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

MAINE YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-309

MAINE YANKEE ATOMIC POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-36

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

   A. The application for amendment by Maine Yankee Atomic Power Company (the licensee) dated December 5, 1979, as supplemented February 15, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformance with the applicatio[illegible], the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.B.(6)(b) of Facility Operating License No. DPR-36 is hereby amended to read as follows:

   (b) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 48 is hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-36

DOCKET NO. 50-309

Revise Appendix A as follows:

Remove Pages	Insert New Pages
2.1-1	2.1-1
2.1-4	2.1-4
2.1-5	2.1-5
2.1-6	2.1-6
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3
3.10-4	3.10-4
3.10-10	3.10-10
3.10-11	3.10-11
3.10-12	3.10-12
3.10-13	3.10-13
3.15-1	3.15-1

2.1 LIMITING SAFETY SYSTEM SETTING - REACTOR PROTECTION SYSTEM

Applicability: Applies to reactor trip settings and bypasses for the instrument channels monitoring the process variables which influence the safe operation of the plant.

Objective: To provide automatic protective action in the event that the process variables approach a safety limit.

Specification: The reactor protective system trip setting limits and bypasses for the required operable instrument channels shall be as follows:

2.1.1 Core Protection

a) Variable Nuclear Overpower

$\leq Q + 10$, or 106.5 (whichever is smaller) for $10 \leq Q \leq 100$
≤ 20 for $Q \leq 10$.

where

Q = Percent thermal or nuclear power, whichever is larger.

b) Thermal Margin/Low Pressure

$\geq A\, Q_{DNB} + BT_c + C$, or 1835 psig (whichever is larger)

where

T_c = cold leg temperature, ^oF
A = 2004.3
B = 17.9
C = -10053
$Q_{DNB} = A_1 \times QR_1$

A_1 and QR_1 are given in Figure 2.1-1a and 2.1-1b, respectively.

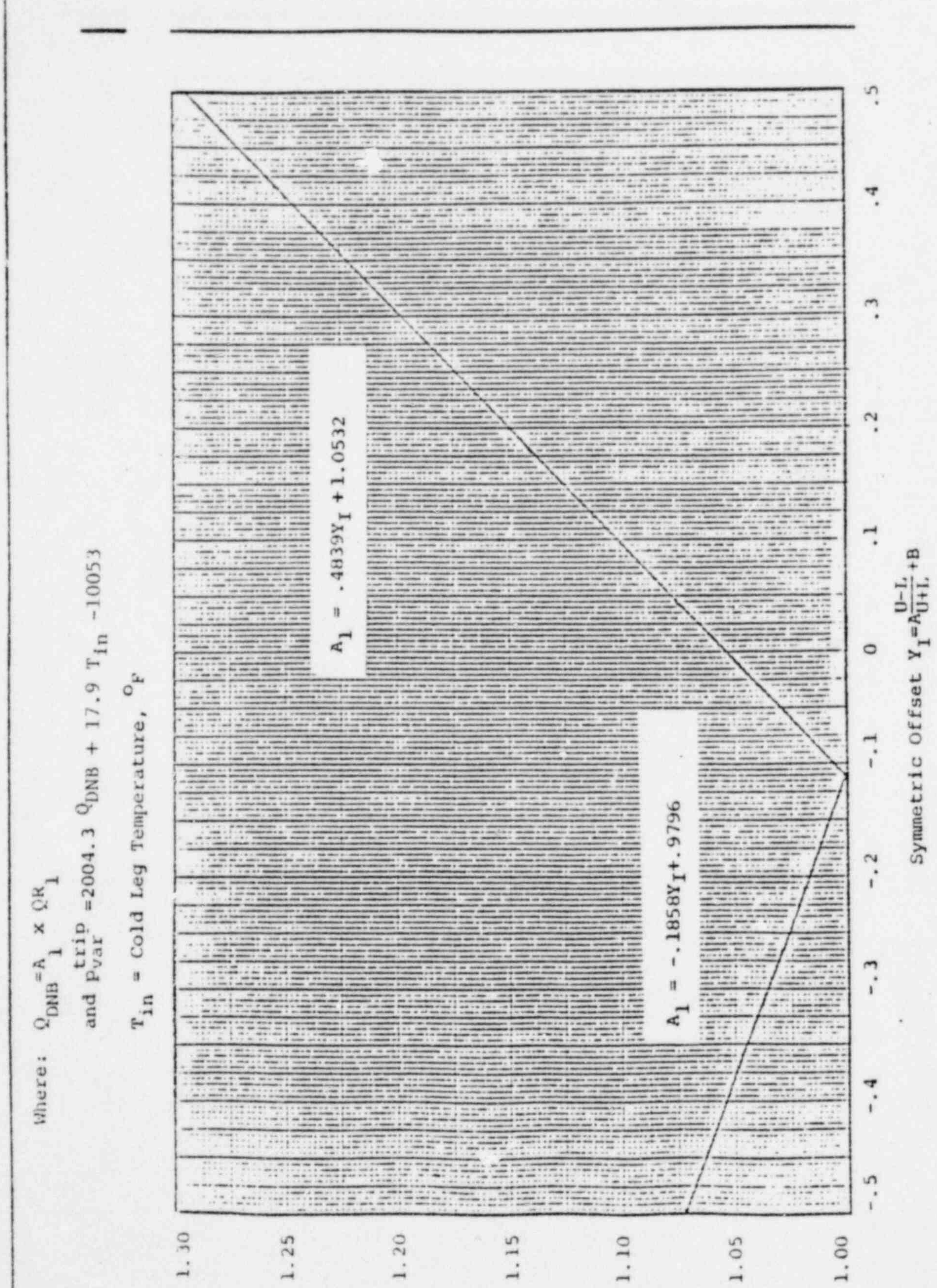
This trip may be bypassed below 10 percent of rated power.

c) The symmetric offset trip and pretrip function shall not exceed the limits shown in Figure 2.1-2, for three loop operation. This trip may be bypassed below 15 percent of rated power.

d) Low Reactor Coolant Flow

≥ 93 percent of 360,000 GPM (3 pump operation)

This trip may be bypassed below 2 percent of rated power.



MAINE YANKEE TECHNICAL SPECIFICATION	Thermal Margin/Low Pressure Trip Setpoint versus (Y_I versus A_I)	Figure 2.1-1a

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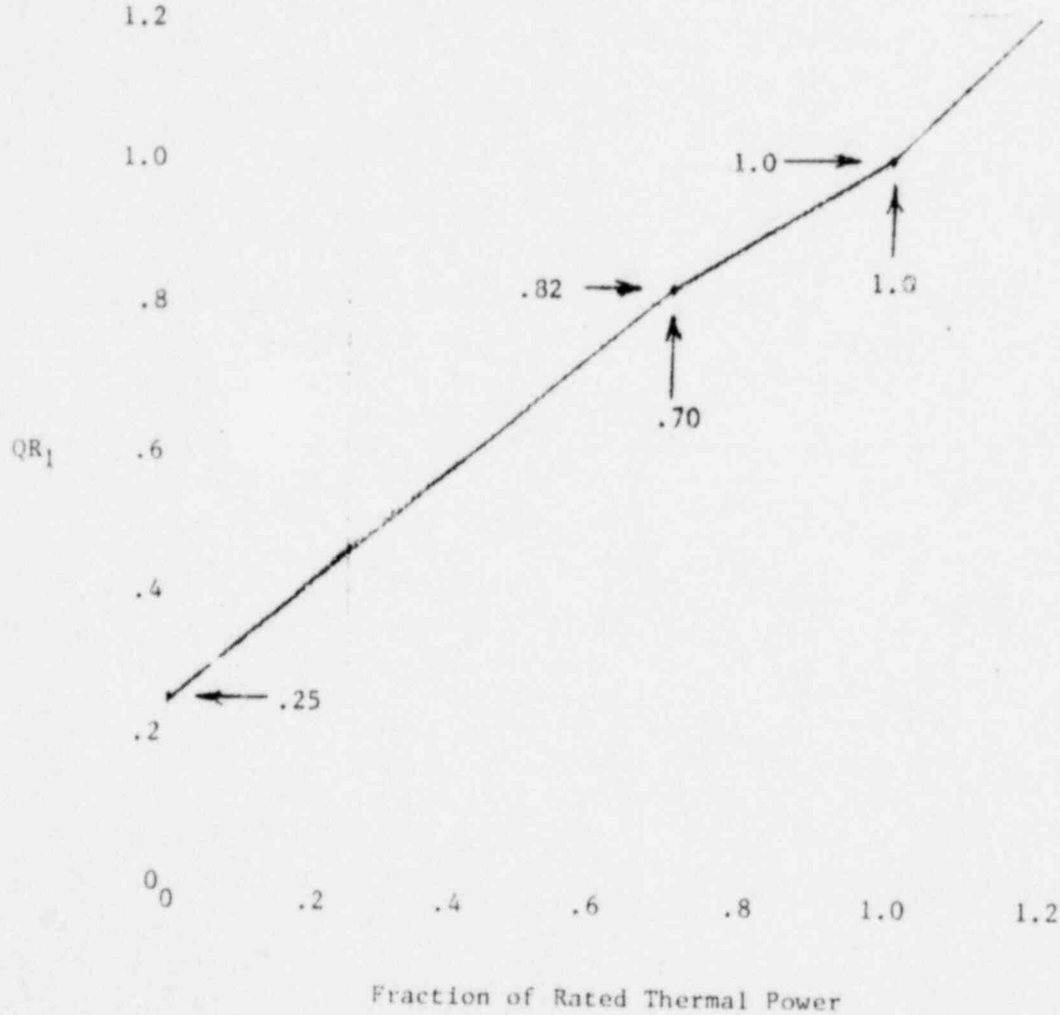
POOR ORIGINAL

Where: $A_1 \times QR_1 = Q_{DNB}$

and $P^{trip}_{var} = 2004.3\ Q_{DNB} + 17.9\ T_{in} - 10053$

T_{in} = Cold leg temperature



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MAINE YANKEE Technical Specifications	Thermal Margin/Low Pressure Trip Setpoint Part 2 (Fraction of Rated Thermal Power versus QR_1) 2.1-5	Figure 2.1-1b

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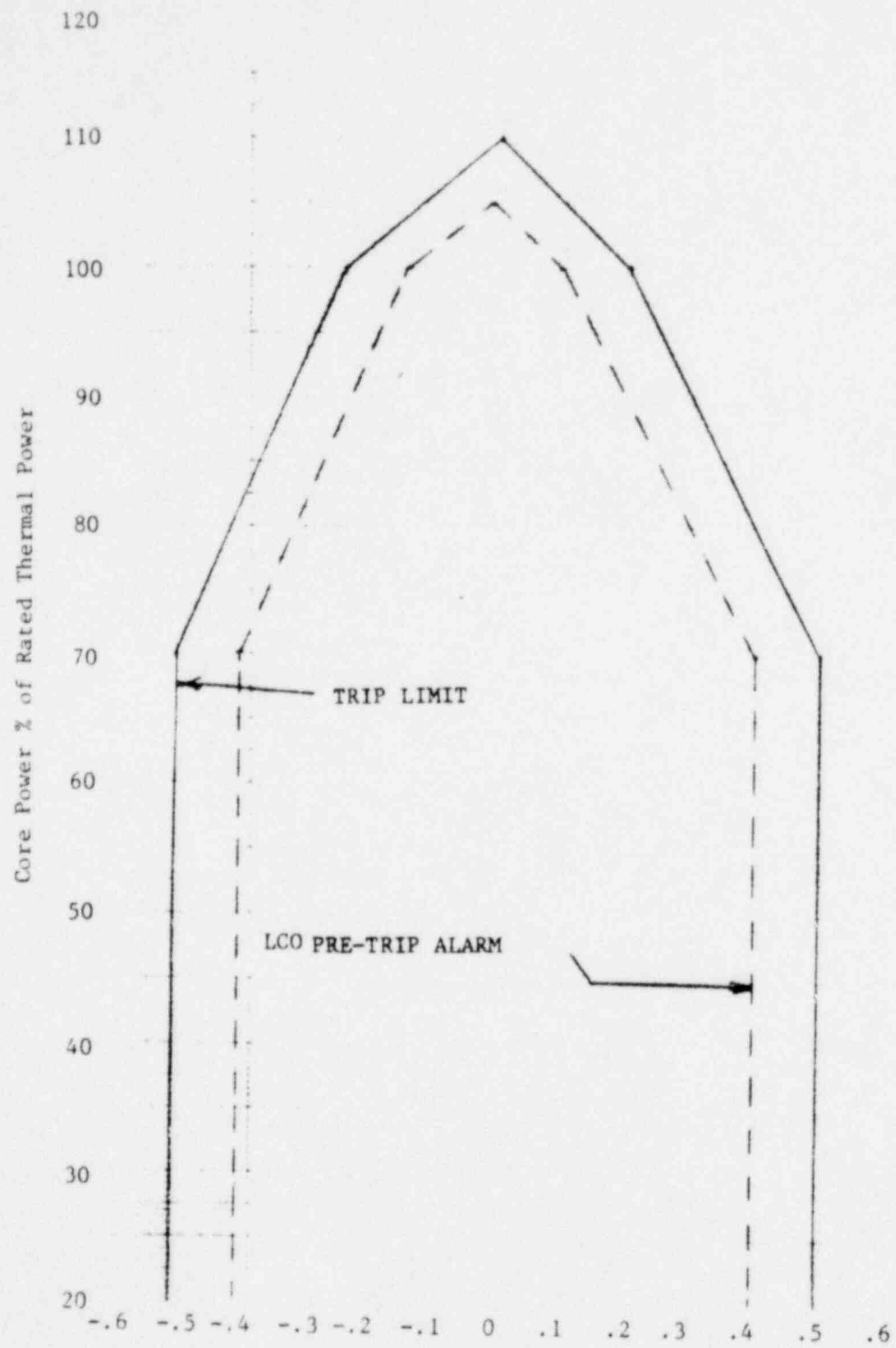

Symmetric Offset = $A\frac{U-L}{U+L} + B$

MAINE YANKEE Technical Specification	Symmetric Offset Function, Three Pump Operation 2.1-6	Figure 2.1-2

3.10 CEA GROUP, POWER DISTRIBUTION, MODERATOR TEMPERATURE COEFFICIENT LIMITS AND COOLANT CONDITIONS

Applicability: Applies to insertion of CEA groups and peak linear heat rate during operation.

Objective: To ensure (1) core subcriticality after a reactor trip, (2) limited potential reactivity insertions from a hypothetical CEA ejection, and (3) an acceptable core power distribution, moderator temperature coefficient, core inlet temperature, and reactor coolant system pressure during power operation.

Specification: A. CEA Insertion Limits

1. When the reactor is critical, except for physics tests and CEA exercises, the shutdown CEA's (Groups A, B and C) shall be fully withdrawn.*

2. When the reactor is critical, except for physics tests and CEA exercises, the regulating CEA groups (1 through 5) shall be no further inserted than the limits shown in Figure 3.10-1* for 3 loop operation.

3. When the reactor is critical, the available shutdown margin with one CEA stuck out will not be less than 3.2% in reactivity. During low power physics testing at the beginning of a cycle, CEA insertion is permitted such that the minimum shutdown margin is no less than 2% in reactivity.

4. Operation of the CEA's in the automatic mode is not permitted.

* NOTE - CEA's shall be considered fully withdrawn when positioned such that rods are inserted within 4 steps from their upper electrical limit.

B. Power Distribution Limits

1. The peak linear heat rate with appropriate consideration of normal flux peaking, measurement-calculational uncertainty (8%), engineering factor (3%), increase in linear heat rate due to axial fuel densification and thermal expansion (0.3% for Types E,G,H&I only) and power measurement uncertainty (2%) shall not exceed:

Type J: 13.5 kw/ft $\frac{X}{L} > 0.50$ and CAB $\leq$ 792 MWD/MTU

14 kw/ft $\frac{X}{L} > 0.50$ and CAB > 792 MWD/MTU

16 kw/ft $\frac{X}{L} \leq 0.50$

Types E,G,H,&I: 14.0 kw/ft $\frac{X}{L} > 0.50$

16.0 kw/ft $\frac{X}{L} \leq 0.50$

where $\frac{X}{L}$ is fraction of core height and CAB is cycle average burnup.

Should any of these limits be exceeded, immediate action will be taken to restore the linear heat rate to within the appropriate limit specifed above.

2. The total radial peaking factor, defined as $F_R^T = F_R^P (1+Tq)$, shall be evaluated at least once a month during power operation above 50% of rated full power.

2.1 F_R^P is the latest available unrodded radial peak determined from the incore monitoring system for a condition where all CEA's are at or above the 100% power insertion limit. T_q is given by the following expression:

$$T_q = 2\sqrt{\frac{(Pa-Pc)^2 + (Pb-Pd)^2}{(Pa+Pb + Pc+Pd)^2}}$$

Pi = relative quadrant power determined from incore system for quadrant i,

when the incore system is operable and by Specification 3.10.B.4 otherwise.

2.2 If the measured value of F_R^T exceeds the value given in Figure 3.10-4, perform one of the following within 24 hours.

a) Reduce symmetric offset pre-trip alarm and trip band (Figure 2.1-2), thermal margin/low pressure trip limit (Figure 2.1-1 and Tech. Spec. 2.1), and Excore LOCA monitoring limit (Figure 3.10-3) by a factor $\geq \frac{F_{R\ measured}^T}{F_R^T \text{ (Figure 3.10-4)}}$

or

b) Reduce THERMAL POWER at a rate of at least 1%/hour to bring the combination of THERMAL power and % increase in F_R^T to within the limits of Figure 3.10-5, while maintaining CEA's at or above the 100% power insertion limit; or

c) Be in at least HOT STANDBY.

3. Incore detector alarms shall be set at least weekly.

Alarms will be based on the latest power distribution obtained, so that the peak linear heat rate does not exceed the linear heat rate limit defined in Specification 3.10.B.1. 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 the alarm setpoint.

3.1 If the incore monitoring system becomes inoperable, perform one of the following within 4 E.F.P.H.

a) Initiate a power reduction to $\leq$P at a rate of at least 1%/hour where P(% of rated Power) is given by:

$$P = 0.85 \; \frac{\text{Linear heat rate permitted by Specification 3.10.B.1}}{\text{Latest measured peak linear heat rate corrected to 100\% Power}} \times 100$$

while maintaining CEA's above the 100% power insertion limit and monitor symmetric offset once a shift to insure that it remains within $\pm$ 0.05 of the value measured at the time when the above equation is evaluated. This procedure may be employed for up to 2 effective full power weeks, or

b) Comply with the alarm band given in Figure 3.10-3. If a power reduction is required, reduce power at a rate of at least 1%/hour.

4. The azimuthal power tilt, Tq, shall be determined prior to operation above 50% of full rated power after each refueling and at least once per day during operation above 50% of full rated power.

Tq is given by the following expression:

$$Tq = 2\sqrt{\frac{(Da-Dc)^2+(Db-Dd)^2}{(Da+Db+Dc+Dd)^2}}$$

Di = signal from excore detector channel i. Tq shall not exceed 0.03.

4.1 If the measured value of $T_q > 0.03$ but ≤ 0.10, or an excore channel is inoperable, assure that the total radial peaking factor (F_R^T) is within the provisions of Specification 3.10.B.2 once per shift.

4.2 If the measured value of T_q is > 0.10, operation may proceed for up to 4 hours as long as F_R^T is maintained within the provisions of Specification 3.10.B.2. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided:

a) The THERMAL POWER level is restricted to $\leq 20\%$ of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination, and

b) Reduce setpoints in accordance with Specification 3.10.B.2.2.

5. The incore detector system shall be used to confirm power distribution, such that the peaking assumed in the safety analysis is not exceeded, after initial fuel loading and after each fuel reloading, prior to operation of the plant at 50% of rated power.

6. If the core is operating above 50% of rated power with one excore nuclear channel out of service, then the azimuthal power tilt shall be determined once per shift by at least one of the following means:

a) Neutron detectors (at least 2 locations per quadrant).

b) Core-exit thermocouples (at least 2 thermocouples per quadrant).

7. The pre-trip limits of Figure 2.1-2 constitute Limiting Conditions of Operation.

C. CEA Drop Times

1. At operating temperature and 3 pump flow, the requirement for the maximum drop time of each CEA shall be not greater than 2.7 seconds from the time the holding coil is de-energized until the rod reaches 90% of its full insertion.

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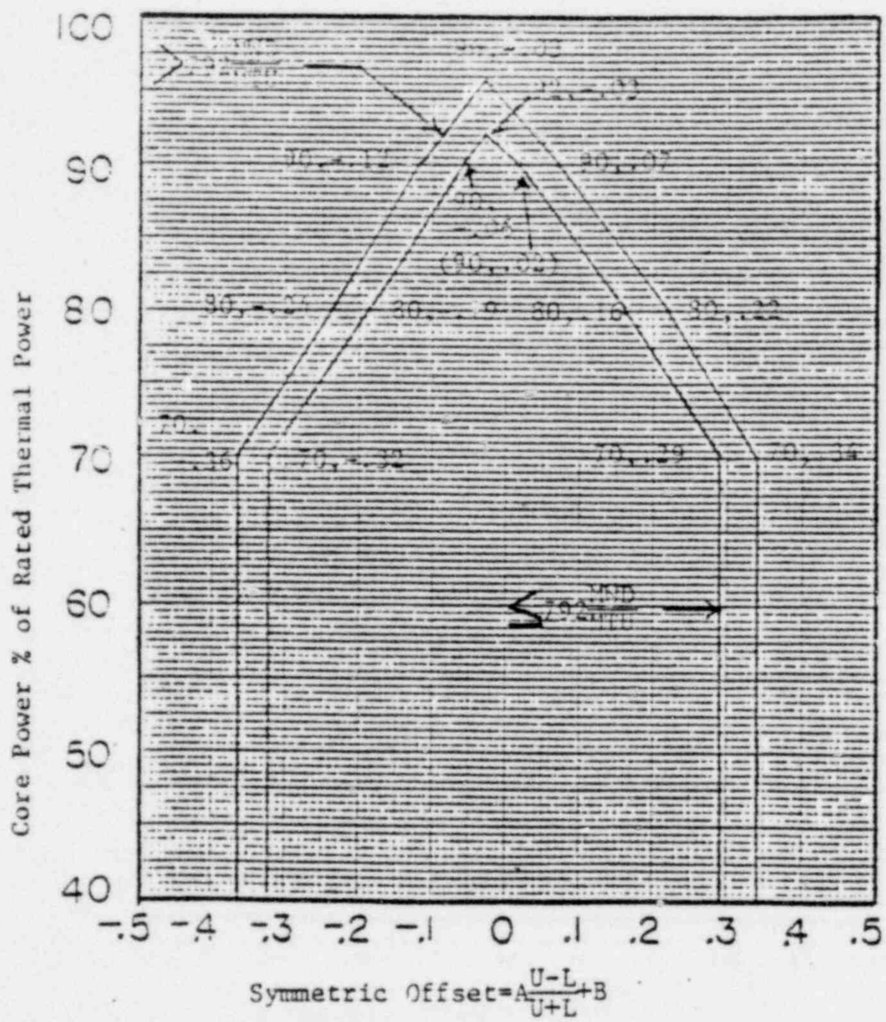
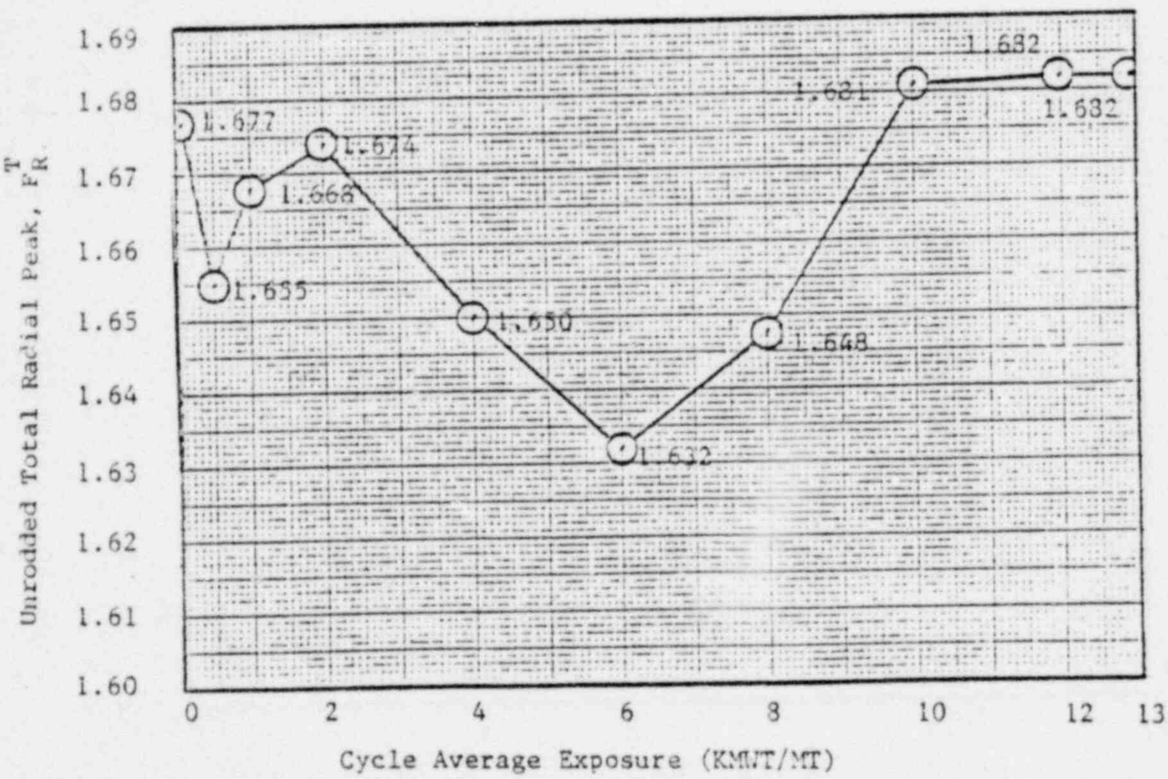
MAINE YANKEE Technical ° ecifications	Flux Peaking Augmentation Factors	Figure 3.10-2



MAINE YANKEE Technical Specifications	Excore Monitor S.O. Alarm Band for LOCA Limiting Conditions of Operation When Incore Monitors are Inoperable 3.10-11	Figure 3.10-3

Note: 1. This curve includes 10% calculational uncertainty
2. $F_R^T = F_R^P * 1.03$
3. Measured F_R^T should be augmented by measurement uncertainty (8%) before comparison to this curve.



MAINE YANKEE Technical Specifications	Allowable Unrodded Radial Peak Versus Cycle Average Burnup 3.10-12	Figure 3.10-4

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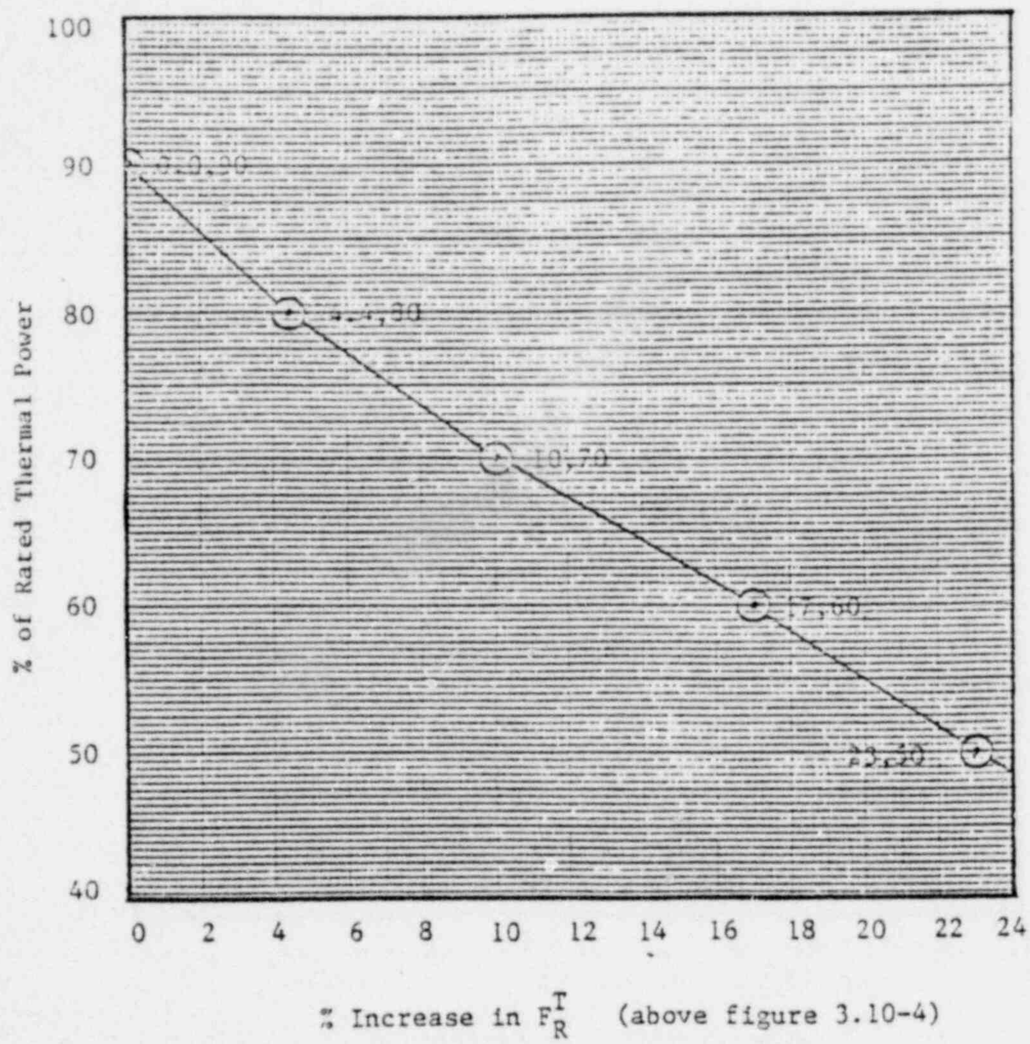
Note: CEA's are to be maintained at or above the 100% power insertion limit when applying 3.10.3.2.2b



MAINE YANKEE Technical Specifications	Allowable Power Level vs. Increase in Unrodded Total Radial Peak 3.10-13	Figure 3.10-5

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3.15 REACTIVITY ANOMALIES

Applicability: Applies to potential reactivity anomalies.

Objective: To require evaluation of reactivity anomalies within the reactor.

Specification: Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the reactor coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of 1% in reactivity, the Nuclear Regulatory Commission shall be notified and an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission in accordance with Technical Specification 5.9.1.6.

Basis: To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the CEA groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction and the reactivity status of the core can be continuously evaluated, and its occurrence would be thoroughly investigated and evaluated. The methods employed in calculating the reactivity of the core vs. burnup, and the reactivity worth of boron vs. burnup, are given in the FSAR.[1]

References: (1) FSAR, Section 3.4.7