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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. NPF-10

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 2 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated February 29, April 2, July 2, August 7, October 1 and 3, 1984, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

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

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.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 32, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Those changes requested as Proposed Change Nos. 52, 85, 168 and 169 are effective as of the date of issuance and shall be fully implemented within 30 days of issuance of the amendment. Those changes requested as Proposed Change Nos. 148, 150, 151, 152, 153, 160 and 162 are effective on initial entry into the applicable operational mode following first refueling.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: MAR 1 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 32

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
VIII	VII
XIV	XIII
2-1	2-2
2-3	-
2-4	-
2-5	2-6
2-7	-
B 2-1	-
B 2-2	-
B 2-5	-
B 2-6	-
3/4 1-4	3/4 1-3
3/4 1-24	3/4 1-23
3/4 2-5	3/4 2-6
3/4 2-7	-
3/4 2-7a	-
3/4 2-8	-
3/4 2-8a	-
3/4 2-8b	-
3/4 2-8c	-
3/4 2-11	3/4 2-12
3/4 3-6	3/4 3-5
3/4 3-7	-
3/4 3-7a	-
3/4 3-9	3/4 3-8
3/4 3-9a	-
3/4 3-9b	-
-	*3/4 3-10
3/4 10-4	3/4 10-3
B 3/4 1-4	B 3/4 1-3
B 3/4 2-3	B 3/4 2-4
B 3/4 4-1	B 3/4 4-2
B 3/4 10-1	B 3/4 10-2
5-6	5-5

*Reissued without change.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.31.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.31, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 19.95 kw/ft	≤ 19.95 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grids is 1.31.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.02
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted
98	TCREF	Reference Cold Leg Temperature	$520°F \leq$ TCREF $\leq 580°$
104	PCALIB	Calorimetric Power	≤ 102.0

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERRO	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient
103	RPCLIM	Reactor Power Cutback Time Limit

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which 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 temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB 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 normal operational occurrences is limited to 1.31 for the CE-1 correlation and is established as a Safety Limit.

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 steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1971 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.31 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.31 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	< +0.5
d.	Axial Shape Index-Negative	$\geq$ -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1825 psia
f.	Pressurizer Pressure-High	$\leq$ 2375 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	< 0

The DNBR Trip setpoint in CPC and COLSS is 1.31. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old}\ [1 + \Delta DNBR(\%)^{*} \frac{d(\%\ POL)}{d\ (\%\ DNBR)} {}^{*}0.01]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$d(\%\ POL)/d(\%\ DNBR)$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = (1 + \Delta DNBR(\%)^{*} \frac{d(\%\ POL)}{d\ (\%\ DNBR)} {}^{*}0.01)^{*}(1 + EPOL2_{old})-1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,and
6. Samarium concentration.
7. Whenever the reactor coolant level is below the hot leg centerline, one and only one charging pump shall be operable; by verifying that power is removed from the remaining charging pumps.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or

Less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

ACTION: (Continued)

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

   2. Be in at least HOT STANDBY within 6 hours.

When COLSS is out of service and the regulating CEA groups are inserted beyond the Short Term Steady State Insertion Limit except for Surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

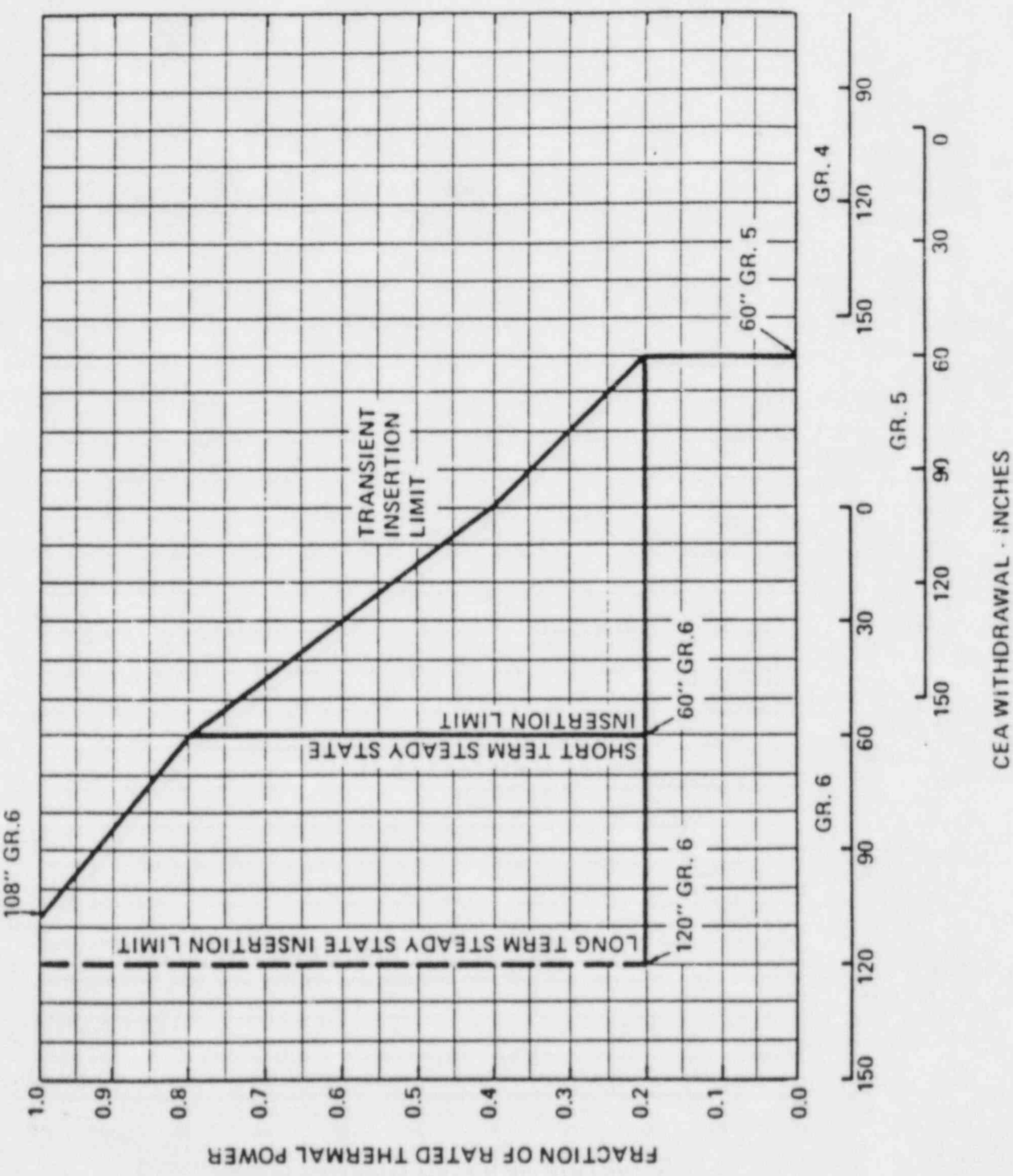
a. Restore the regulating CEA group to within the limit, or

b. Reduce thermal power to less than or equal to the fraction of Rated Thermal Power which is allowed by the CEA group position and the Short Term Steady State Insertion Limit.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

FIGURE 3.1-2


TRANSIENT INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
LONG TERM STEADY STATE INSERTION LIMIT
108" GR.6
60" GR.6
120" GR. 6
60" GR. 5
FRACTION OF RATED THERMAL POWER
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
CEA WITHDRAWAL - INCHES
GR. 6
150
120
90
60
30
0
GR. 5
150
120
90
60
30
0
GR. 4
150
120
90

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable and the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

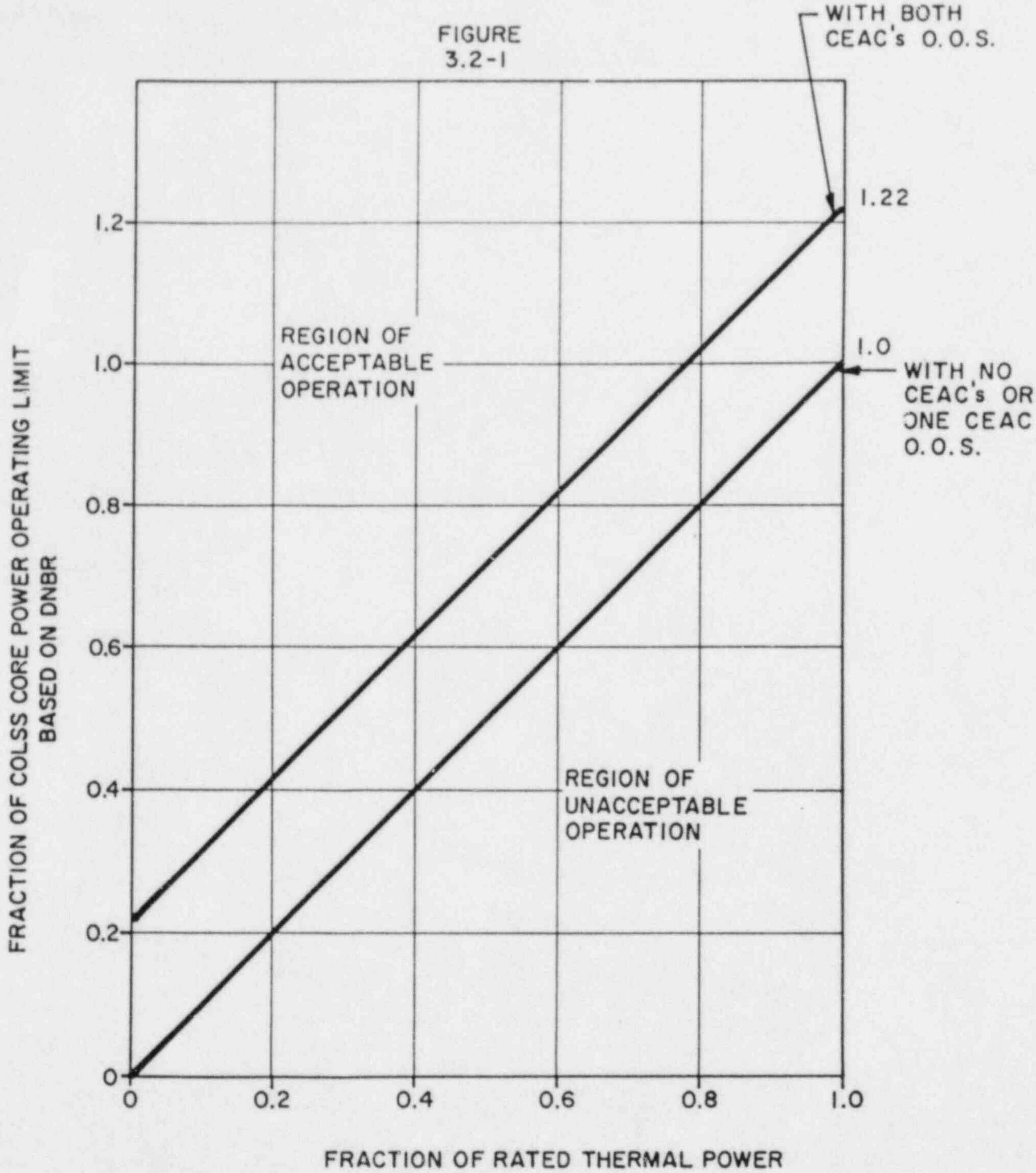


FIGURE 3.2-1 CYCLE 1 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

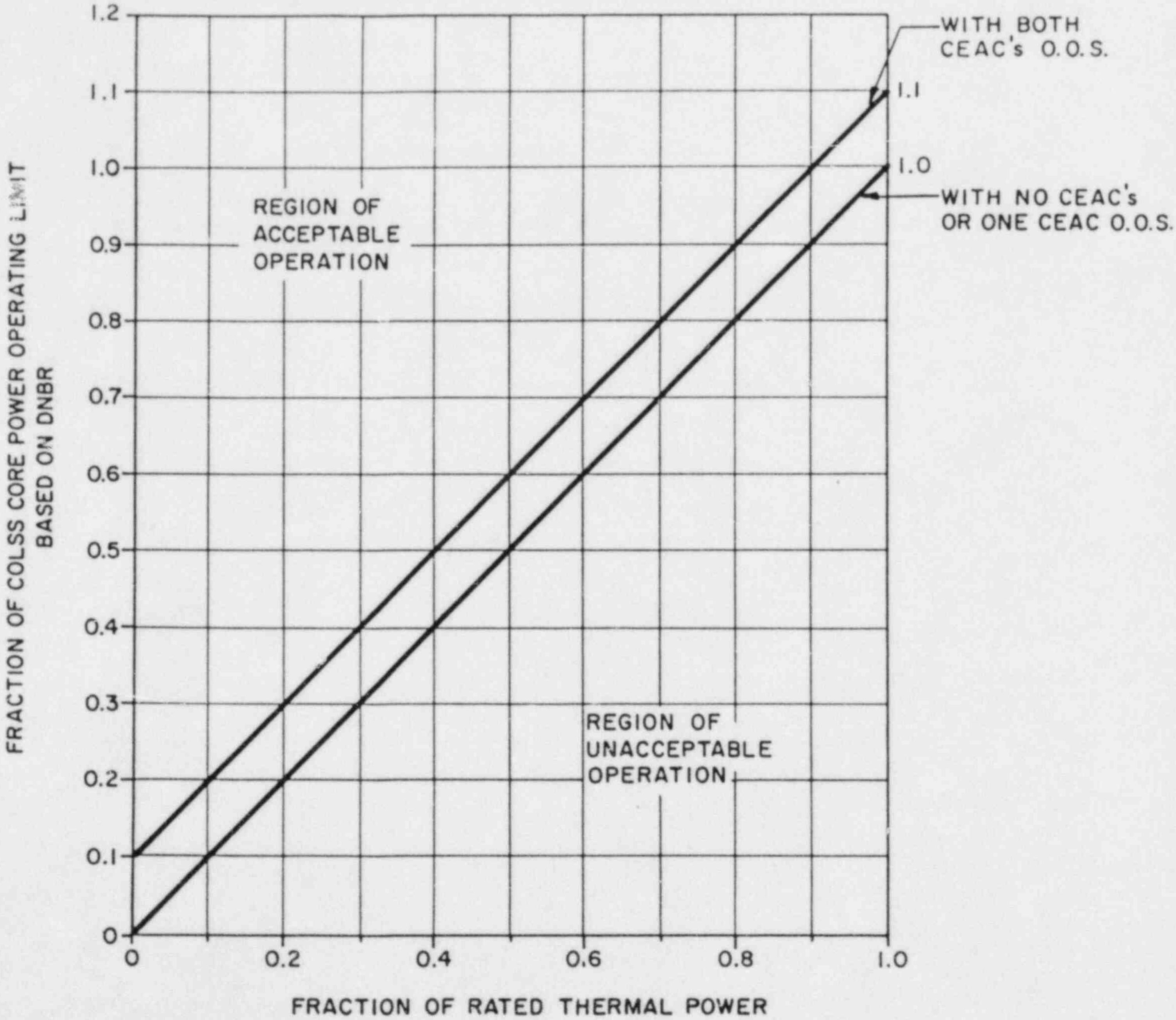


FIGURE 3.2-1 CYCLE 2 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

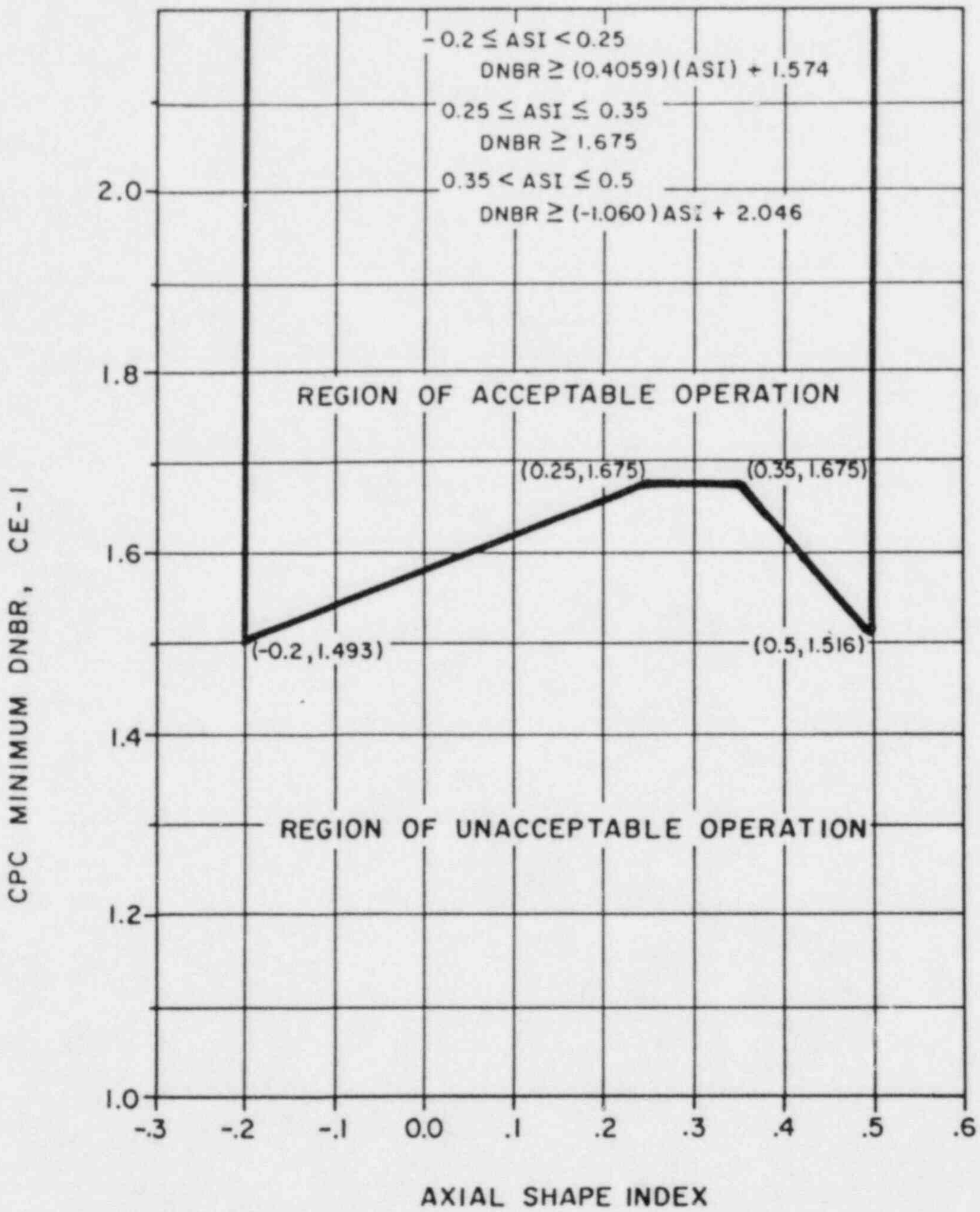


FIGURE 3.2-2 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER ≥ 80%)

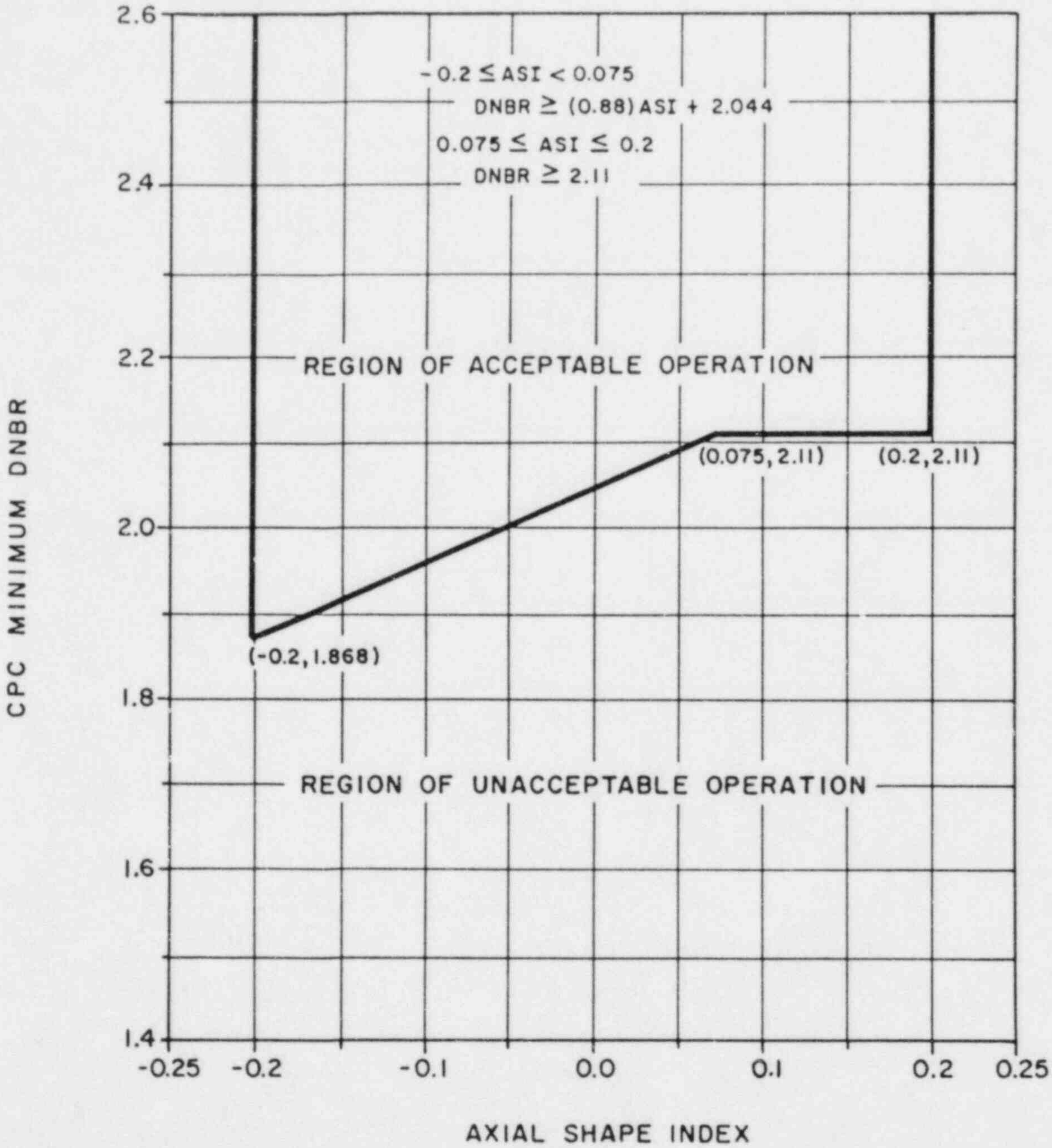


FIGURE 3.2-2 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER ≥ 70%)

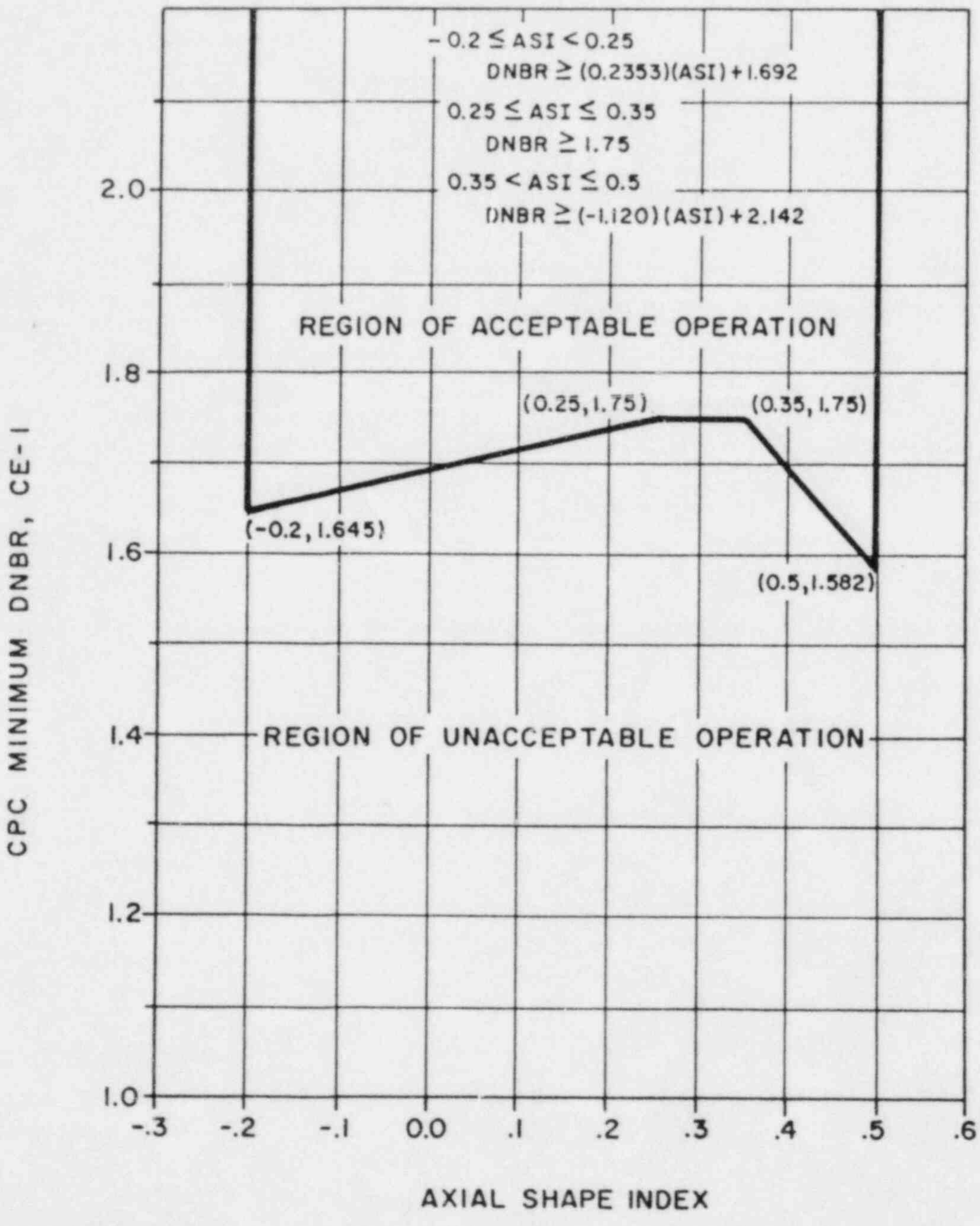


FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 80%)

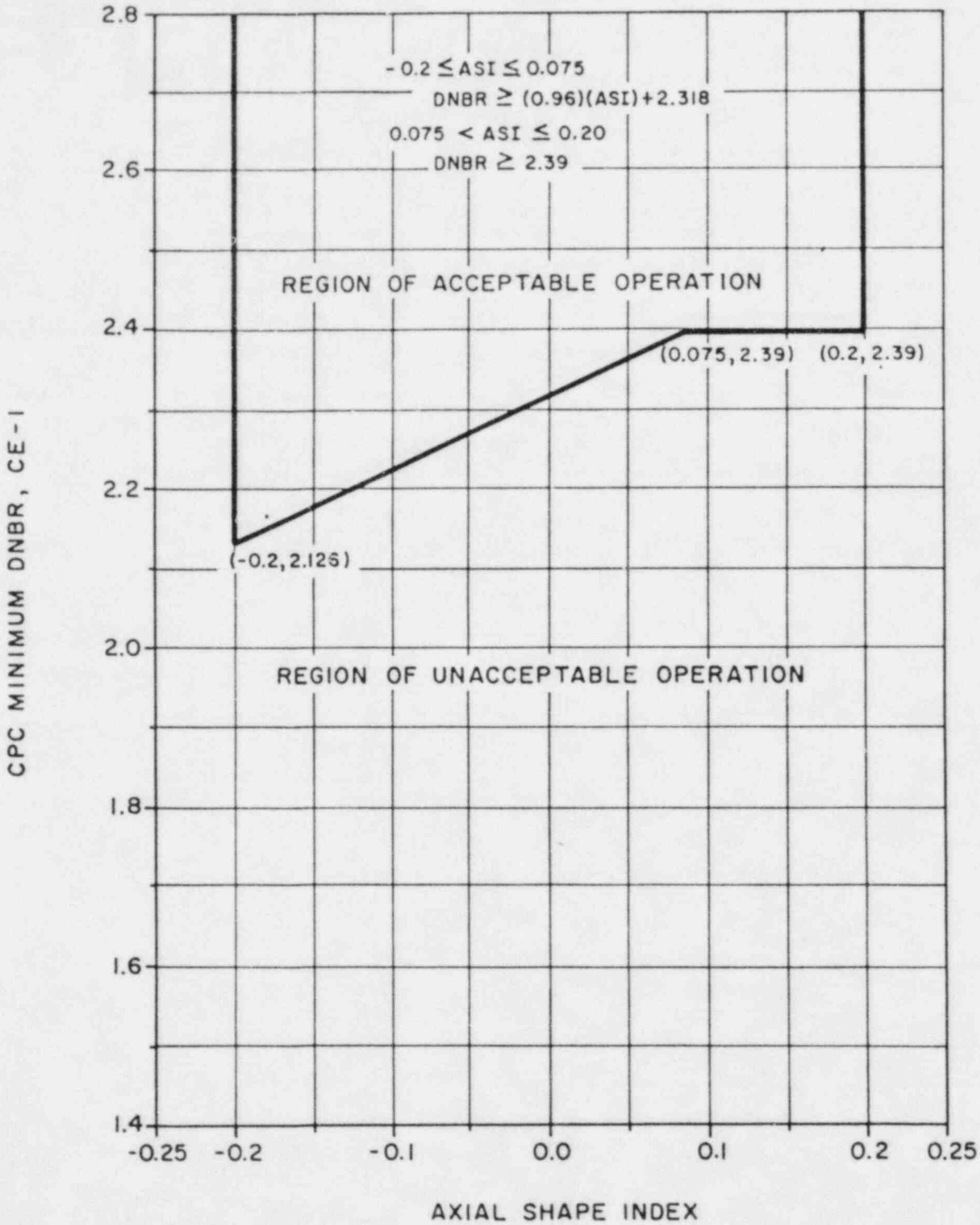


FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 70%)

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq + 0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq + 0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the average pressurizer pressure exceeding its limit, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5. Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6. Core Protection Calculator	Local Power Density - High DNBR - Low

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that Action Items 6.b.1, .2 and .3 are met with COLSS in-service, or Action Items 6.c.1, .2 and .3 are met with COLSS out-of-service*.

b. With both CEAC's inoperable and COLSS in-service, operation may continue provided that:*

1. Within 1 hour the DNBR margin operating limit required by Specification 3.2.4 (Figure 3.2-1) is satisfied for both CEAC's out-of-service

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE 3.3-1 (Continued)

TABLE NOTATION

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

c. With both CEAC's inoperable and COLSS out-of-service, operation may continue provided that:*

   1. Within 1 hour multiply the CPC value of BERR1 corresponding to COLSS in-service by 1.13 (CYCLE 1) or 1.05 (CYCLE 2) and re-enter into the CPC's.

   2. Within 4 hours:

      a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE 3.3-1 (Continued)

TABLE NOTATION

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 seconds*
3. Logarithmic Power Level - High	≤ 0.45 seconds*
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Pressurizer Pressure - Low	≤ 0.90 seconds
6. Containment Pressure - High	≤ 0.90 seconds
7. Steam Generator Pressure - Low	≤ 0.90 seconds
8. Steam Generator Level - Low	≤ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. CEA Positions: CEAC Penalty Factor	≤ 0.53 seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. Cold Leg Temperature	≤ 0.68 seconds##
d. Hot Leg Temperature	≤ 0.68 seconds##
e. Primary Coolant Pump Shaft Speed	≤ 0.68 seconds#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.68 seconds
g. CEA positions: CEAC Penalty Factor	≤ 0.53 seconds

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 13.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. Adjustments to the CPC addressable constants in Table 3.3-2a and reductions in the DNBR Power Operating Limit in Table 3.3-2b shall be made to accommodate measured values of RTD time constants.

TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD Delay Time τ	BERR0 Increase %		BERR2 Increase %		BERR4 Increase %	
	Cycle 1	Cycle 2	Cycle 1	Cycle 2	Cycle 1	Cycle 2
$\tau \leq$ 6.0 sec.	0.0	0.0	0.0	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	0.0	2.0	3.5	5.0	3.0	0.0
8.0 sec. $< \tau \leq$ 10.0 sec.	3.5	5.0	4.0	8.5	9.0	3.0
10.0 sec. $< \tau \leq$ 13.0 sec.	10.5	9.0	5.5	12.0	17.0	6.0

NOTE: BERR term increases are not cumulative, i.e., if the values of the BERR terms are currently 10.0, then for an RTD delay time of > 6.0 to $\leq$ 8.0 sec., in Cycle 1: BERR0 = 10.0 + 0.0 = 10.0; BERR2 = 10.0 + 3.5 = 13.5; and, BERR4 = 10.0 + 3.0 = 13.0. For RTD delay times of > 8.0 to $\leq$ 10.0 sec., in Cycle 1: BERR0 = 10.0 + 3.5 = 13.5; BERR2 = 10.0 + 4.0 = 14.0; and BERR4 = 10.0 + 9.0 = 19.0. Computed values in this paragraph are examples only.

NOTE: In Cycle 1 only, when any of the above increases are applied to the BERR terms for any CPC channel, the COLSS constant EPOL2 is reduced by 0.04. This applies for Cycle 1 only.

TABLE 3.3-2b

DNBR LCO POWER OPERATING LIMIT ADJUSTMENTS

RTD Delay Time (sec)	Adjustment to EPOL1,[1] COLSS In Service (% power)	Adjustment to BERR2,[1,2] COLSS Out of Service (% power)	
		Cycle 1	Cycle 2
$\tau \leq 6.0$ sec.	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	-4.0	+4.0	+5.0
8.0 sec. $< \tau \leq$ 10.0 sec.	-5.0	+5.0	+8.5
10.0 sec $< \tau \leq$ 13.0 sec.	-7.0	+7.0	+12.0

NOTES:

1. Adjustments are not cumulative: i.e., if τ increases from 7.0 seconds to 9.0 seconds, EPOL1 is reduced by 5.0 from its original value, not 4.0 + 5.0 = 9.0 from its original value.
2. If COLSS is out-of-service, these adjustments are to be used in place of, not in addition to, the increases required by Table 3.3-2a and the limit in Figure 3.2-2 or 3.2-3, as applicable, must be maintained for all operable CPC channels.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4), M(3,4), Q(4), R(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and the noted requirements of Table 2.2-1 and Table 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each logarithmic and linear power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, or only Regulating CEA Group 6 is inserted beyond the Transient Insertion Limit of Specification 3.1.3.6, and

b. The limits of Specifications 3.2.1 and 3.2.4 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specifications 3.2.1 or 3.2.4 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, and 3.2.4 or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate and DNBR Margin shall be determined to be within the limits of Specifications 3.2.1 and 3.2.4, respectively, by monitoring them continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable, CEA to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

MOVABLE CONTROL ASSEMBLIES (Continued)

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits. Setting the "RSPT/CEAC Inoperable" addressable constant in the CPC's to indicate to the CPC's that one or both of the CEAC's is inoperable does not necessarily constitute the inoperability of the RSPT rod indications from the respective CEAC. Operability of the CEAC rod indications is determined from the normal surveillance.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 or 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Uncertainty terms already taken into account in the CPC's safety monitoring are removed from Figures 3.2-2 and 3.2-3 since the curves are intended to monitor only the LCO during steady state operation.

DNBR Margin (Continued)

The DNBR penalty factors listed in section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 235°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 235°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

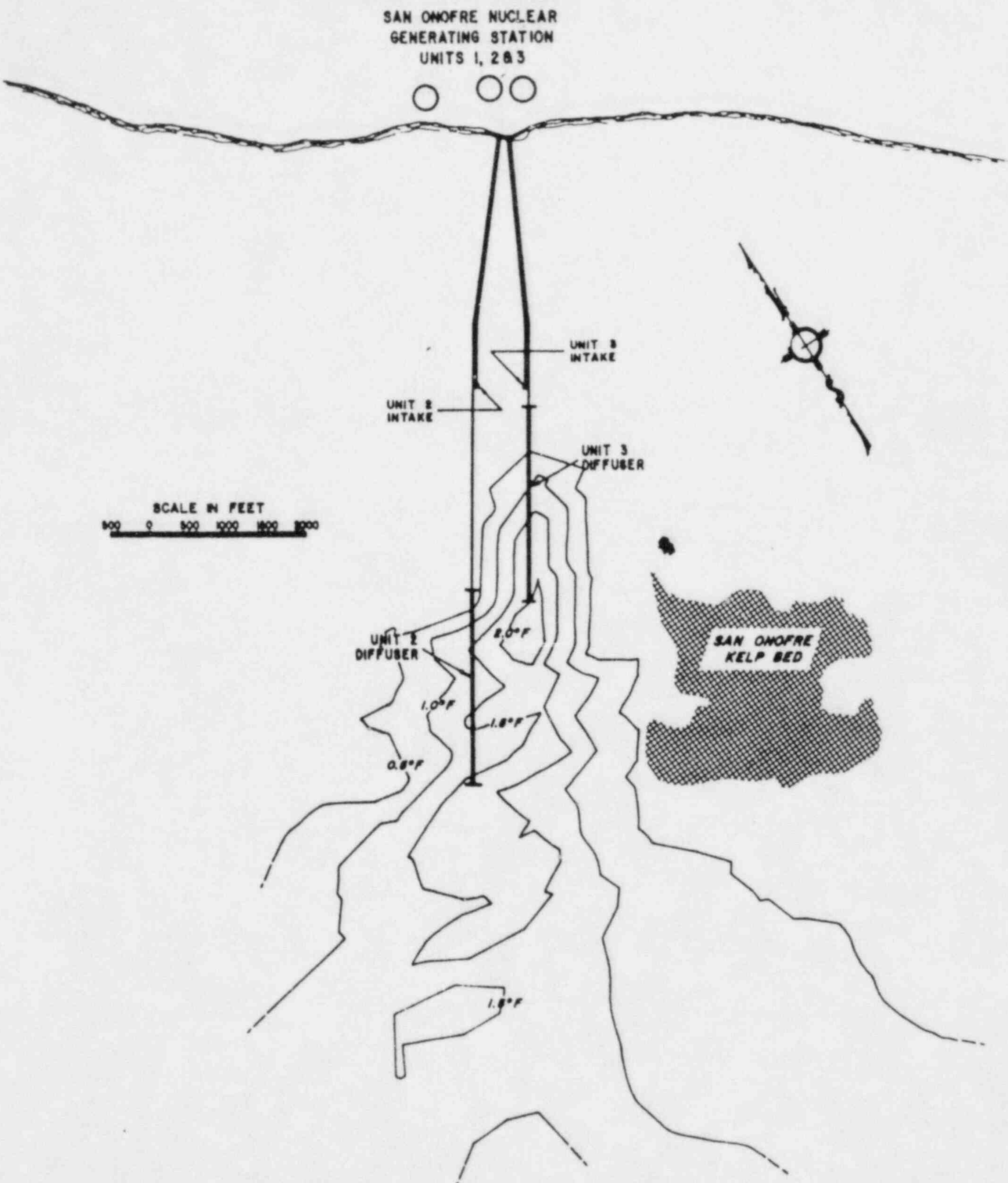
This special test exception permits the center CEA to be misaligned or Regulating Group 6 inserted beyond the Transient Insertion Limit during PHYSICS TESTS required to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient.

3/4.10.5 RADIATION MONITORING/SAMPLING

This special test exception permits fuel loading and reactor operation with radiation monitoring/sampling instrumentation calibration and quality assurance conforming to either FSAR procedures or Regulatory Guide 4.15 Rev 1, February 1979. This test exception is required to allow for a phased implementation of Regulatory Guide 4.15 Rev. 1, February 1979. Equivalent instrumentation, quality assurance and/or calibration is provided until full implementation of Regulatory Guide 4.15 Rev. 1, February 1979.

BASES

The containment airborne monitors and associated sampling media test exception is required to allow for operation prior to and during installation of upgraded monitors/media. Adequate monitoring is provided until and subsequent to the completion of the upgraded installation. Extensive containment air mixing during high volume purge (MODES 5 and 6) occurs as a result of containment HVAC and fans resulting in representative air monitoring via either 2RT-7804-1 or 2RT-7807-2. During low volume purge operations (MODES 1, 2, 3 and 4) 2RT-7804-1 provides representative indication of purged air due to its location in the immediate vicinity of the low volume purge exhaust.



SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. NPF-15

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 3 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) February 29, April 2, July 2, August 7, October 1 and 3, 1984, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

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

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.C(2) of Facility Operating License No. NPF-15 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 21, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Those changes requested as Proposed Change Nos. 52, 85, 168 and 169 are effective as of the date of issuance and shall be fully implemented within 30 days of issuance of the amendment. Those changes requested as Proposed Change Nos. 148, 150, 151, 153, 160 and 162 are effective on initial entry into the applicable operational mode following first refueling.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: MAR 1 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
VIII	VII
XII	XI
2-1	2-2
2-3	-
2-4	-
2-5	2-6
2-7	-
B 2-1	-
B 2-2	-
B 2-5	-
B 2-6	-
3/4 1-4	3/4 1-3
3/4 1-24	3/4 1-23
3/4 2-5	3/4 2-6
3/4 2-7	-
3/4 2-7a	-
3/4 2-8	-
3/4 2-8a	-
3/4 2-8b	-
3/4 2-8c	-
3/4 2-11	3/4 2-12
3/4 3-6	3/4 3-5
3/4 3-7	-
3/4 3-7a	-
3/4 3-9	3/4 3-8
3/4 3-9a	-
3/4 3-9b	*3/4 3-10
-	-
3/4 10-4	3/4 10-3
B 3/4 1-4	B 3/4 1-3
B 3/4 2-3	B 3/4 2-4
B 3/4 4-1	B 3/4 4-2
B 3/4 10-1	-
5-6	5-5

*Reissued without change.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

BASES

SECTION | PAGE

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.31.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.31, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 19.95 kw/ft	$\leq$ 19.95 kw/ft
10. DNBR - Low	$\geq$ 1.31 (5)	$\geq$ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.22 psid/sec (6)(8)	$\leq$ 0.231 psid/sec (6)(8)
b) Floor	$\geq$ 13.2 psid (6)(8)	$\geq$ 12.1 psid (6)(8)
c) Step	$\leq$ 6.82 psid (6)(8)	$\leq$ 7.231 psid (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grid is 1.31.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥1.02
64	TPC	Thermal power calibration constant	≥0.90
65	KCAL	Neutron flux power calibration constant	≥0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted
98	TCREF	Reference Cold Leg Temperature	520°F ≤TCREF ≤580°F
104	PCALIB	Calorimetric Power	≤102.0

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERR0	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient
103	RPCLIM	Reactor Power Cutback Time Limit

BASES

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which 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 temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB 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 normal operational occurrences is limited to 1.31 for the CE-1 correlation and is established as a Safety Limit.

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 steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1971 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.31 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.31 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	≥ 495°F
b.	RCS Cold Leg Temperature-High	≤ 580°F
c.	Axial Shape Index-Positive	< +0.5
d.	Axial Shape Index-Negative	≥ -0.5
e.	Pressurizer Pressure-Low	≥ 1825 psia
f.	Pressurizer Pressure-High	≤ 2375 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 4.28
i.	Quality Margin-Low	< 0

The DNBR Trip setpoint in CPC and COLSS is 1.31. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old}\,[1 + \Delta DNBR(\%) * \frac{d(\%\ POL)}{d\ (\%\ DNBR)} * 0.01]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$d(\%\ POL)/d(\%\ DNBR)$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = (1 + \Delta DNBR(\%) * \frac{d(\%\ POL)}{d\ (\%\ DNBR)} * 0.01) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

    1. Reactor coolant system boron concentration,
    2. CEA position,
    3. Reactor coolant system average temperature,
    4. Fuel burnup based on gross thermal energy generation,
    5. Xenon concentration,and
    6. Samarium concentration.
    7. Whenever the reactor coolant level is below the hot leg centerline, one and only one charging pump shall be operable; by verifying that power is removed from the remaining charging pumps.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

ACTION: (Continued)

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

   2. Be in at least HOT STANDBY within 6 hours.

When COLSS is out of service and the regulatiing CEA groups are inserted beyond the Short Term Steady State Insertion Limit except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

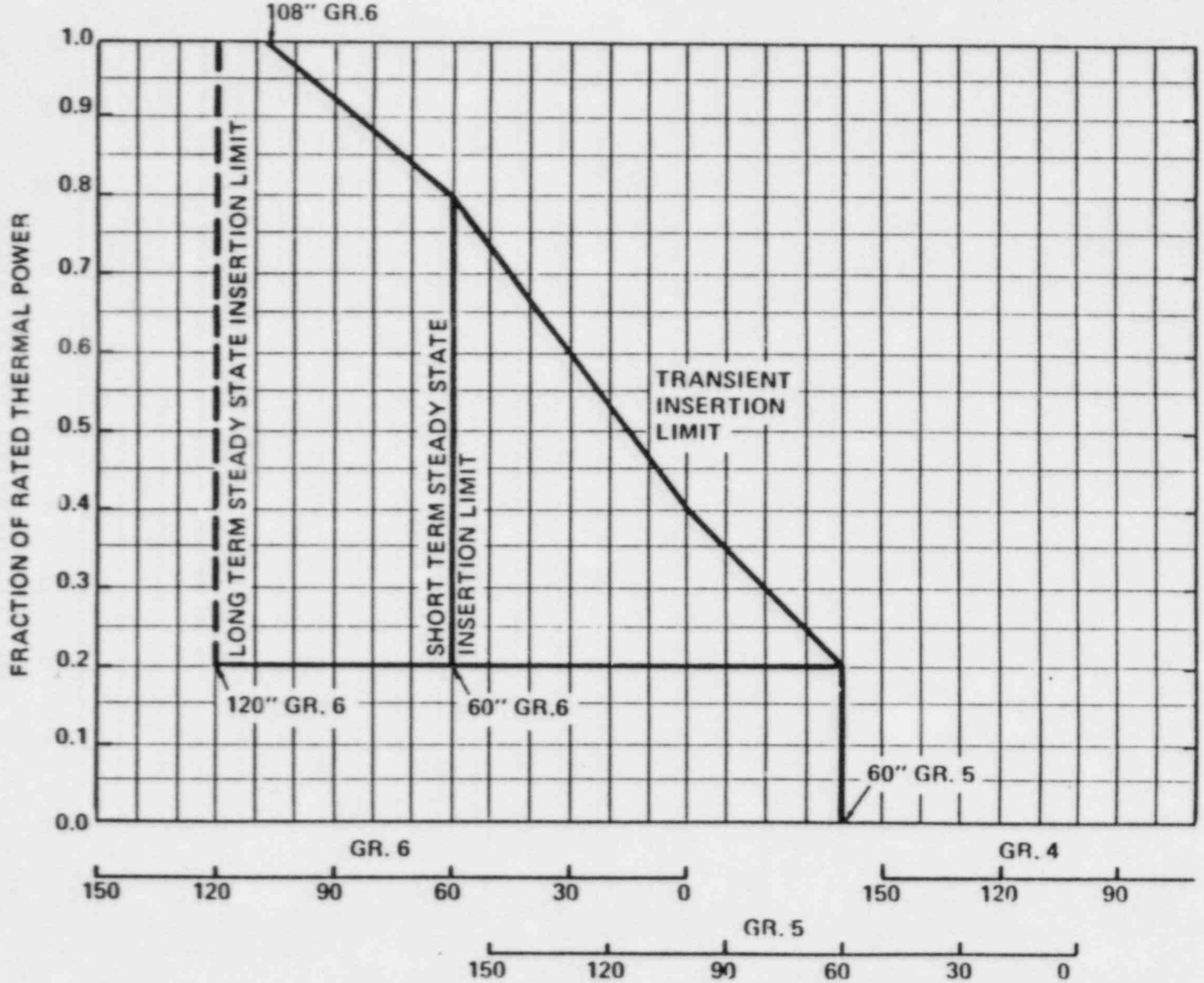
a. Restore the regulating CEA group to within the limit, or

b. Reduce thermal power to less than or equal to that fraction of Rated Thermal Power which is allowed by the CEA group position and the Short Term Steady State Insertion Limit.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

FIGURE 3.1-2


108" GR.6
LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE
INSERTION LIMIT
TRANSIENT
INSERTION
LIMIT
120" GR. 6
60" GR.6
60" GR. 5
FRACTION OF RATED THERMAL POWER
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
GR. 6
150
120
90
60
30
0
GR. 4
150
120
90
GR. 5
150
120
90
60
30
0
CEA WITHDRAWAL - INCHES

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable and the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

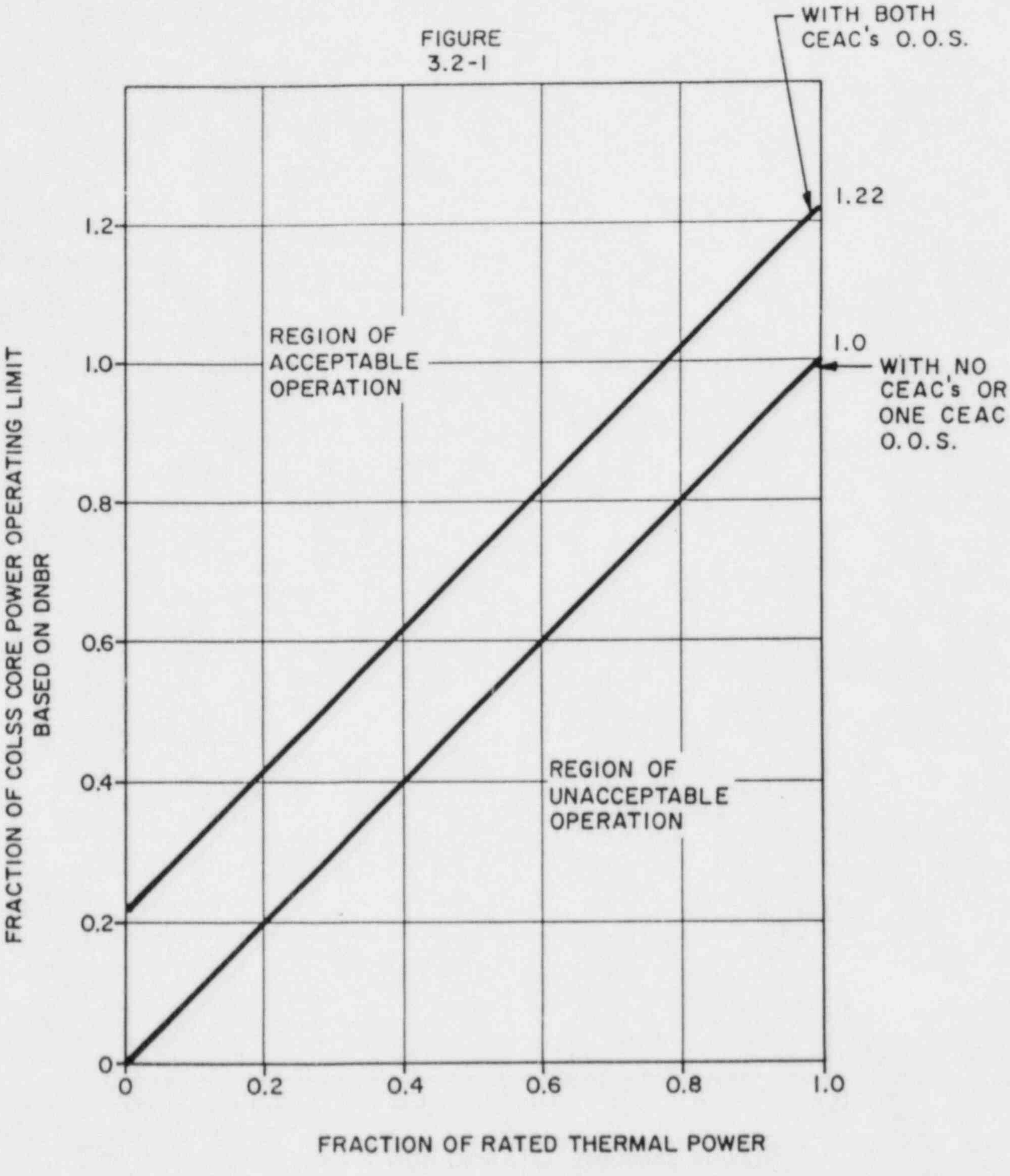


FIGURE 3.2-1 CYCLE 1 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

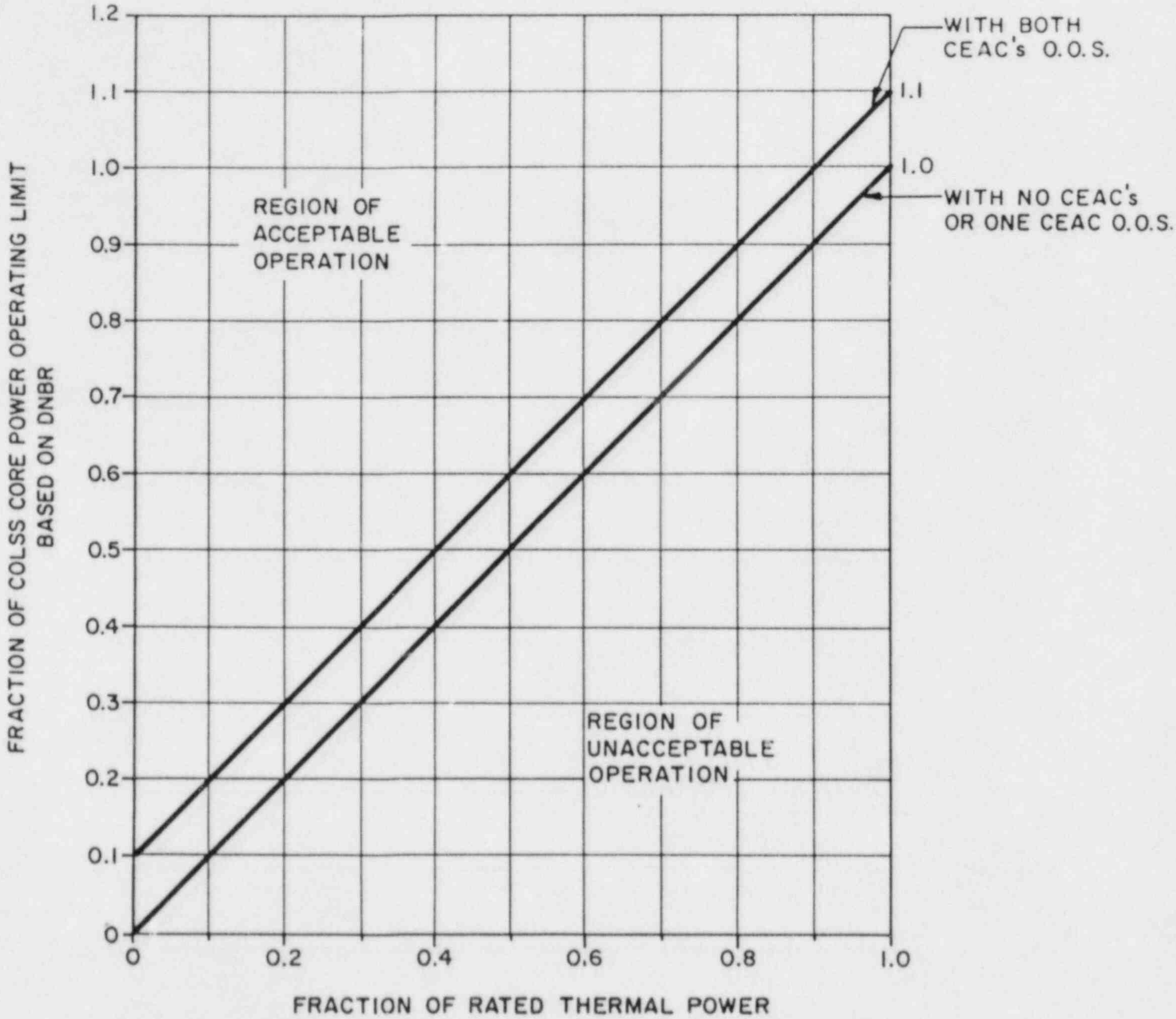


FIGURE 3.2-1 CYCLE 2 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

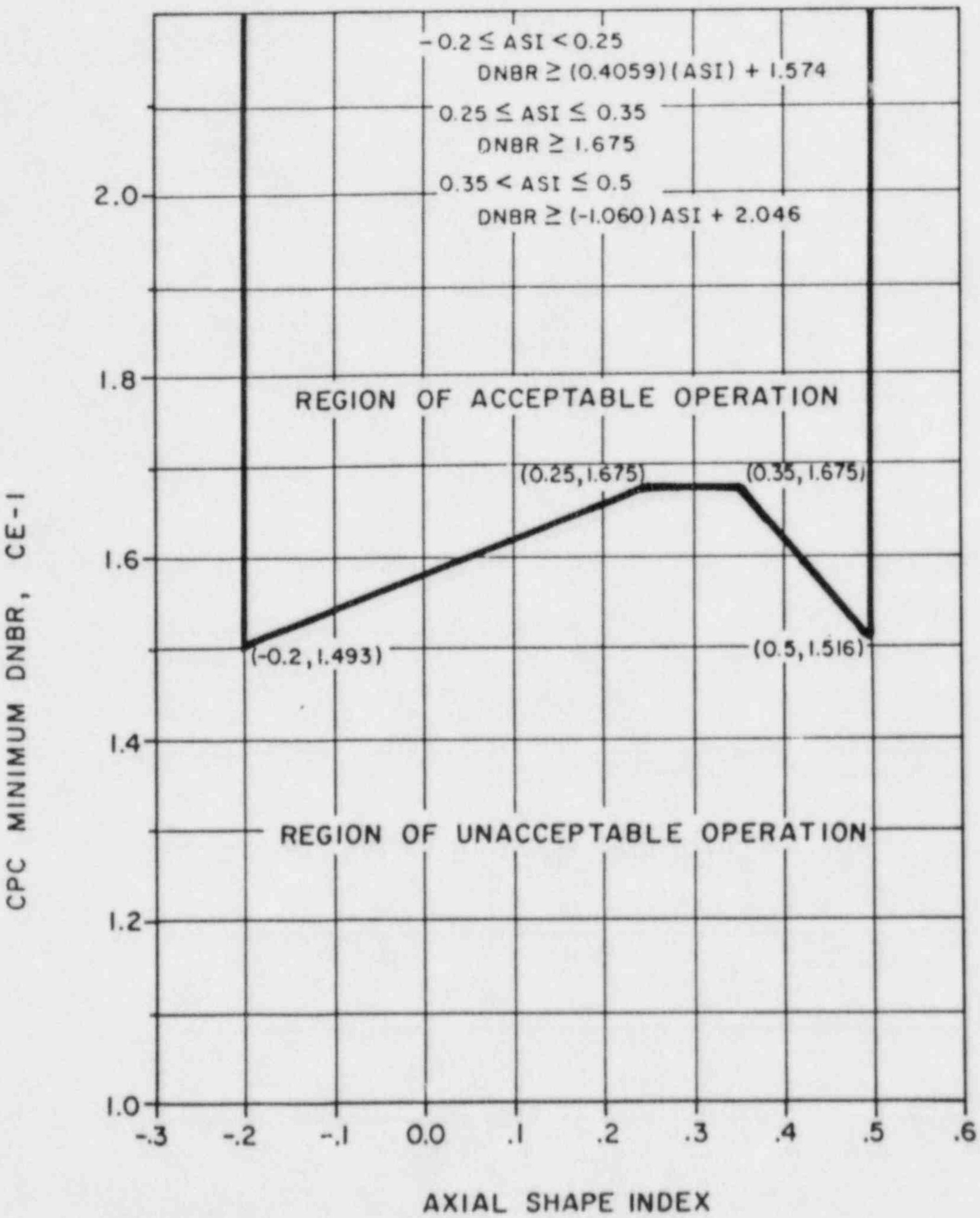


FIGURE 3.2-2 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER ≥ 80%)

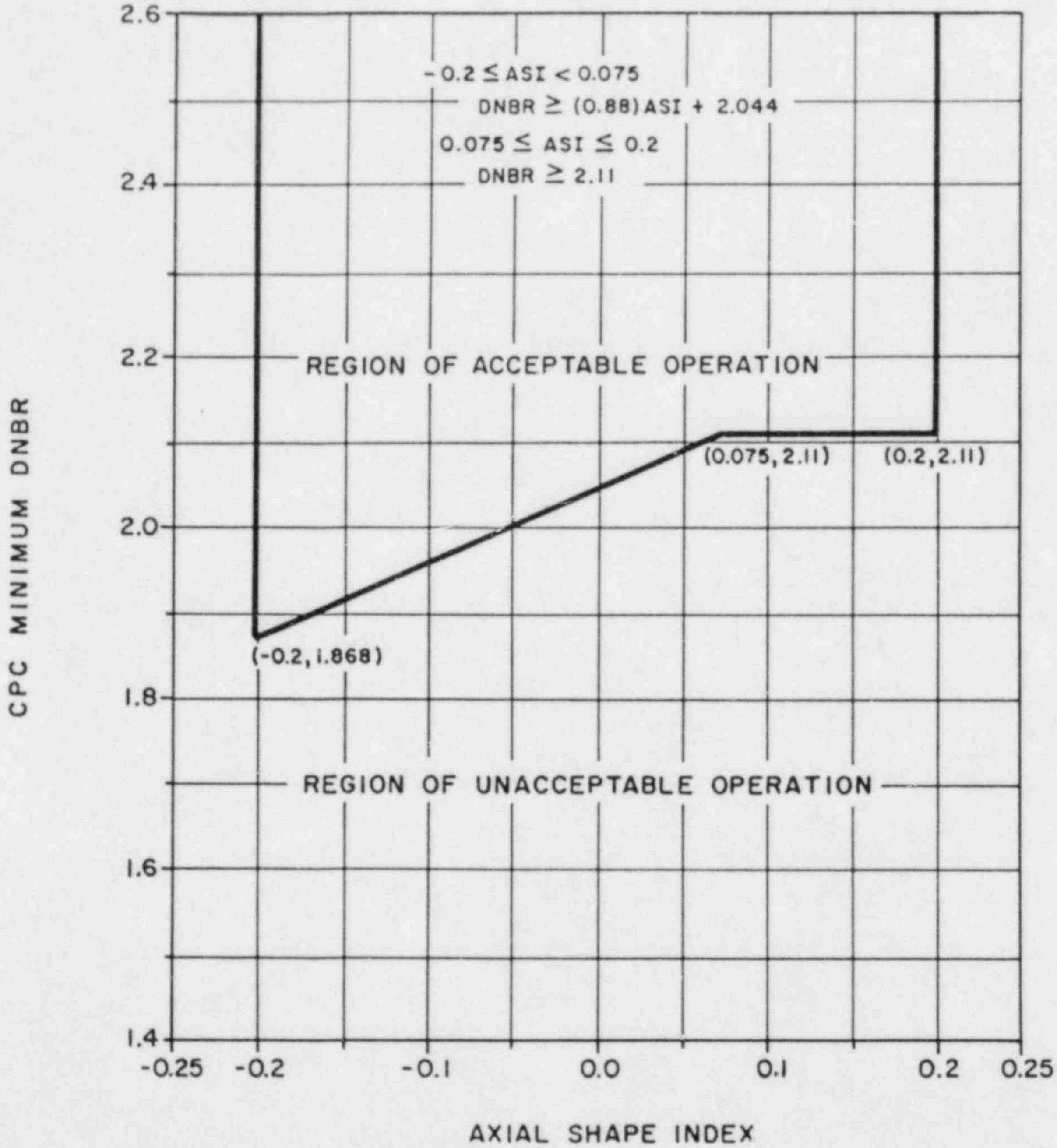


FIGURE 3.2-2 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER ≥ 70%)

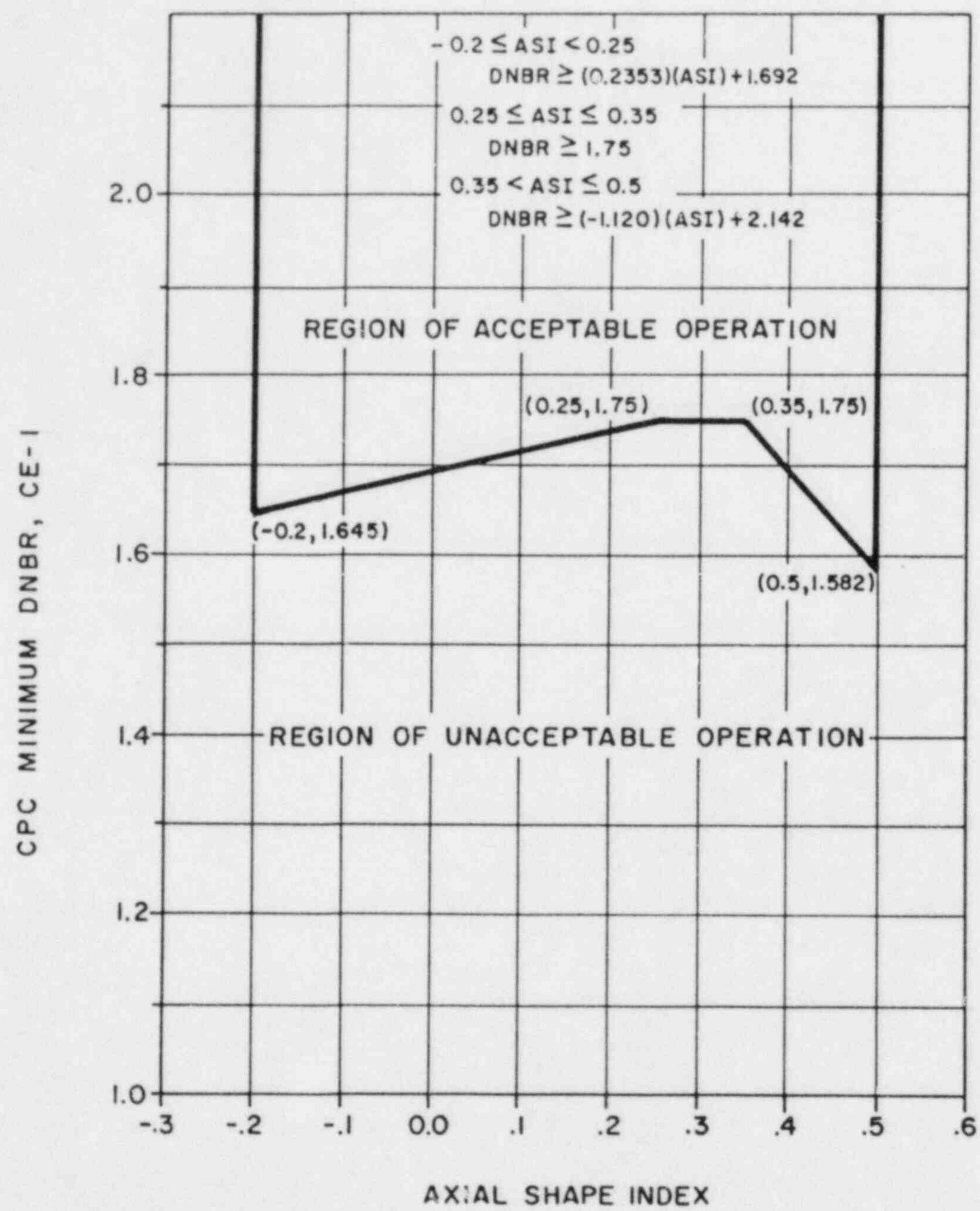


FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 80%)

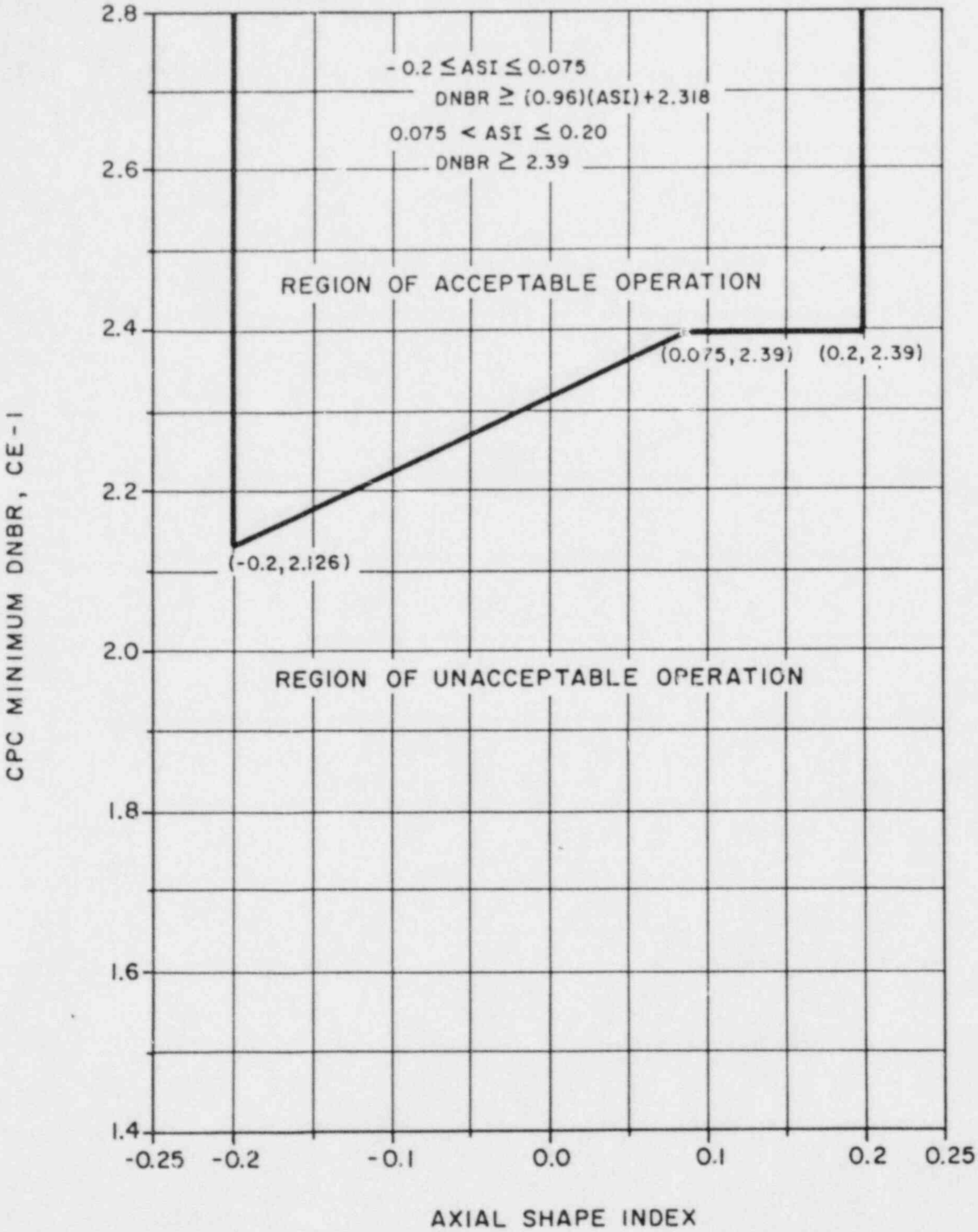


FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 70%)

POWER DISTRIBUTION LIMITS

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq ASI \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the pressurizer pressure exceeding its limit, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5. Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6. Core Protection Calculator	Local Power Density - High DNBR - Low

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that Action Items 6.b.1, .2 and .3 are met with COLSS in-service, or Action Items 6.c.1, .2 and .3 are met with COLSS out-of-service*.

b. With both CEAC's inoperable and COLSS in-service, operation may continue provided that:*

   1. Within 1 hour the DNBR margin operating limit required by Specification 3.2.4 (Figure 3.2-1) is satisfied for both CEAC's out-of-service.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE 3.3-1 (Continued)

TABLE NOTATION

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

c. With both CEAC's inoperable and COLSS out-of-service operation may continue provided that:*

   1. Within 1 hour multiply the CPC value of BERR1 corresponding to COLSS in-service by 1.13 (CYCLE 1) or 1.05 (CYCLE 2) and re-enter into the CPC's.

   2. Within 4 hours:

      a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

---

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2

TABLE 3.3-1 (Continued)

TABLE NOTATION

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	$\leq$ 0.40 seconds*
3. Logarithmic Power Level - High	$\leq$ 0.45 seconds*
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Pressurizer Pressure - Low	$\leq$ 0.90 seconds
6. Containment Pressure - High	$\leq$ 0.90 seconds
7. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
8. Steam Generator Level - Low	$\leq$ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 0.68 seconds*
b. CEA Positions	$\leq$ 0.68 seconds**
c. CEA Positions: CEAC Penalty Factor	$\leq$ 0.53 seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 0.68 seconds*
b. CEA Positions	$\leq$ 0.68 seconds**
c. Cold Leg Temperature	$\leq$ 0.68 seconds##
d. Hot Leg Temperature	$\leq$ 0.68 seconds##
e. Primary Coolant Pump Shaft Speed	$\leq$ 0.68 seconds#
f. Reactor Coolant Pressure from Pressurizer	$\leq$ 0.68 seconds
g. CEA positions: CEAC Penalty Factor	$\leq$ 0.53 seconds

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 13.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. Adjustments to the CPC addressable constants in Table 3.3-2a and reductions in the DNBR Power Operating Limit in Table 3.3-2b shall be made to accommodate measured values of RTD time constants.

TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD Delay Time τ	BERR0 Increase %		BERR2 Increase %		BERR4 Increase %	
	Cycle 1	Cycle 2	Cycle 1	Cycle 2	Cycle 1	Cycle 2
$\tau \leq 6.0$ sec.	0.0	0.0	0.0	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	0.0	2.0	3.5	5.0	3.0	0.0
8.0 sec. $< \tau \leq$ 10.0 sec.	3.5	5.0	4.0	8.5	9.0	3.0
10.0 sec. $< \tau \leq$ 13.0 sec.	10.5	9.0	5.5	12.0	17.0	6.0

NOTE: BERR term increases are not cumulative, i.e., if the values of the BERR terms are currently 10.0, then for an RTD delay time of > 6.0 to ≤ 8.0 sec., in Cycle 1: BERR0 = 10.0 + 0.0 = 10.0; BERR2 = 10.0 + 3.5 = 13.5; and, BERR4 = 10.0 + 3.0 = 13.0. For RTD delay times of > 8.0 to ≤ 10.0 sec., in Cycle 1: BERR0 = 10.0 + 3.5 = 13.5; BERR2 = 10.0 + 4.0 = 14.0; and BERR4 = 10.0 + 9.0 = 19.0. Computed values in this paragraph are examples only.

NOTE: In Cycle 1 only, when any of the above increases are applied to the BERR terms for any CPC channel, the COLSS constant EPOL2 is reduced by 0.04. This applies for Cycle 1 only.

TABLE 3.3-2b

DNBR LCO POWER OPERATING LIMIT ADJUSTMENTS

RTD Delay Time (sec)	Adjustment to EPOL1,[1] COLSS In Service (% power)	Adjustment to BERR2,[1,2] COLSS Out of Service (% power)	
		Cycle 1	Cycle 2
$\tau \leq 6.0$ sec.	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	-4.0	+4.0	+5.0
8.0 sec. $< \tau \leq$ 10.0 sec.	-5.0	+5.0	+8.5
10.0 sec. $< \tau \leq$ 13.0 sec.	-7.0	+7.0	+12.0

NOTES:

1. Adjustments are not cumulative: i.e., if τ increases from 7.0 seconds to 9.0 seconds, EPOL1 is reduced by 5.0 from its original value, not 4.0 + 5.0 = 9.0 from its original value.

2. If COLSS is out-of-service, these adjustments are to be used in place of, not in addition to, the increases required by Table 3.3-2a and the limit in Figure 3.2-2 or 3.2-3, as applicable, must be maintained for all operable CPC channels.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4), R(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and the noted requirements of Table 2.2-1 and Table 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each logarithmic and linear power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, or only regulating CEA Group 6 is inserted beyond the Transient Insertion Limit of Specification 3.1.3.6; and

b. The limits of Specifications 3.2.1 and 3.2.4 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specifications 3.2.1 or 3.2.4 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specifications 3.2.1 and 3.2.4, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate and DNBR margin shall be determined to be within the limits of Specifications 3.2.1 and 3.2.4, respectively, by monitoring them continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time-dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

MOVABLE CONTROL ASSEMBLIES (Continued)

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits. Setting the "RSPT/CEAC Inoperable" addressable constant in the CPC's to indicate to the CPC's that one or both of the CEAC's is inoperable does not necessarily constitute the inoperability of the RSPT rod indications from the respective CEAC. Operability of the CEAC rod indications is determined from the normal surveillance.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/ confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 or 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Uncertainty terms already taken into account in the CPC's safety monitoring are removed from Figures 3.2-2 and 3.2-3 since the curves are intended to monitor only the LCO during steady state operation.

The DNBR penalty factors listed in Section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps (RCPs)in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than four reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 with one or more RCS cold legs less than or equal to 285°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 285°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 285°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

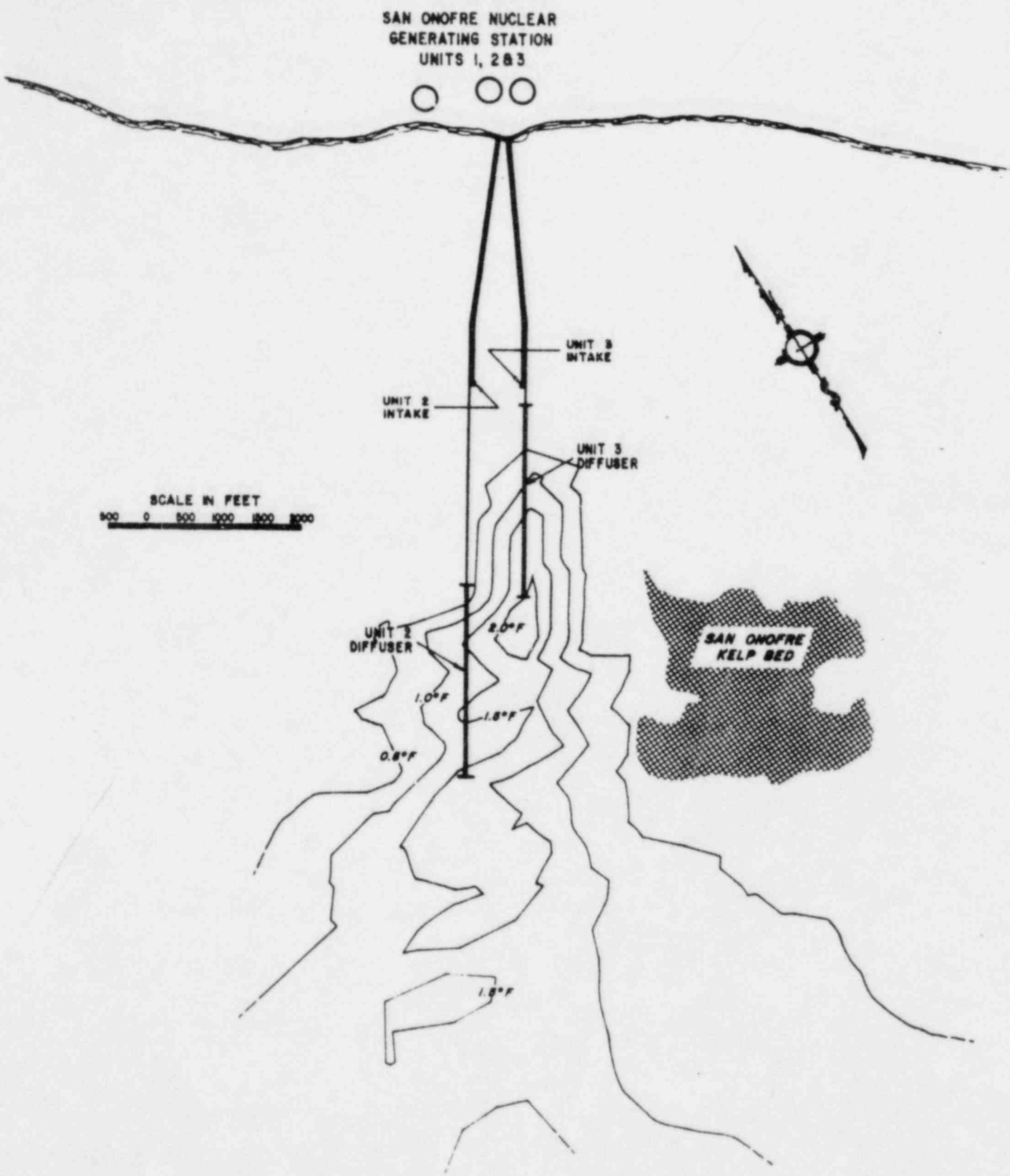
Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.



SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

This special test exception permits the center CEA to be misaligned or Regulating Group 6 inserted beyond the Transient Insertion Limit during PHYSICS TESTS required to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient.