ATTACHMENT A - DESCRIPTION OF PROPOSED CHANGES

Item No.	Technical Specification	Description of Change	Reason for Change
1.	Definitions	a) Add shutdown margin definition.	a) Provide consistent interpretation of shutdown margin (Section 4.9.5 of YAEC-1324).
		b) Remove "With all control rods in" from HOT and COLD SHUTDOWN BORON CONCENTRATION definitions.	b) Removes ambiguity in application to non-all-control-rods-in cases.
2.	1.3 page 1.3-1	a) Modified to reflect addition of new CEAs and trippable versus non-trippable status.	a) Reflects plant hardware changes (Section 3.1.5 and 5.7.1 of YAEC-1324).
		b) Maximum fuel enrichment increased from 3.03 to 3.30 weight percent.	b) Reflects present fuel management strategy (Section 4.1 of YAEC-1324).
3.	2.1 pages 2.1-1 through 2.1-5	a) Thermal margin/low pressure trip coefficients modified.	a) Reflects Cycle 7 power distributions.
		b) DNB SAFDL reference modified.	b) Reflects use of YAEC-1 DNB SAFDL to generate LSSS (Section 6.0 of YAEC-1324).
		c) Figures 2.1-1a and 2.1-1b modified.	c) Reflects Cycle 7 power distributions and YAEC-1 DNB SAFDL (Section 6.0 of YAEC-1324).
4.	2.2 pages 2.2-1 and 2.2-2	a) YAEC-1 DNB SAFDL added as a Safety Limit.	a & b) YAEC-1 CHF correlation used as SAFDL in generation of LSSS (Section 6.0 of YAEC-1324).
		b) Reference for YAEC-1 CHF correlation added.	
		c) Steady-state peak linear heat rate of 20 kW/ft specified for Type J fuel.	c) Required fuel centerline melt SAFDL for Type J (Section 3.2.2 of YAEC-1324).

Item No.	Technical Specification	Description of Change	Reason for Change
5.	3.3 pages 3.3-1 and 3.3-2	a) 3.3.A.2 modified to require three reactor coolant pumps operating whenever the reactor is critical.	a) Consistency with reference safety analysis.
		b) 3.3.A.3 merged with 3.3.A.2	b) Clarity.
		c) Basis clarified.	c) Removes ambiguity.
6.	3.10 pages 3.10-1 through 3.10-13 and 3.10-15	a) 3.10.A.1 defines CEA Group 5 as consisting of 2 CEA subgroups.	a) Reflects plant hardware changes (Sections 3.1.5 and 5.7.1 of YAEC-1324).
		b) 3.10.A.2 modified to remove requirement to maintain CEAs at upper electrical limit when RCS boron concentration is less than 100 ppm.	b) Increased available scram reactivity sufficient to allow removal (Section 4.9.4 of YAEC-1324).
		c) 3.10.A.3 defines misaligned CEA in terms of steps misaligned.	c) Moved from 3.10.D.1 for clarity and a change from eight inches to 10 steps since steps are indicated on control panel.
		d) 3.10.A.3 defines misaligned CEA subgroups for CEA Group 5.	d) Consistent with assumptions in RPS setpoint analysis to accommodate CEA Group 5 movement (Section 3.1.5 of YAEC-1324).
		e) 3.10.A.3.a defines power level and CEA restrictions in the event of a misaligned CEA.	e) Moved from 3.10.D.4.a and changed to be consistent with dropped CEA analysis and bounded by RPS setpoint analysis assumptions (Section 4.9.3.2 of YAEC-1324). Distinction between misaligned CEA and dropped CEA eliminated for clarity.

Item No.	Technical Specification	Description of Change	Reason for Change
		f) 3.10.A.3.b defines power distribution limits verification after CEA realignment.	f) Moved from 3.10.D.4.b for clarity.
		g) 3.10.A.4 moved from 3.10.D.1.	g) No change in wording. Moved to a more appropriate section.
		h) Shutdown margin limits Section 3.10.B created from 3.10.A.3, 3.10.C, 3.10.D.2 and 3.10.D.3.	h) For clarity, all shutdown margin related specifications are moved to one section
		i) 3.10.B.2 and 3.10.B.3 add the word "trippable" to inoperable or slow CEA definitions.	i) CEAs in former part-length CEA locations are non-trippable and should be excluded from these requirements (Section 3.1.5 of YAEC-1324).
		j) 3.10.B.3 incorporates 3.10.C and 3.10.D.3.	j) Clarity in defining a slow CEA.
		k) 3.10.C.1 applies thermal expansion factor to Type E fuel only for power distribution calculations (CE design fuel).	k) Reflects fuel types resident in core for Cycle 7 (Section 3.1.1 of YAEC-1324).
		l) 3.10.C.2.3 specifies HOT SHUTDOWN as non-operating mode in event of exceeding F_R^T limits.	l) Exceeding F_R^T by an amount greater than can be accounted for in 3.10.C.2.1 and 3.10.C.2.2 is outside bounds of safety analysis therefore, plant should be placed in a subcritical condition.
		m) 3.10.C.3.1.2 expanded to 3.10.C.3.1.2 and 3.10.C.3.1.3 to include required power level if CEAs are restricted to 100% power insertion limit.	m) Provide greater operating flexibility in the event the Incore Monitoring System becomes inoperable.

Item No.	Technical Specification	Description of Change	Reason for Change
		n) 3.10.D specifies MTC restriction.	n) Consistent with safety analysis assumptions (Section 5.1.3 of YAEC-1324).
		o) 3.10.E.1 modified to include 3.10.F.1 and 3.10.F.3.	o) Reflects assumptions in safety analysis for critical conditions.
		p) 3.10.E.2 modified from 3.10.F.2.	p) Reflects assumptions in safety analysis for critical conditions.
		q) 3.10.E.2 exception for natural circulation test added to this section.	q) Consistent with existing exception in 3.3.A.2.
		r) Bases rearranged to follow order of specifications.	r) No change in wording. Rearranged for clarity.
		s) Bases discussion on misaligned CEA changed.	s) Eight inches changed to 10 steps. Misaligned CEA restriction basis provided.
		t) Bases discussion on shutdown margin changed.	t) Clarification of shutdown margin.
		u) Bases discussion on natural circulation exception added.	u) Clarification of exception.
		v) Figure 3.10-1 modified.	v) Reflects Cycle 7 CEA insertion limits (Section 4.9.1 of YAEC-1324).
		w) Figures 3.10-2 and 3.10-3 modified.	w) Reflects Cycle 7 power distributions.
		x) Figure 3.10-4 modified.	x) Reflects Cycle 7 radial peaking (Section 4.3 of YAEC-1324).
		y) Figure 3.10-5 modified.	y) Reflects Cycle 7 power distributions and RPS setpoints.
		z) Figure 3.10-7 modified.	z) Reflects Cycle 7 required shutdown margin (Section 4.9.5 of YAEC-1324). Provides power level and RCS boron concentration dependence consistent with shutdown margin in definition section.

ATTACHMENT B -TECHNICAL SPECIFICATION CHANGES

REACTOR STATUS

Refueling Boron Concentration

The boron concentration shall be sufficient to maintain the reactor at least 5% delta k/k subcritical under all refueling conditions.

REACTOR STATUS (Continued)

Cold Shutdown Boron Concentration

The boron concentration shall be sufficient to maintain the reactor at least 5% delta k/k subcritical.]]

Hot Shutdown Boron Concentration

The boron concentration shall be sufficient to maintain the reactor at least 5% delta k/k subcritical.]]

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power. The reactor is considered subcritical when it is not critical.

Shutdown Margin]]

Shutdown margin shall be the sum of:]]

(1) the reactivity by which the reactor is subcritical in its present condition, and]]
(2) the reactivity associated with the withdrawn trippable CEAs less the reactivity associated with the highest worth withdrawn trippable CEA.]]

Low Power Physics Testing

Testing performed under approved written procedures to determine control rod worths and other core nuclear properties. Reactor power during these tests shall not exceed 2% of rated power, not including decay heat, and primary system temperature and pressure shall be in the range of 260°F to 550°F and 415 psia to 2300 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions in these Technical Specifications.

Power Range Physics Testing

Tests performed under approved written procedures to verify core nuclear design properties at power and plant response characteristics. Reactor power may be greater than 2% during these measurements. Primary system average temperature and pressure shall be in the range of 500°F to 580°F and between 1700 psia to 2300 psia, respectively. Certain deviations from normal operating practices which are necessary to enable the performance of some of these tests are permitted in accordance with specific provisions of these Technical Specifications.

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Rated Power

A steady-state reactor core output of 2630 MWt.

Quadrant Power Tilt

The difference between nuclear power in any core quadrant and the average in all quadrants.

$$TILT = \left[\frac{\text{Power in any quad}}{\text{Avg power of all quad}}\right] - 1$$

REACTOR PROTECTIVE SYSTEM

Instrument Channels

One of four independent measurement channels, complete with the sensors, sensor power supply units, amplifiers, and trip modules provided for each safety parameter.

Reactor Trip

The de-energizing of the magnetic jack holding coils which releases the shutdown and regulating control elements (CEA's) and allows them to drop into the core.

Trip Module

A bistable unit in each of the instrument channels which is tripped when the parameter signal exceeds a specified limit. The relay contact outputs of the trip modules form the reactor protective system logic.

ENGINEERED SAFEGUARDS SYSTEMS

Subsystem

One of two or more redundant grouping of sensors, logic, and circuitry able to bring about automatic or manual initiation of an engineered safeguard.

Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

INSTRUMENTATION SURVEILLANCE

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. This determination shall, where feasible, include comparison of the channel with other independent channels measuring the same variable.

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Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration/Channel Adjustment

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, interlocks or trip and shall include the channel functional test.

MISCELLANEOUS DEFINITIONS

Operable

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified functions(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Operating

A system or component is operating if it is performing its safeguard or operating functions.

Control Element Assemblies

All full-length shutdown and regulating control element assemblies (CEA's).

Partial-Length Control Element Assemblies

Control element assemblies (CEA) that contain neutron absorbing material only in the lower quarter of their length.

Containment Integrity

Containment integrity is defined to exist when all of the following are true:

a. All non-automatic containment isolation valves and blind flanges are closed.

b. The equipment hatch is properly closed and sealed.

c. At least one hatch in the personnel air lock is properly closed and sealed.

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d. All automatic containment isolation valves are operable or are locked closed.

e. The uncontrolled containment leakage satisfies Specification 4.4 Section I.B.3.

Fire Suppression Water System

A fire suppression water system shall consist of: A water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

MISCELLANEOUS DEFINITIONS (Continued)

Radioisotope Release Limits

The Maine Yankee radioisotope release limits are as defined in Technical Specification 3.16, paragraph A, item 2, for liquid releases and Technical Specification 3.17, paragraph A, item 2 for gaseous releases.

E - Average Disintegration Energy

E is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in Mev) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

Dose Equivalent I-131

Dose Equivalent I-131 is determined as that concentration of I-131 (micro Ci/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

Reportable Occurrence

A reportable occurrence is defined in Section 5.9 of these specifications.

Remedial Action

Remedial Action is that part of a specification which prescribes corrective and/or compensatory measures required under designated conditions.

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Noncompliance

Noncompliance with a Limiting Condition for Operation shall exist when neither the requirements of the Limiting Condition for Operation nor the associated Remedial Action (if any) are met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Remedial Action requirements is not required.

Nonconformance

Nonconformance with a specification shall exist when the requirements of the Limiting Condition for Operation are not met without reliance upon Remedial Action statements.

Frequency Notation

The frequency notation specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 0.1.

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TABLE 0.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days if the plant is in the cold shutdown condition
SA	At least once per 6 months
A	At least once per year
R	At least once per 18 months
P	Prior to each reactor startup
N.A.	Not applicable

1.3 REACTOR

Applicability

Applies to the reactor vessel, vessel core and internals, as well as the Reactor Coolant System and components, including associated Emergency Core Cooling Systems.

Objectives

To define those design criteria essential in providing for safe system operation which are not covered in Sections 2 and 3.

Specification

A. Reactor Core

The reactor core shall contain 217 fuel assemblies with each assembly containing 176 rods. Each fuel rod clad with Zircaloy-4 shall have a nominal active fuel length of 136.7 inches. The fuel shall have a maximum nominal enrichment of 3.30 weight percent U-235.]]]]

The core excess reactivity shall be controlled by a combination of boric acid chemical shim, Control Element Assemblies (CEAs) and mechanically fixed non-fuel rods when required. The non-fuel rods may be fixed alumina-boron carbide, solid metal or open tubes.]]]]

There are a total of eighty-one (81) full-length CEAs provided. Forty (40) of these are paired to form twenty (20) dual CEAs. Seventy-seven (77) CEAs, including all dual CEAs, are trippable. Four (4) of the CEAs are non-trippable.]]]]]]]]

There are two types of full-length CEAs. Seventy-seven (77) of the full-length CEAs contain B_4C pellets over 124 inches of their length and silver-indium-cadmium in their lowest eight (8) inches. Four (4) of the full-length CEAs have 3 CEA fingers that contain stainless steel as the absorber material, while the remaining 2 CEA fingers contain B_4C pellets.]]]]]]]]]]

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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:

Less than or equal to Q + 10, or 106.5 (whichever is smaller) for Q greater than or equal to 10 and less than or equal to 100, and less than or equal to 20 for Q less than or equal to 10.

Where

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

b) Thermal Margin/Low Pressure:

Greater than or equal to: $A\ Q_{DNB} + BT_C + C$, or 1835 psig (whichever is larger).

Where

T_C = cold leg temperature, ^oF
A = 1998.0]]
B = 17.9
C = -10053
$Q_{DNB} = A_1 \times QR_1$

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

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

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d) Low Reactor Coolant Flow:

Greater than or equal to 93% of 360,000 GPM (3 pump operation).

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

2.1.2 Other Reactor Trips

a) High Pressurizer Pressure:

Less than or equal to 2385 psig.

b) High Containment Pressure:

Less than or equal to 5 psig.

c) Low Steam Generator Pressure:

Greater than or equal to 485 psig.

This trip may be bypassed when the steam generator pressure is less than 100 psi above the setpoint.

d) Low Steam Generator Water Level:

At or above the centerline of the feedwater ring (5'-0" below normal water level).

Basis

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from those reactivity excursions too rapid to result in a high pressure or thermal margin trip. The prescribed setpoint, with allowance for errors, is conservative relative to the trip point used in the accident analysis.

The high rate-of-change of power reactor trip does not appear in the Specification as this trip is not used in the transient and accident analysis. This trip provides protection during reactor startup and is set at 2.6 decades per minute.

The thermal margin/low pressure trip is provided to prevent the fuel from exceeding thermal criteria[1]. The low setpoint of 1835 psig trips the]]
reactor in the unlikely event of a loss-of-coolant accident.

The symmetric offset trip is provided to assure that excessive axial peaking will not cause fuel damage. The symmetric offset is determined from the axially split excore detectors. The symmetric offset trip and pre-trip, in conjunction with the thermal margin/low pressure trip, assure that neither the DNB SAFDL[1] nor the maximum linear heat rate SAFDL[1] exist as a]]
consequence of axial power maldistributions.]]

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These coefficients were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies, and the uncertainty associated with the excore to incore symmetric offset relationship.

The low reactor coolant flow trip protects the core against DNB should the coolant flow suddenly decrease significantly. The setpoint specified is consistent with the value assumed in the safety analysis.

The 2385 psig high pressurizer pressure reactor trip has been used as a limit on the upper pressure range of the thermal margin/low pressure trip. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig).

A reactor trip on high containment pressure is provided to assure that the reactor is shut down upon the initiation of safety injection. This trip is also a backup to the thermal margin/low pressure trip for the unlikely event of a loss-of-coolant accident.

The low steam generator pressure trip is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.

The low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generator at the time of reactor trip to provide 13 minutes of margin for initiation of the Auxiliary Feedwater System.]]

Reference

(1) Technical Specification 2.2.]]

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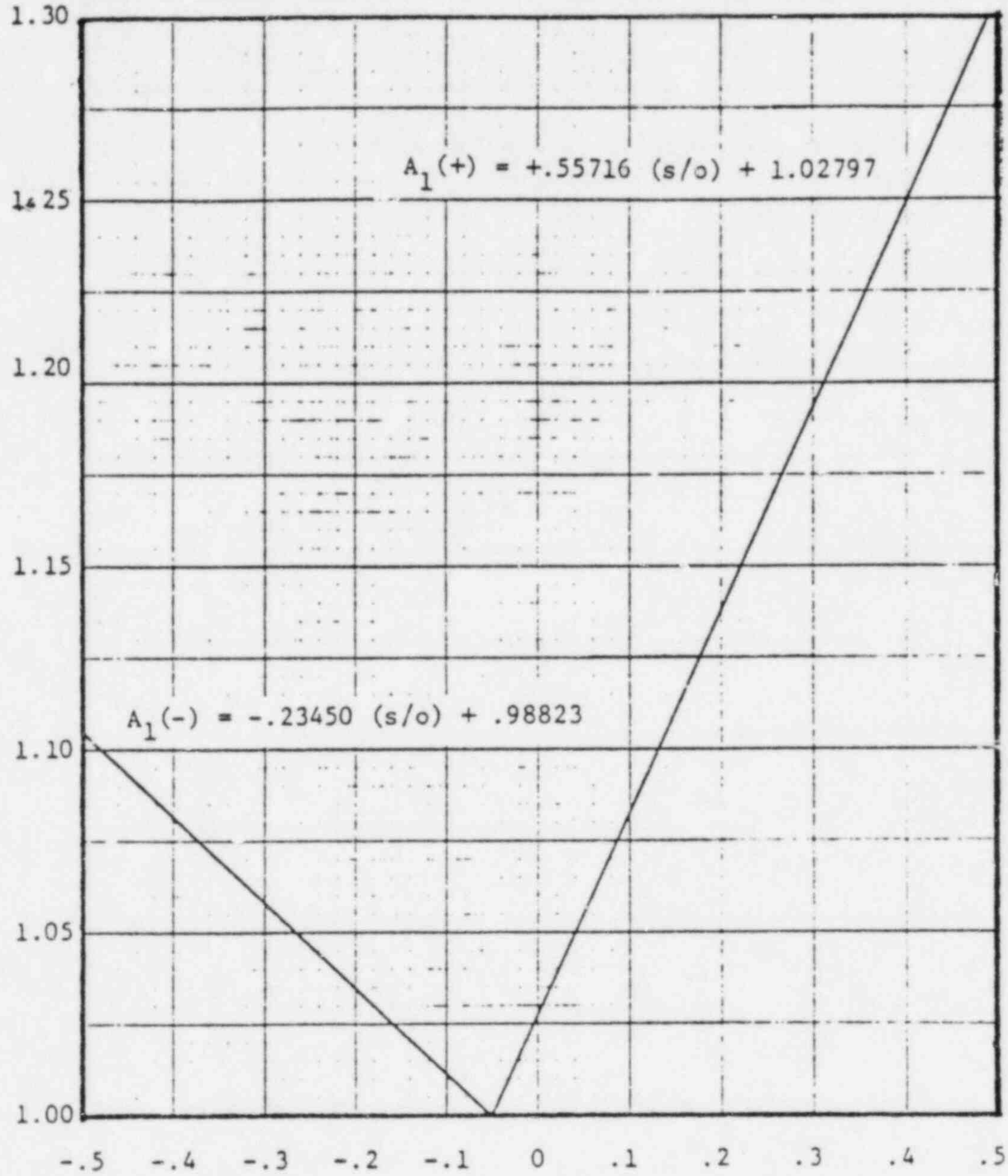
Where: $Q_{DNB} = A_1 \times QR_1$

and $P_{var}^{trip} = 1998.0\ Q_{DNB} + 17.9\ T_{in} - 10053$

T_{in} = Cold leg temperature, ^oF



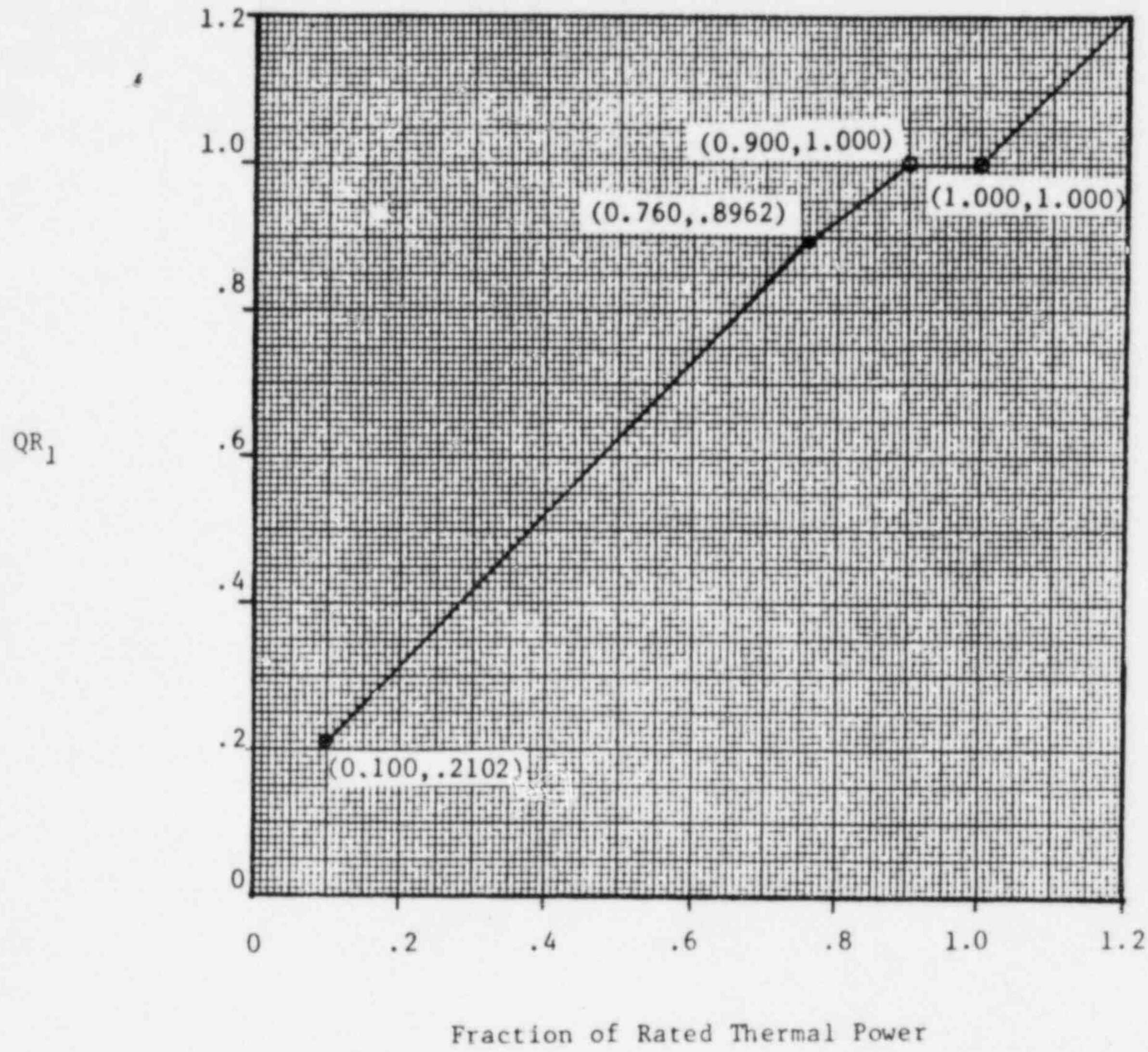
Symmetric Offset $Y_I + A\ \frac{U-L}{U+L} + B$

MAINE YANKEE Technical Specification	Thermal Margin/Low Pressure Trip Setpoint Versus (Y_I versus A_1) 2.1-4	Figure 2.1-1a SEP 29 1982

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

and $P_{var}^{trip} = 1998.0\ Q_{DNB} + 17.9\ T_{in} - 10053$

T_{in} = Cold leg temperature



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 SEP 29 1982

2.2 SAFETY LIMITS - REACTOR CORE

Applicability

Applies to the limiting combinations of reactor power, and Reactor Coolant System flow, temperature, and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specifications

A. The reactor and the Reactor Protection System shall be operated such that the Specified Acceptable Fuel Design Limit (SAFDL) on the departure from nucleate boiling heat flux ratio (DNBR):]]

DNBR = 1.17 using the YAEC-1 DNB heat flux correlation,]]

or]]

DNBR = 1.30 using the W-3 DNB heat flux correlation]]

is not exceeded during normal operation and anticipated operational occurrences.]]

B. The reactor and the Reactor Protection System shall be operated such that the SAFDLs for prevention of fuel centerline melting:]]

A steady-state peak linear heat rate equal to:]]

21 kW/ft for Types E, K, and L fuel,]]

and]]

20 kW/ft for Type J fuel,]]

are not exceeded during normal operation and anticipated operational occurrences.]]

Basis

To maintain the integrity of the fuel cladding, thus preventing fission product release to the Primary System, it is necessary to prevent overheating of the cladding. This is accomplished by operating within the nucleate boiling regime of heat transfer, and with a peak linear heat rate that will not cause fuel centerline melting in any fuel rod. First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "Departure from Nucleate Boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperature and the possibility of cladding failure.

The correlations listed predict DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during anticipated operational occurrences is limited to 1.17 for the YAEC-1 correlation or 1.30]] for the W-3 correlation. The use of either is acceptable as a fuel design]] limit to avoid DNB[1],[2].]]

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the determined values of the peak linear heat rates which would not cause fuel centerline melting for the various fuel types resident in the core are established as specified fuel design limits.

Limiting safety system settings for the TM/LP, symmetric offset trips, and limiting conditions for operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

References

(1) FSAR Section 3.5.
(2) J. Handschuh, DNBR Limit Methodology and Application to the Maine Yankee]]
Plant, YAEC-1296P, January, 1982.]]

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3.3 REACTOR COOLING SYSTEM OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of the Reactor Coolant System equipment.

Objective

To specify conditions of Reactor Coolant System components for reactor operation.

Specification

A. Reactor Coolant Pumps

1. At least one reactor coolant pump or one low pressure safety injection pump operating in the residual heat removal mode shall be in operation providing flow through the reactor when the Reactor Coolant System boron concentration is being reduced.

2. At least three reactor coolant pumps shall be in operation providing flow through the core with their steam generators capable of performing their heat transfer function whenever the reactor is in a critical condition.

   Exception

   The requirements of 3.3.A.2 may be modified during initial testing to permit power levels not to exceed 10% of rated power with three loops operating on natural circulation.

B. Pressurizer Safety and Relief Valves

1. At least one pressurizer code safety valve shall be operable whenever fuel is in the reactor, and the Reactor Coolant System is isolated from the Residual Heat Removal System and the head is on the vessel.

2. At least two pressurizer code safety valves shall be operable whenever the reactor is critical.

3. One Power-Operated Relief Valve (PORV) and its associated block valve shall be operable whenever the Reactor Coolant System temperature is greater than 210°F.

4. In the event either PORV or its associated block valve becomes inoperable, within six hours: either restore the PORV or block valve to operable status, or close and remove power from the associated block valve.

C. Pressurizer

1. The pressurizer shall be operable with at least one bank of proportional heaters and a water level during normal system operation between 28% and 60% whenever the Reactor Coolant System Tave is greater than 500°F.

2. The pressurizer spray system must be lined up to provide continuous pressurizer spray flow whenever the reactor is critical.

Basis

Reactor coolant pump flow and steam generator heat transfer capabilities are specified to assure adequate core heat transfer capability under all operating conditions from criticality to full power. Three loop operation is specified to assure plant operation is restricted to conditions considered in the safety]] analyses.

The exception permits testing to determine decay heat removal capabilities of]] the Primary System while on natural circulation, prior to operation at higher]] power.

Following a loss of off-site power, stored and decay heat from the reactor would normally be removed by natural circulation using the steam generators as the heat sink. Water supply to the steam generators is maintained by the Auxiliary Feedwater System. Natural circulation cooling of the Primary System requires the use of the pressurizer heaters or high pressure safety injection pumps to maintain a suitable overpressure on the Reactor Coolant System. Alternatively, in the event that natural circulation in the Reactor Coolant System is interrupted, the feed and bleed mode of Reactor Coolant System operation can be used to remove decay heat from the reactor. This method of decay heat removal requires the use of the Emergency Core Cooling System (ECCS) and the Power-Operated Relief Valves (PORVs) in the pressurizer.

The PORVs can be operated either manually or automatically in the Maine Yankee design. Block valves are provided upstream of the relief valves to isolate the valve in the event that a PORV fails.

When reactor coolant boron concentration is being reduced, the process must be uniform throughout the Reactor Coolant System volume to prevent stratification of reactor coolant at a lower boron concentration which could result in a reactivity insertion.

Sufficient mixing of the reactor coolant is assured by one Low Pressure Safety Injection (LPSI) pump operating in the RHR mode. When operated in this mode it will circulate the Reactor Coolant System volume in less than 12 minutes. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the Reactor Coolant System during a dilution operation. A continuous pressurizer spray flow will maintain a nominal spread between the boron concentration in the pressurizer and the Reactor Coolant System during the addition of boron. Without residual heat removal, the amount of steam which could be generated at safety valve lift pressure with the reactor subcritical would be less than half of the valve's capacity. One valve, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Overpressure protection is provided for all critical conditions. The safety valves are sized to relieve steam at a rate equivalent to the peak volumetric pressure surge rate. For this purpose one safety valve is sufficient; however, a minimum of two safety valves is required by Section III of the ASME Code.

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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 Operational 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 and the regulating CEA's (Groups 1 through 5) shall be no further inserted than the limits shown in Figure 3.10-1 for 3 loop operation. CEA Group 5 consists of two subgroups designated Subgroup 5A and 5B.

2. A CEA is considered fully withdrawn if the CEA is withdrawn to 4 steps or less from its upper electrical limit.

3. Except during physics testing, a CEA misalignment is considered to be any one of the following:

   - A CEA in Group A, B, C, 1, 2, 3, or 4 that is out of position from the remainder of the group by more than 10 steps.

   - A CEA in Subgroup 5A or 5B that is out of position from the remainder of the subgroup by more than 10 steps.

   - The indicated subgroup positions of Subgroup 5A and 5B differ by more than 15 steps.

   If a CEA misalignment is not corrected within 15 minutes, operation with a CEA misalignment is permitted for a period of 4 hours provided:

   a. Thermal power is reduced by at least 10% of rated power within one-half hour and by at least 20% of rated power within one hour of identification of the misalignment. The CEA insertion limits specified for the initial thermal power must be maintained.

b. Within two hours after realignment, the peak linear heat rate will be shown to be within the limits specified in 3.10.C.1 and the total radial peaking factor will be shown to be within the limits specified in 3.10.C.3 using the latest unrodded radial peaking factor.]]

4. If the CEA deviation alarms from both the computer pulse counting system and the reed switch indication system are not available, individual CEA positions shall be logged and misalignment checked every 4 hours.

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

B. Shutdown Margin Limits]]

1. When the reactor is critical, the shutdown margin will not be less than that shown in Figure 3.10-7, except during low power]] physics tests when the shutdown margin will not be less than 2%]] in reactivity.

2. A trippable CEA is considered inoperable if it cannot be tripped.]] A CEA that cannot be driven shall be assumed not able to be tripped until it is proven that it can be tripped. Operation with an inoperable CEA is permitted provided:

   a. The shutdown margin specified in 3.10.B.1 is satisfied without]] the reactivity associated with the inoperable CEA within 2 hours of identification of the inoperable CEA.

   b. Except for low power physics tests and CEA exercises, only one CEA is inoperable.

3. A trippable CEA is considered to be a slow CEA if the drop time from de-energizing its holding coil to reaching 90% of its full insertion exceeds 2.7 seconds at operating temperature and 3 pump flow. Operation with a slow CEA is permitted provided:

   a. The shutdown margin specified in 3.10.B.1 is satisfied]] without 1.5 times the reactivity associated with the slow CEA after 2.5 seconds of drop time.

C. 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 Type E only) and power measurement uncertainty (2%) shall not exceed:]]

   Fresh Fuel: 13.5 kW/ft for $\frac{X}{L}$ greater than 0.50 and CAB less than or equal to 792 MWD/MTU

   14.0 kW/ft for $\frac{X}{L}$ greater than 0.50 and CAB greater than 792 MWD/MTU

   16.0 kW/ft for $\frac{X}{L}$ less than or equal to 0.50

   Exposed Fuel: 14.0 kW/ft for $\frac{X}{L}$ greater than 0.50

   16.0 kW/ft for $\frac{X}{L}$ less than or equal to 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 specified above.

2. The total radial peaking factor, defined as

   $F_R^T = F_R^P (1 + T_q)$

   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:

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

   where Pi is the relative quandrant power determined from the incore system for quandrant i, when the incore system is operable and by Specification 3.10.C.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:

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1. 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 greater than or equal to:

   $[F_R^T \text{ measured}] \,/\, [F_R^T \text{ (Figure 3.10-4)}]$

   or

2. 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

3. Be in at least HOT SHUTDOWN.]]

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.C.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.

1. Initiate a power reduction to less than or equal to P at a rate of at least 1%/hour where P (% of rated Power) is given by:

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

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

2. Comply with the alarm band given in Figure 3.10-2 while maintaining the CEA's above the 100% power insertion limit. If a power reduction is required, reduce power at a rate of at least 1%/hour.]]

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

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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:

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

   Where Di is the signal from excore detector channel i. Tq shall not exceed 0.03.

   4.1 If the measured value of Tq is greater than 0.03 but less than or equal to 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.C.2 once per shift.

   4.2 If the measured value of Tq is greater than 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.C.2. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided:

   1. The thermal power level is restricted to less than or equal to 20% of the maximum allowable thermal power level for the existing Reactor Coolant Pump combination, and

   2. Reduce setpoints in accordance with Specification 3.10.C.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.

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D. Moderator Temperature Coefficient (MTC):

Except during low-power physics testing, the MTC shall be less positive than 0.5×10^{-4} delta rho/^{o}F.

E. Coolant Conditions

1. Except for low power physics testing, the reactor coolant pressure and the reactor coolant temperature at the inlet to the reactor vessel shall be maintained within the limits of Figure 3.10.6 during steady-state operation whenever the reactor is critical.

2. Except for low power physics testing, the reactor coolant flow rate shall be maintained at or more than a nominal value of 360,000 gpm during steady-state operation whenever the reactor is critical.

<u>Exception</u>

The requirements of 3.10.E.2 may be modified during initial testing to permit power levels not to exceed 10% of rated power with three loops operating on natural circulation.

<u>Basis</u>:

The CEA insertion limit shown in Figure 3.10-1 assures that the individual CEA worths used for the CEA ejection analyses are not exceeded. The CEA insertions used for the CEA withdrawal accident are also not exceeded by this insertion limit. In addition, the limit ensures that the reactor can be brought to a safe hot shutdown condition even with the highest worth CEA not inserted. This restriction provides more shutdown margin than is required at BOL, since the moderator temperature coefficient is more negative at EOL. For this regulating group insertion limit, the peak linear heat rate will be well within the design values.

The limit applies also to two loop operation, in which case the power coordinate is rescaled to 100% of the rated two loop power. This ensures that the CEA induced peaking will not lead to worse thermal conditions than for 3 loop operation since the flow to power ratio is greater for two loop operation. This CEA insertion limit may be revised on the basis of physics calculations and physics data obtained during plant startup and subsequent operation.

For a full length CEA, with misalignment up to 10 steps from the remainder of the group, the peaking factors will be well within design limits. The power level and CEA restrictions imposed for operation with a misaligned CEA assure that the assumptions used in the generation of the RPS setpoints are not violated. The 4 hour time limit restriction is short with respect to the probability of an independent incident occurring. The requirement that no more than one inoperable CEA is allowed and that the shutdown margin is maintained ensures that the reactor can be brought to a safe shutdown condition at any time.

Shutdown margin is assured within the required CEA drop time by operating in accordance with 3.10.A-1 and measuring CEA core height vs. time and CEA worths after intial loading and each refueling. The maximum CEA drop time specified is consistent with values used in the safety analysis. Should a CEA drop time be in excess of 3.10.B.3, then the core height on that CEA at 2.5 seconds would be conservatively determined. Reactivity worth of the CEA from the above core height to the bottom of the core would then be determined. Appropriate action would be taken, if necessary, during power operation to compensate for 1.5 times the above measured reactivity in order to maintain adequate shutdown margin.

Incore detector alarms are set based on the latest power distributions obtained from incore detector analyses. These techniques reflect actual radial and axial power distribution which exist in the core. Should the system become unavailable, continued operation is permitted under either the more conservative excore symmetric offset pretrip (alarm) envelope or at a power level consistent with maintaining an appropriate margin to the peak linear heat rate assumed in the LOCA. Both these functions ensure that operation is within the limiting peak linear heat rates assumed as initial conditions for the Loss of Coolant Accident (LOCA). Further, since rod position information is not available to this excore system, this function assumes the most limiting radial power distributions permitted at each power level.

The split excore detectors monitor the axial component of the power distribution. The signal generated from the excore detectors is provided as input to both the Symmetric Offset and Thermal Margin/Low Pressure Trip Systems. Limiting Safety System Settings (LSSS) are, therefore, generated as a function of the excore detector response.

The radial component of the power distribution is monitored as a Limiting Condition of Operation (LCO) by Technical Specification 3.10.C.3. The intent of the specification is to monitor the radial component of the power distribution and to ensure that assumptions made in the generation of Reactor Protective System (RPS) LSSS remain valid. The LCO on the radial power distribution is specified in Figure 3.10-4 in the form of a steady-state unrodded total radial peak (F_R^T) and provides indication that the core power distribution is behaving as predicted. Figure 3.10-4 includes 10% for calculational uncertainties. The measured steady-state value of F_R^T, augmented by 8% for measurement uncertainty, is compared to this limit on a monthly basis. Should the measured steady-state unrodded total radial peak including uncertainties exceed the limit of Figure 3.10-4 at any time in the cycle, specific action is to be taken to assure that the LSSS remain valid. The specific action includes a) the reduction of RPS LSSS and LCO by the ratio of [F_R^T (measured)/F_R^T (Figure 3.10-4)] to directly compensate for the higher radial peaks, or b) the imposition of additional restrictions on power and CEA position (Figure 3.10-5) to assure that the assumptions made in establishing the RPS LSSS and LCO remain valid. Figure 3.10-5 in

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conjunction with restricted CEA insertion allows for an increase in the steady-state unrodded total radial peak above the limits of Figure 3.10-4 without a modification of the RPS LSSS. The allowed increase in radial peak is derived from the difference between the radial peaks assumed in the RPS setpoints for rodded conditions at reduced power and the radial peaks reflected in the CEA insertion limit at 100% power. This assures that the radial peaking factors vs. power assumed in the RPS LSSS remain valid.

The power distribution in the core can be determined in two ways. The normal method is through analysis of the fixed and movable neutron detector signals with the on-line computer. The alternative is to determine the radial and axial peaking factors by hand. The radial peaking factor can be determined from the core exit thermocouples, the fixed incore detectors or the movable incore detector traces. The axial peaking factor can be determined from the fixed incore detectors, the movable incore detector traces or the excore detectors. The requirement that the core power distribution be shown to be within the design limits after every refueling not only ensures that the reactor can be operated safely but will also provide reasonable verification that the core was properly loaded. The requirement for operability of incore instrumentation in the instance of an excore detector channel being out of service ensures that an unobserved quandrant power tilt will not occur.

The moderator temperature coefficient, coolant pressure, flow rate, and temperature specified are consistent with the value assumed in the safety analysis. The safety analysis assumes a maximum reactor inlet temperature of 554^{o}F. The specified value includes 4^{o}F for temperature measurement uncertainties. The exception permits testing]]
to determine decay heat removal capabilities of the Primary System]]
while on natural circulation, prior to operation at higher power.]]

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MAINE YANKEE
Technical
Specification

Power-Dependent Insertion Limit
(P D I L)
for CEA's

Figure
3.10-1

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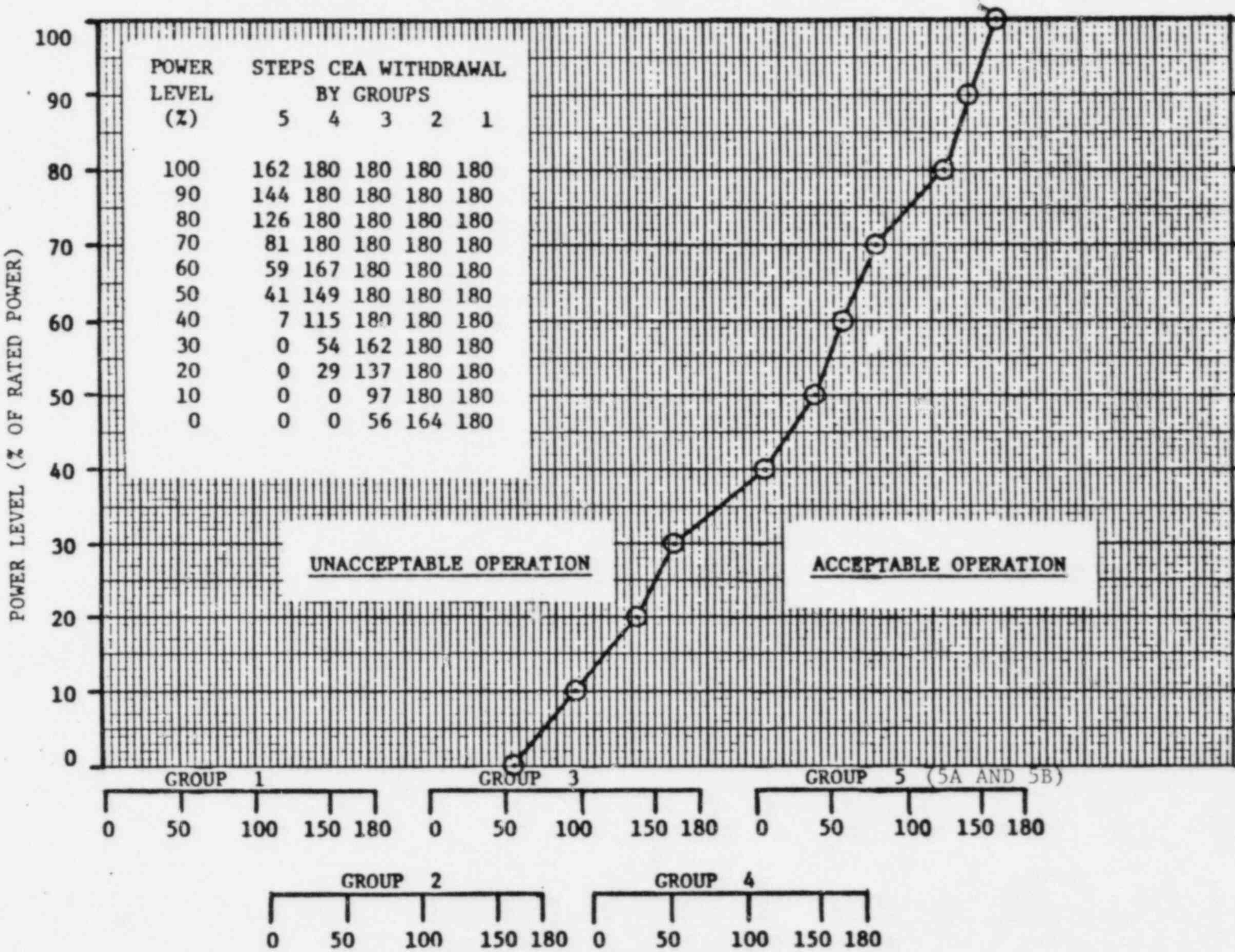
POWER LEVEL (% OF RATED POWER) VS. CEA WITHDRAWAL (STEPS) FOR THREE LOOP OPERATION
OR (% OF 0.63 X RATED POWER) FOR TWO LOOP OPERATION

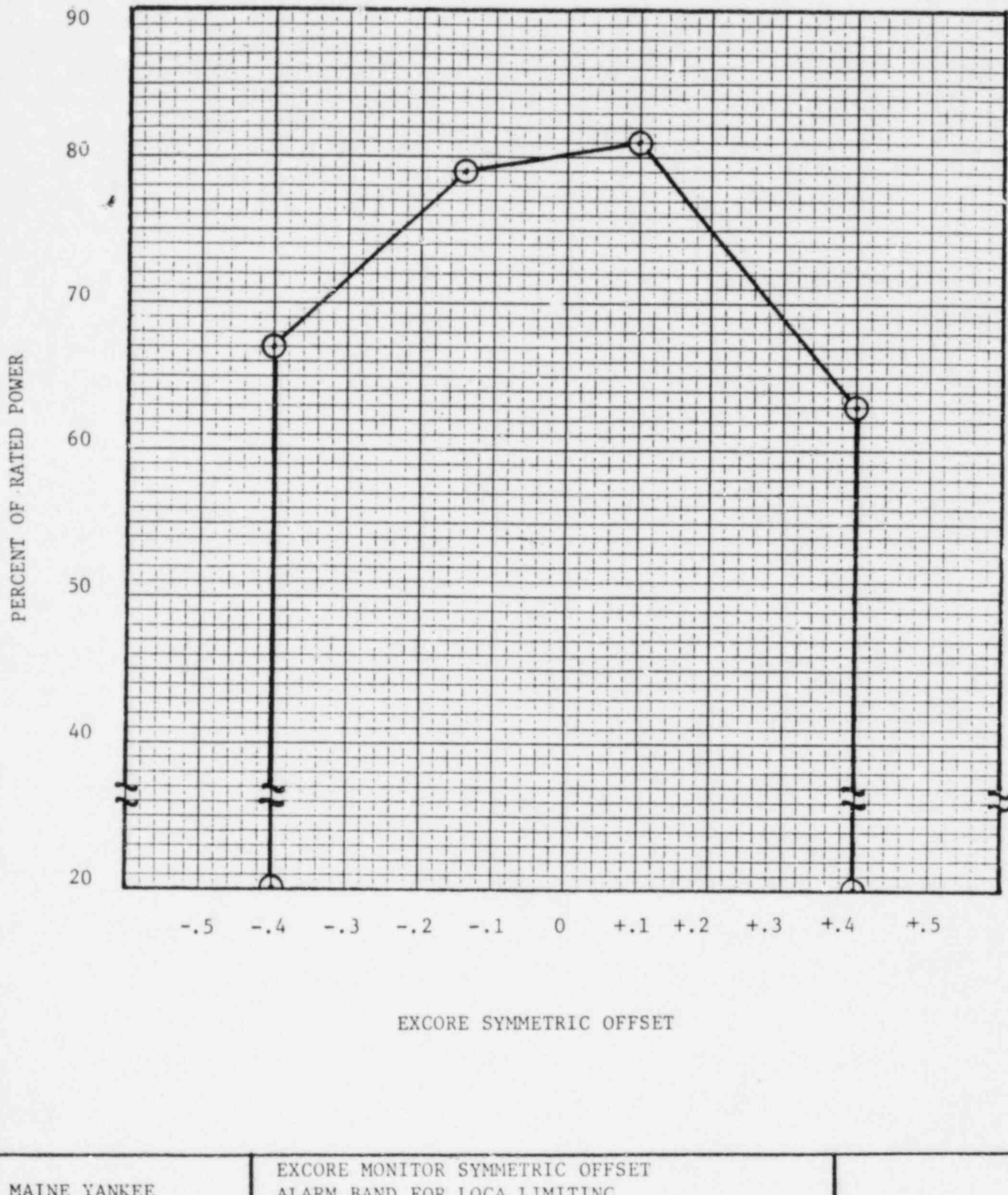


POWER LEVEL (%)	STEPS CEA WITHDRAWAL BY GROUPS 5	4	3	2	1
100	162	180	180	180	180
90	144	180	180	180	180
80	126	180	180	180	180
70	81	180	180	180	180
60	59	167	180	180	180
50	41	149	180	180	180
40	7	115	180	180	180
30	0	54	162	180	180
20	0	29	137	180	180
10	0	0	97	180	180
0	0	0	56	164	180

NOTE: IN APPLYING THIS RESTRICTION TO TWO LOOP OPERATION, THE RATED POWER FOR TWO LOOP OPERATION MUST BE USED WITHIN FOUR (4) HOURS OF ATTAINING TWO LOOP CONDITIONS

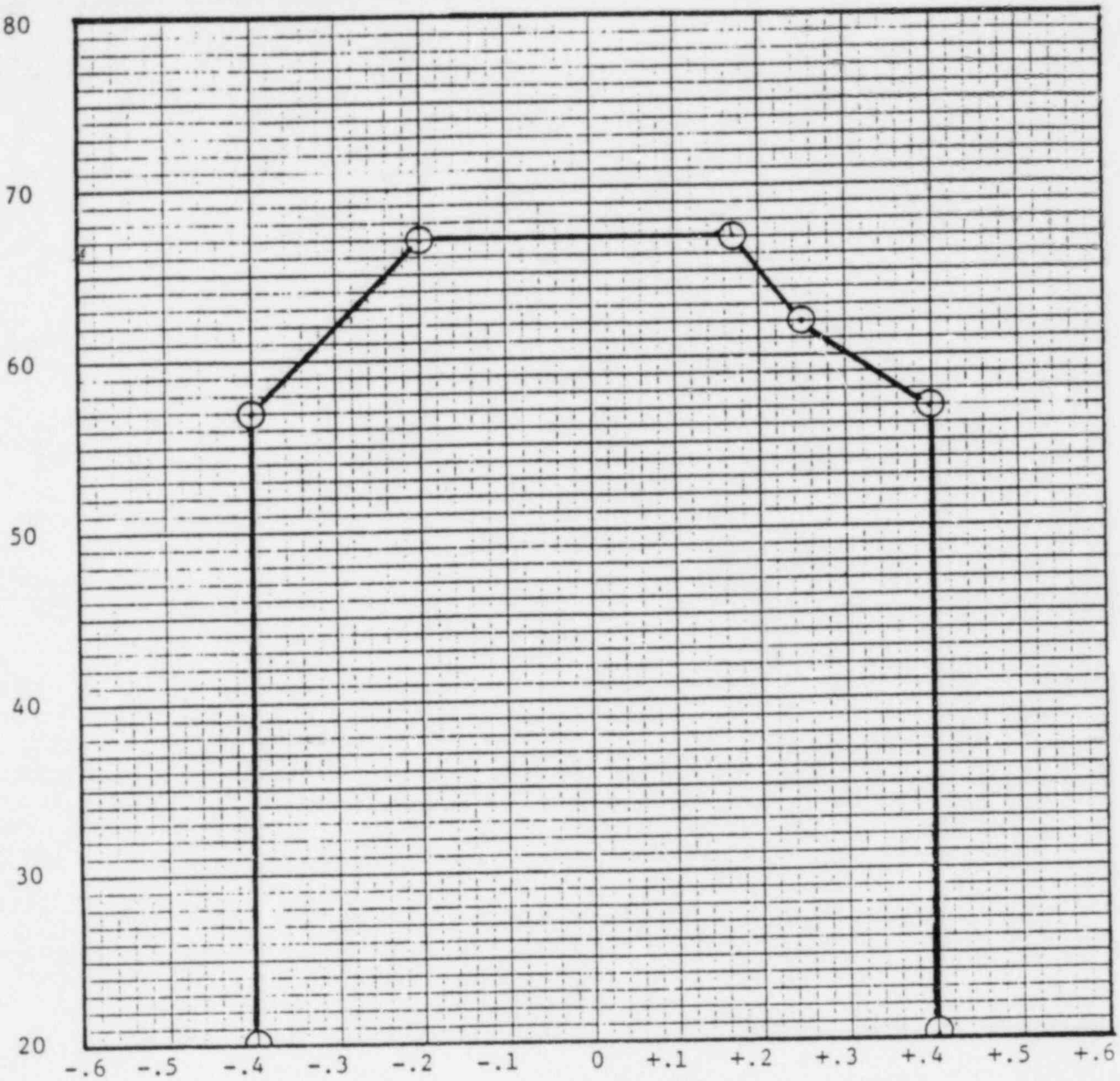
COORDINATES (S/O,% RATED POWER)

(-.40, .20) (-.40, .67) (-.14, 79)(+.10 .81) (+.40, .63) (+.40, .20)



MAINE YANKEE TECHNICAL SPECIFICATIONS	EXCORE MONITOR SYMMETRIC OFFSET ALARM BAND FOR LOCA LIMITINC CONDITIONS OF OPERATION WHEN INCORE MONITORS ARE INOPERABLE AND CEAs RESTRICTED ABOVE THE 100% RATED POWER INSERTION LIMIT. 3 .10 - 10	FIGURE 3.10-2 SEP 2 9 1982

COORDINATES (S/O, % RATED POWER)
(-.40, 20) (-.40, 57) (-.20, 67) (+.17, 67) (+.25, 62) (+.40, 20)



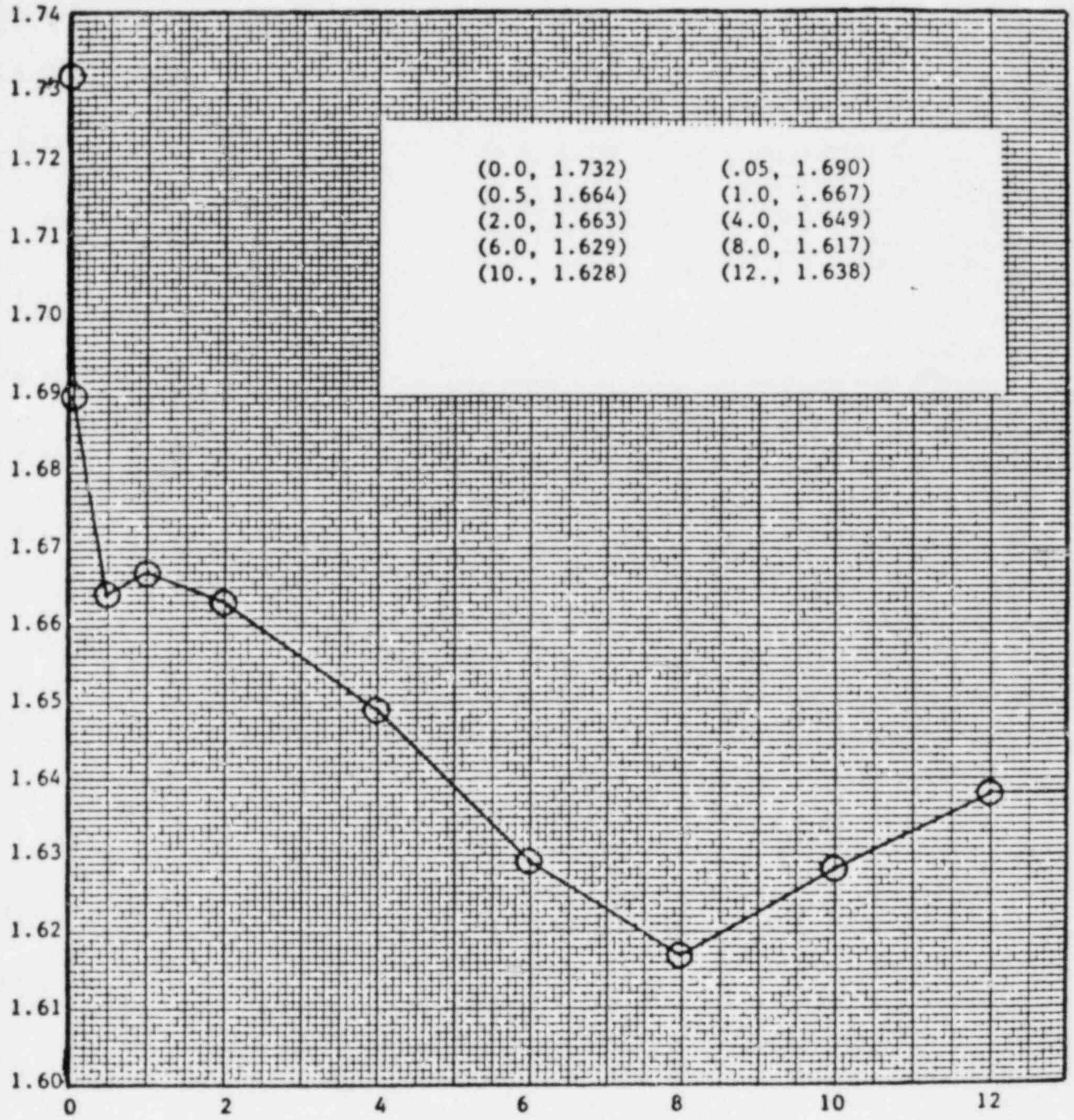
EXCORE SYMMETRIC OFFSET

MAIN YANKEE TECHNICAL SPECIFICATIONS	EXCORE MONITOR SYMMETRIC OFFSET ALARM BAND FOR LOCA LIMITING CONDITIONS OF OPERATION WHEN INCORE MONITORS ARE INOPERABLE 3.10-11	FIGURE 3.10-3 SEP 29 1982

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 Specification	Allowable Unrodded Radial Peak Versus Cycle Average Burnup	Figure 3.10-4

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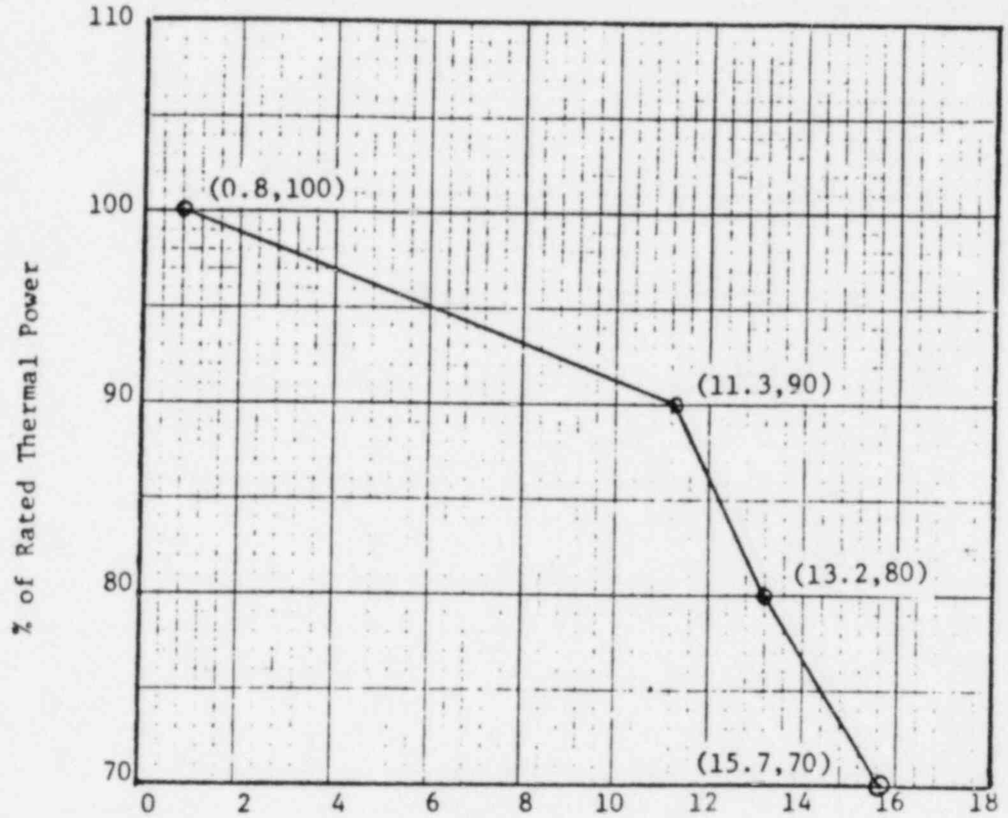
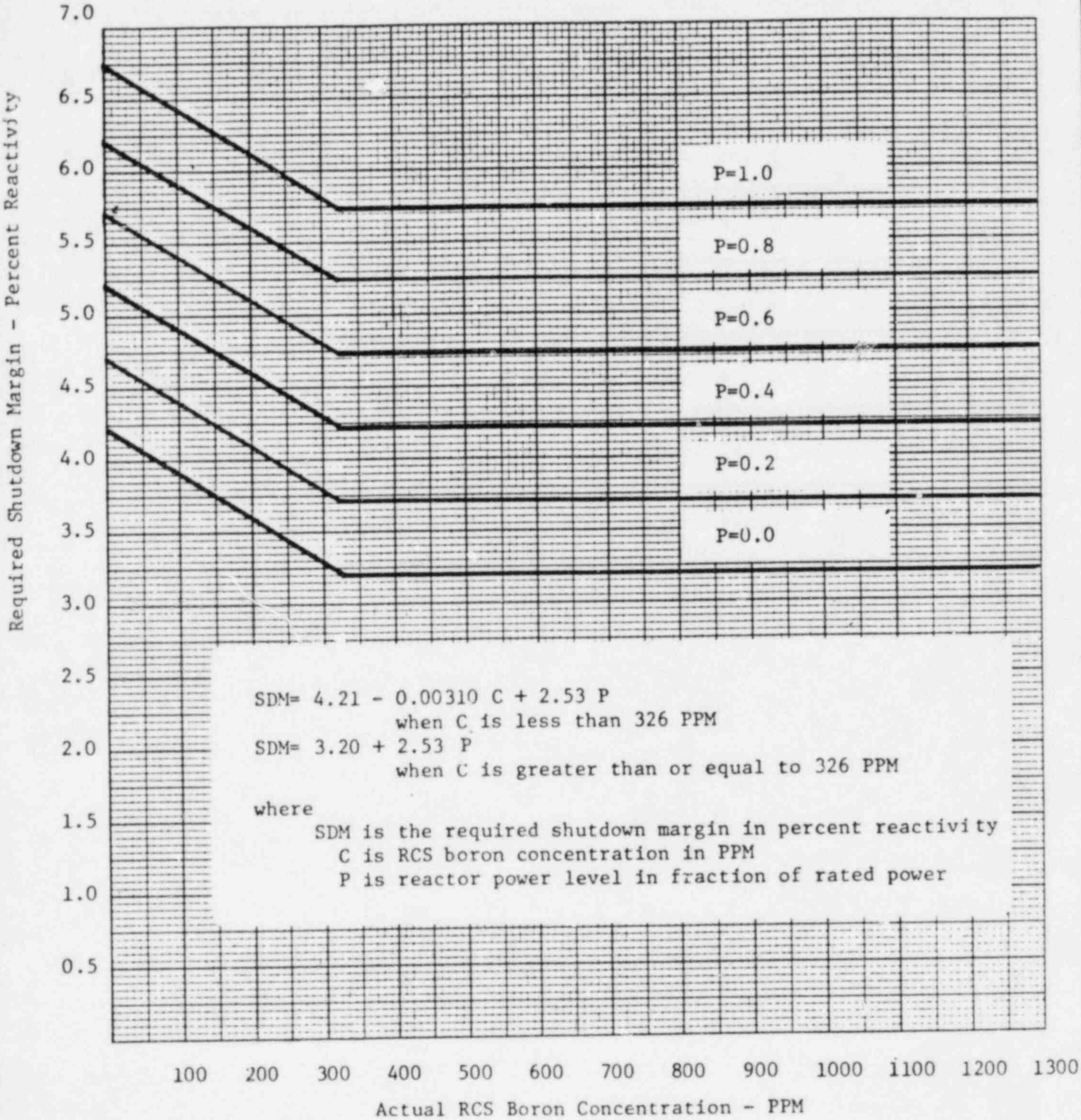
Note: CEA's are maintained at or above the 100% power insertion limit when applying 3.10.C.2.2.2



Allowable % Increase in F_R^T (above Figure 3.10-4)

MAINE YANKEE Technical Specification	Allowable Power Level vs. Increase in Total Radial Peak	Figure 3.10-5
	3.10-13	SEP 2 9 1982



Maine Yankee Technical Specification	Required Shutdown Margin Versus RCS Boron Concentration 3.10-15	Figure 3.10-7 SEP 29 1982