MOV-544	Closed	
PU-MOV-545	Open	
PU-MOV-546	Open	
PU-MOV-547	Closed	
PU-MOV-548	Closed	1

The following manually-controlled, electrically operated valves shall have power disconnected (both series breakers in open position) to meet single failure requirements during normal power operation.

Valve No.	Position
CH-MOV-522	Closed
CH-MOV-523	Open
CH-MOV-524	Open

(6) The reactor shall be scrammed automatically, when the power level is above 15 MWe, by two or more low steam generator level signals.

ROUTINE MECHANICAL TESTS

Control Rod Scram - Circuit Check - During plant operation it is not advisable to make any tests in the scram circuitry. All indications that are available will be noted at least once per shift, and adjustments will be made only if necessary. At no time will the scram circuitry be disconnected during operation. All rod scram tests will be made during scheduled reactor shutdown or during various emergency shutdowns.

Safety Injection System - Monthly, during power operation, all of the pumps and active valves of the safety injection system will be operated individually from the safety injection control panel in the control room to determine their serviceability and correct light position indication. The pumps will be run to determine their starting capability only, because the loop fill and injection valves are open. Whenever the reactor plant is shutdown and depressurized, the entire system operation will be checked by manual operation of the safety injection control switch provided on the nuclear section of the main control board. Surveillance of the Safety Injection Actuation Signal (SIAS) Initiation Channels will include: (1) a comparison of the separate pressure indications by each shift; (2) a channel calibration each refueling by applying known pressures to each sensor; (3) a system functional test each refueling by application of test signals to each channel to verify system capability; and (4) a monthly operational check of the two containment air pressure switches. 12

Pressurizer Spray System - In order to assure reliable performance, the pressurizer surge and circulation spray systems will be tested periodically preferably during the normal startup operation. While maintaining normal operating conditions in the main coolant system and the pressure control and relief system, that is with surge spray deenergized, it is determined that correct circulation spray exists (hand control valve initially positioned so as to maintain pressurizer equilibrium conditions) by noting frequency of heater cycling while maintaining pressurizer at normal values. Verification of surge spray operation will be accomplished by obtaining maximum spray flow and observing the resultant pressurizer pressure decrease, temperature increase of water flowing through the surge line, and changes in pressurizer heater cycling.

Pressurizer Solenoid Relief Valve - The pressurizer solenoid relief valve, which is provided to limit the duty of the pressurizer safety valves, will be tested after refueling or after completion of maintenance on this valve.

The set pressure and the blowdown pressure shall be observed during the test operation and shall be compared with the design conditions. After the valve has discharged, note the downstream pipe temperatures to assure that the valve disc has properly reseated.

The steam pressure required to test the solenoid relief valve is obtained by operation of the pressurizer heaters.