

May 5, 1988

Docket No.: STN 50-529

Mr. E. E. Van Brunt, Jr.
Executive Vice President
Arizona Nuclear Power Project
Post Office Box 52034
Phoenix, Arizona 85072-2034

Dear Mr. Van Brunt:

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SUBJECT: ISSUANCE OF AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-51
FOR THE PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2
(TAC NO. 66784)

The Commission has issued the subject Amendment, which is enclosed, to the Facility Operating License for Palo Verde Nuclear Generating Station, Unit 2. The Amendment consists of changes to the Technical Specifications (Appendix A to the license) in response to your application transmitted by letter dated December 3, 1987, as supplemented by the supporting Reload Analysis Report, dated December 2, 1987 and the clarifying information on the Report submitted on February 4, 1988 in response to staff questions.

The Amendment revises several portions of the Technical Specifications to incorporate changes in support of Cycle 2 operation for Palo Verde, Unit 2. One of the proposed changes to the Technical Specifications in your amendment request, relating to Azimuthal tilt limits, is still under staff review. The completion of review for this request, which is not required for restart of Palo Verde, Unit 2, is pending on receipt of the additional information that we understand is currently in preparation.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/

E. A. Licitra, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

8805200319 880505
PDR ADDCK 05000529
P PDR

Enclosures:

1. Amendment No. 19 to NPF-51
2. Safety Evaluation

cc: See next page

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04/18/88 04/23/88

OGC-WF: *Joe Licitra*
04/21/88

12 for 6K
DRSP/D:PDV *AK*
GWKnighton
04/23/88 *SK/88*

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Sincerely,

/s/

E. A. Licitra, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:

1. Amendment No. 19 to NPF-51
2. Safety Evaluation

cc: See next page

DRSP/PDV *EAL* DRSP/PDV
EALicitra: JLee
04/18/88 04/18/88

OGC-WF: *Joe Licitra*
04/27/88

HL for 6K
DRSP/D:PDV *HL*
GWKnighton
04/23/88 *SK/SP*



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

May 5, 1988

Docket No.: STN 50-529

Mr. E. E. Van Brunt, Jr.
Executive Vice President
Arizona Nuclear Power Project
Post Office Box 52034
Phoenix, Arizona 85072-2034

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SUBJECT: ISSUANCE OF AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-51
FOR THE PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2
(TAC NO. 66784)

The Commission has issued the subject Amendment, which is enclosed, to the Facility Operating License for Palo Verde Nuclear Generating Station, Unit 2. The Amendment consists of changes to the Technical Specifications (Appendix A to the license) in response to your application transmitted by letter dated December 3, 1987, as supplemented by the supporting Reload Analysis Report, dated December 2, 1987 and the clarifying information on the Report submitted on February 4, 1988 in response to staff questions.

The Amendment revises several portions of the Technical Specifications to incorporate changes in support of Cycle 2 operation for Palo Verde, Unit 2. One of the proposed changes to the Technical Specifications in your amendment request, relating to Azimuthal tilt limits, is still under staff review. The completion of review for this request, which is not required for restart of Palo Verde, Unit 2, is pending on receipt of the additional information that we understand is currently in preparation.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

A handwritten signature in cursive script, reading "E. A. Licitra", is positioned above the typed name.

E. A. Licitra, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:

1. Amendment No. 19 to NPF-51
2. Safety Evaluation

cc: See next page

Mr. E. E. Van Brunt, Jr.
Arizona Nuclear Power Project

Palo Verde

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Arizona Nuclear Power Project

- 2 -

Palo Verde

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 19
License No. NPF-51

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment, dated December 3, 1987, as supplemented by letters dated December 2, 1987 and February 4, 1988, by the Arizona Public Service Company (APS) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of 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;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;
 - 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 enclosure to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-51 is hereby amended to read as follows:


8805200323 880505
PDR ADOCK 05000529
P PDR

(2) Technical Specifications and Environmental Protection Plan

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

3. This license amendment is effective as of the date of issuance. The changes in the Technical Specifications are to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensees shall adhere to the Technical Specifications existing at the time. The period of time during changeover shall be minimized.

FOR THE NUCLEAR REGULATORY COMMISSION


George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1988

May 5, 1988

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-51

DOCKET NO. STN 50-529

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 areas of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages

Overleaf Pages

IV	III
XIX	-
XX	-
2-1	2-2
2-3	2-4
2-5	2-6
B 2-1	-
B 2-2	-
B 2-3	B 2-4
B 2-5	-
B 2-6	-
3/4 1-2a	-
3/4 1-5	3/4 1-6
3/4 1-21	-
3/4 1-22	-
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3/4 1-25	3/4 1-26
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3/4 1-32	-
3/4 1-33	-
3/4 1-34	-
3/4 2-1	3/4 2-2
3/4 2-5	-
3/4 2-6	-
3/4 2-7	-
3/4 2-7a	-
3/4 2-8	-
3/4 2-12	3/4 2-11

DOCKET NO. STN 50-529

Amendment Pages

Overleaf Pages

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3/4 3-8	-
3/4 3-9	-
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3/4 3-11	-
3/4 3-12	-
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3/4 3-26	3/4 3-25
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3/4 10-4	3/4 10-3
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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to 1.24.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor has decreased to less than 1.24, 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 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 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.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	≤ 2383 psia	≤ 2388 psia
2. Pressurizer Pressure - Low	≥ 1837 psia (2)	≥ 1822 psia (2)
3. Steam Generator Level - Low	$\geq 44.2\%$ (4)	$\geq 43.7\%$ (4)
4. Steam Generator Level - High	$\leq 91.0\%$ (9)	$\leq 91.5\%$ (9)
5. Steam Generator Pressure - Low	≥ 919 psia (3)	≥ 912 psia (3)
6. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
7. Reactor Coolant Flow - Low		
a. Rate	≤ 0.115 psi/sec (6)(7)	≤ 0.118 psi/sec (6)(7)
b. Floor	≥ 11.9 psid(6)(7)	≥ 11.7 psid (6)(7)
c. Band	≤ 10.0 psid(6)(7)	≤ 10.2 psid (6)(7)
8. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
9. DNBR - Low	≥ 1.24 (5)	≥ 1.24 (5)
B. Excore Neutron Flux		
1. Variable Overpower Trip		
a. Rate	$< 10.6\%$ /min of RATED THERMAL POWER (8)	$< 11.0\%$ /min of RATED THERMAL POWER (8)
b. Ceiling	$< 110.0\%$ of RATED THERMAL POWER (8)	$< 111.0\%$ of RATED THERMAL POWER (8)
c. Band	$< 9.8\%$ of RATED THERMAL POWER (8)	$< 10.0\%$ of RATED THERMAL POWER (8)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	< 0.010% of RATED THERMAL POWER	< 0.011% of RATED THERMAL POWER
b. Shutdown	< 0.010% of RATED THERMAL POWER	< 0.011% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	≤ 2409 psia	≤ 2414 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

- (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) In MODES 3-4, value may be decreased manually, to a minimum of 100 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) In MODES 3-4, 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 lower level wide range instrument nozzles.
- (5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties. 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.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS (Continued)

- (6) RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are based on steam generator differential pressure.
- (7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.
- (8) RATE is the maximum rate of increase of the trip setpoint. (The rate at which the setpoint can decrease is no slower than five percent per second.)
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the steady state input signal unless limited by the rate or the ceiling.
- (9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

2.1 and 2.2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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 operation and design basis anticipated operational occurrences is limited to 1.24 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit. The DNBR limit of 1.24 includes a rod bow compensation of 1.75% on DNBR.

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. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

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, 1974 Edition, Summer 1975 Addendum, 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 is 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 Safety Limits of 1.24 and 21 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 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 the latest applicable revision of CEN-305-P, "Functional Design Requirements for a Core Protection Calculator," and CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator."

REACTOR TRIP SETPOINTS (Continued)

The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V), Rev. 2, dated August 29, 1986.

Manual Reactor Trip

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

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10⁻⁴% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10⁻⁴% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

BASES

Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

BASES

Local Power Density - High (Continued)

- a. Nuclear flux power and axial power distribution from the excore flux monitoring system;
- b. Radial peaking factors from the position measurement for the CEAs;
- c. Delta T power from reactor coolant temperatures and coolant flow measurements.

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 Linear Heat Rate 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 design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1860 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.

SAFETY LIMITS AND LIMITING SAFETY SYSTEMS SETTINGS

BASES

DNBR - Low (Continued)

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 calculated core DNBR is sufficiently greater than 1.24 such that the decrease in calculated core 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.

	<u>Parameter</u>	<u>Limiting Value</u>
a.	RCS Cold Leg Temperature-Low	> 470°F
b.	RCS Cold Leg Temperature-High	< 610°F
c.	Axial Shape Index-Positive	Not more positive than + 0.5
d.	Axial Shape Index-Negative	Not more negative than - 0.5
e.	Pressurizer Pressure-Low	> 1860 psia
f.	Pressurizer Pressure-High	< 2388 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	< 4.28
i.	Quality Margin-Low	> 0

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carryover. This trip's setpoint does not correspond to a safety limit, and provides protection in the event of excess feedwater flow. The setpoint is identical to the main steam isolation setpoint. Its functional capability at the specified trip setting enhances the overall reliability of the reactor protection system.

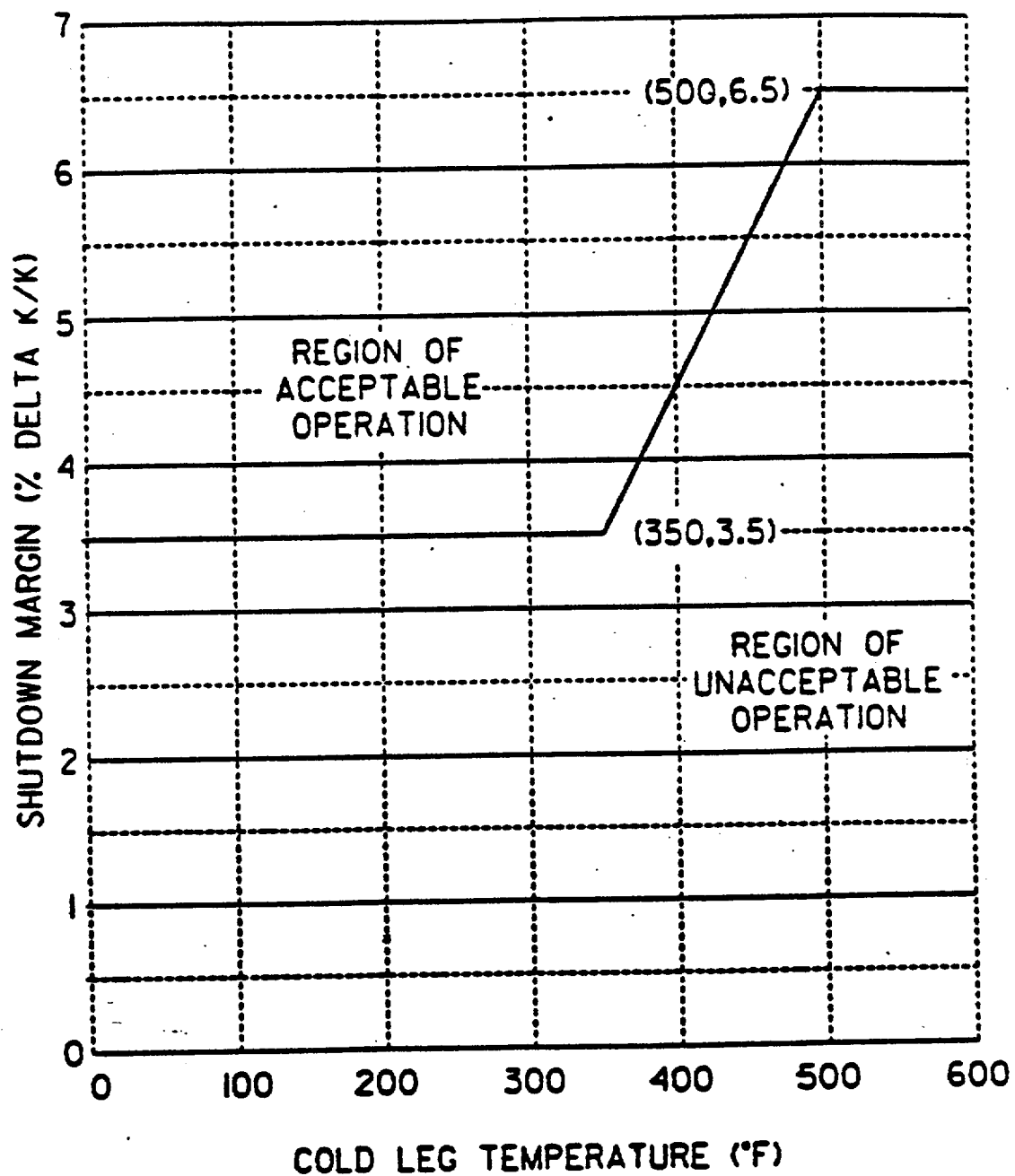


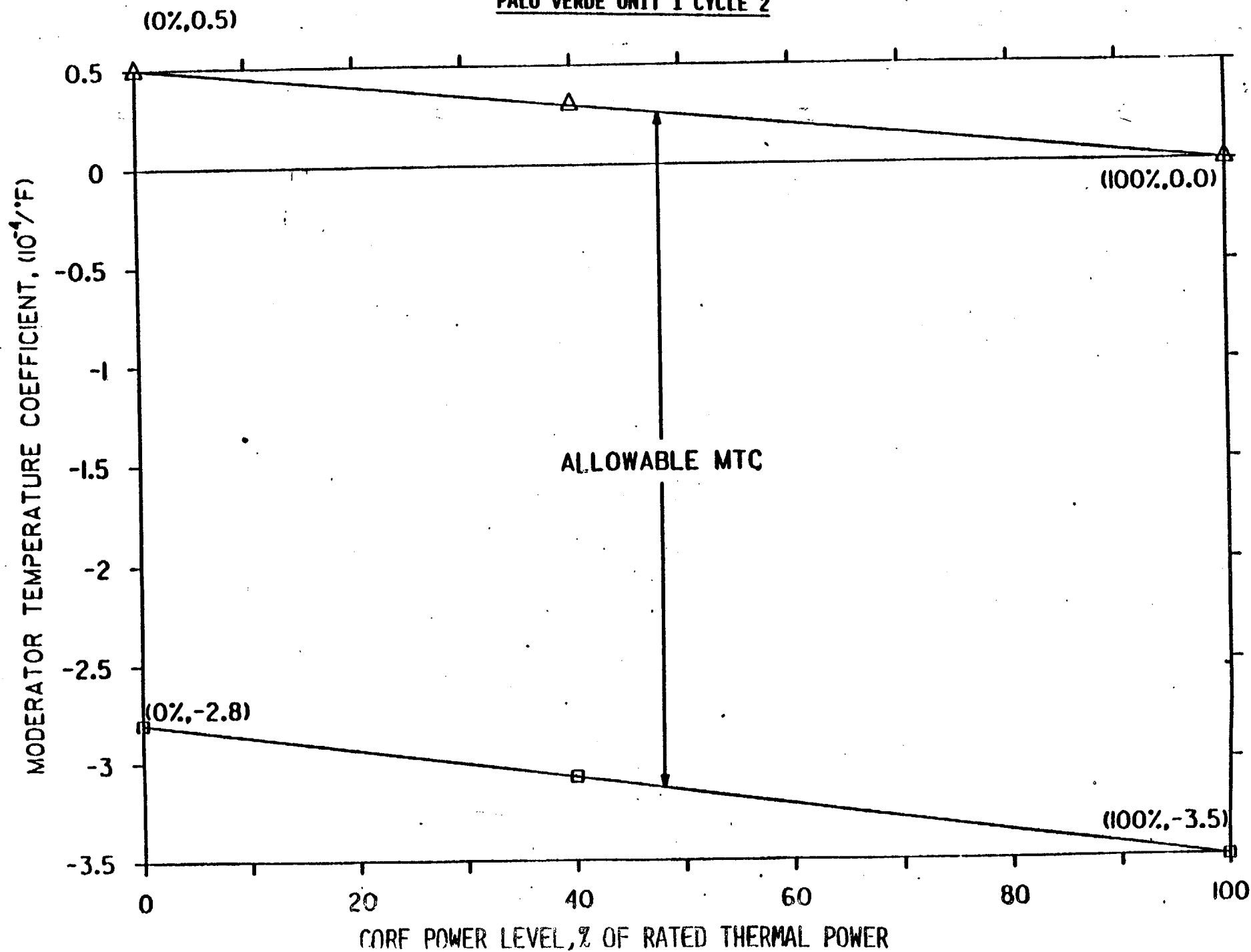
FIGURE 3.1- 1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE

FIGURE 3.1-1

ALLOWABLE MTC MODES 1 AND 2

PALO VERDE UNIT 1 CYCLE 2



MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{cold}) shall be greater than or equal to 552°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{cold}) less than 552°F, restore T_{cold} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{cold}) shall be determined to be greater than or equal to 552°F:

- a. Within 15 minutes prior to achieving reactor criticality, and
- b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{cold} is less than 557°F.

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

*See Special Test Exception 3.10.5.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

- a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.2 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.
- b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.
- c. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-2A and that within 1 hour the misaligned CEA(s) is either:
 1. Restored to OPERABLE status within its above specified alignment requirements, or
 2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.2 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specifications 3.1.3.6 and 3.1.3.7 provided:
 - a) - Within 1 hour the remainder of the CEAs in the group with the inoperable CEA(s) shall be aligned to within 6.6 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figures 3.1-3 and 3.1-4; the THERMAL POWER level shall be restricted pursuant to Specifications 3.1.3.6 and 3.1.3.7 during subsequent operation.

*See Special Test Exceptions 3.10.2 and 3.10.4.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.2 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

- d. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.
- e. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group and the CEA is maintained pursuant to the requirements of Specification 3.1.3.7.

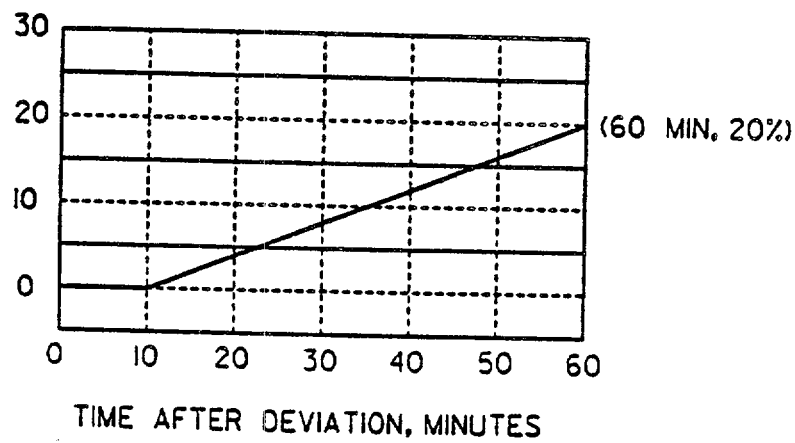
SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

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MINIMUM REQUIRED POWER REDUCTION
(% OF RATED THERMAL POWER)



• WHEN CORE POWER IS REDUCED TO 55% OF RATED THERMAL POWER PER THIS LIMIT CURVE, FURTHER REDUCTION IS NOT REQUIRED

FIGURE 3.1-2A

CORE POWER LIMIT AFTER CEA DEVIATION*

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

- a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,
- b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and
- c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

- a. Restore the inoperable position indicator channel to OPERABLE status, or
- b. Be in at least HOT STANDBY, or
- c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.1.3.7. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.*

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part-length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

* With the reactor trip breakers in the closed position.

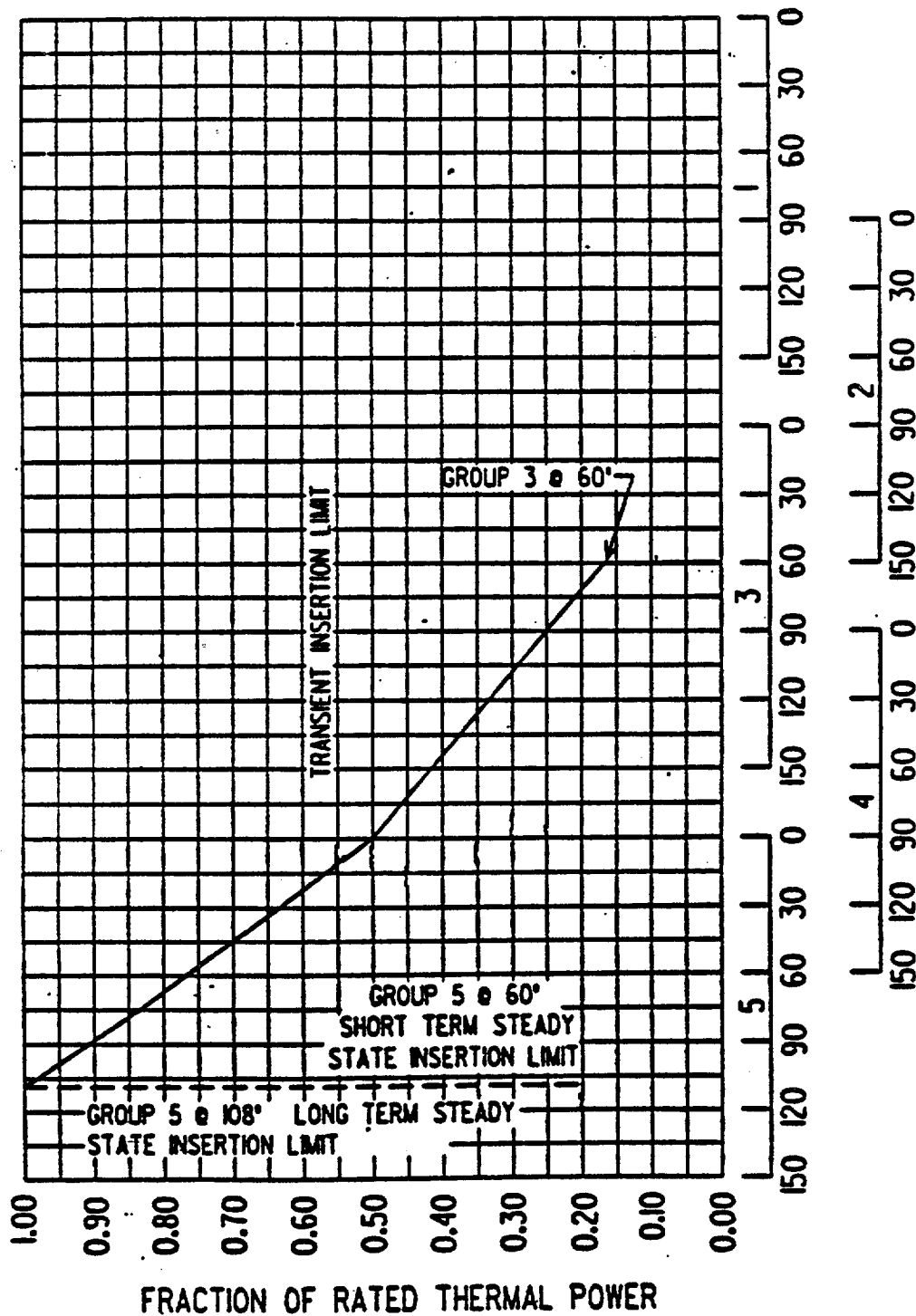


FIGURE 3.1-3
CEA INSERTION LIMITS VS THERMAL POWER
(COLSS IN SERVICE)

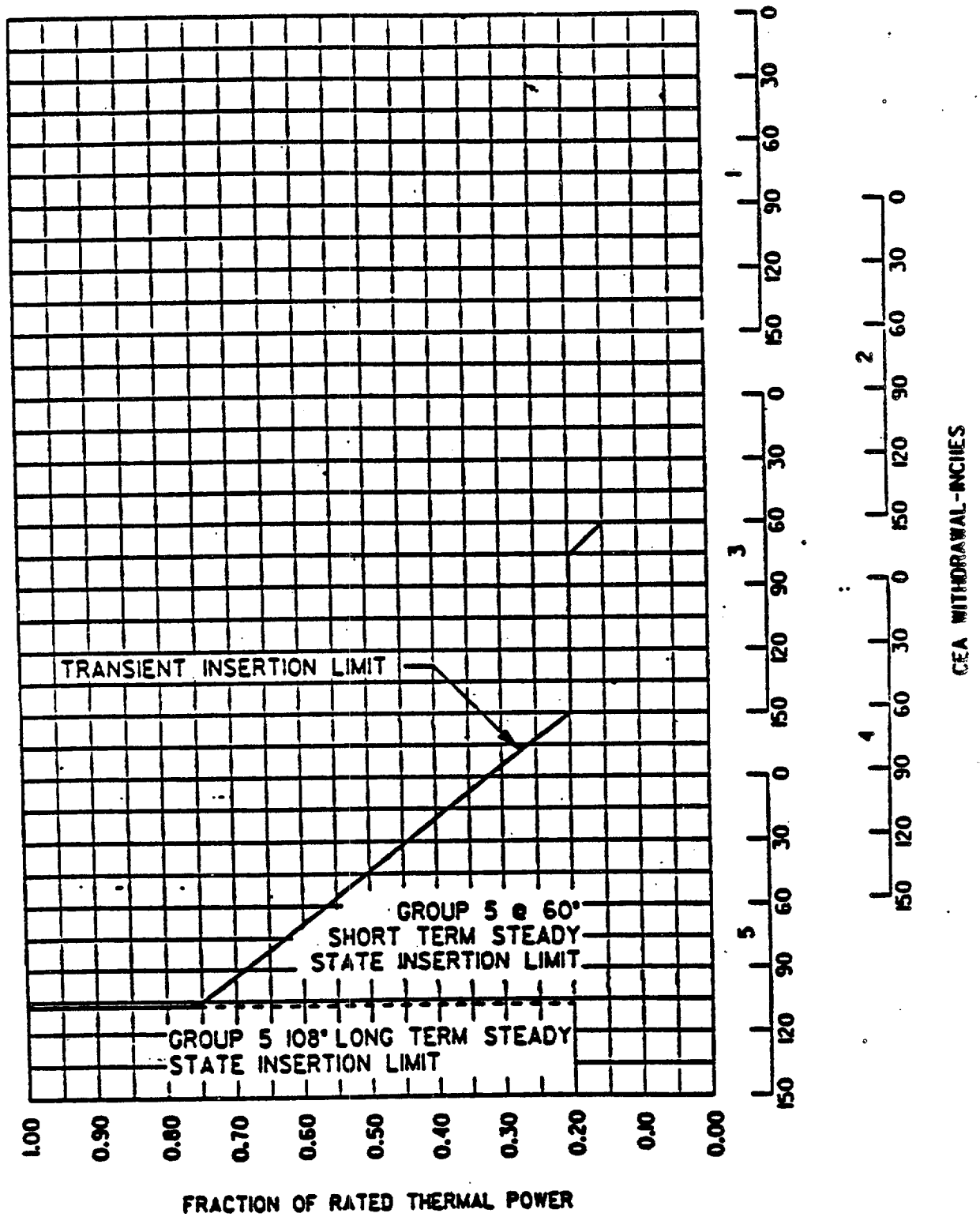


FIGURE 3.1-4

CEA INSERTION LIMITS VS THERMAL POWER
(COLSS OUT OF SERVICE)

REACTIVITY CONTROL SYSTEMS

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The part length CEA groups shall be limited to the insertion limits shown on Figure 3.1-5 with PLCEA insertion between the Long Term Steady State Insertion Limit and the Transient Insertion Limit restricted to:

- a. ≤ 7 EFPD per 30 EFPD interval, and
- b. ≤ 14 EFPD per calendar year.

APPLICABILITY: MODELS 1* and 2*

ACTION:

- a. With the part length CEA groups inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours, either:
 - 1. Restore the part length CEA group to within the limits, or
 - 2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the PLCEA group position using Figure 3.1-5.
- b. With the part length CEA groups inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 7 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:
 - 1. Restore the part length group within the Long Term Steady State Insertion Limit within two hours, or
 - 2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined to be within the Transient Insertion Limit at least once per 12 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

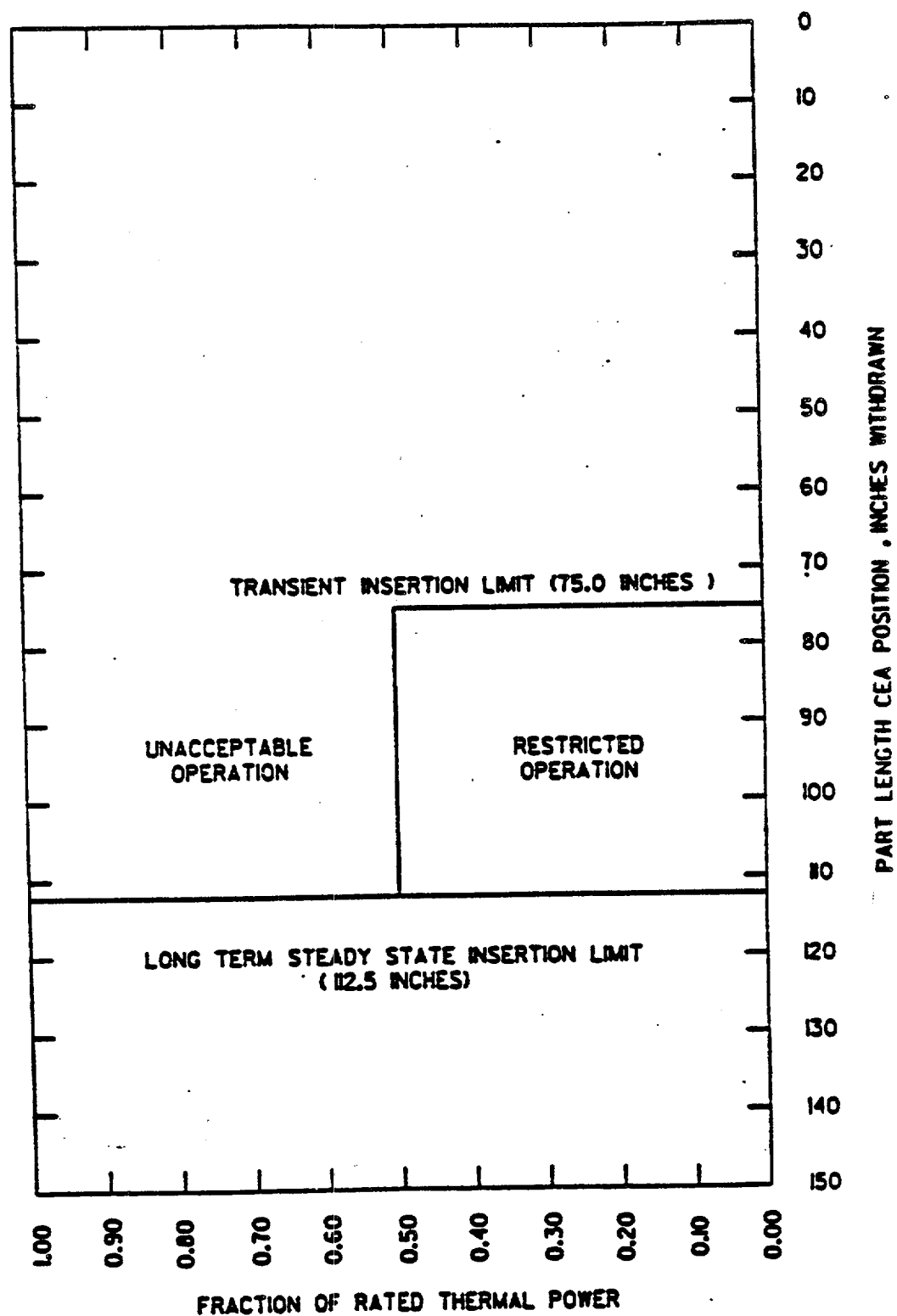


FIGURE 3.1-5
PART LENGTH CEA INSERTION LIMIT VS THERMAL POWER

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit of 13.5 kW/ft shall be maintained by one of the following methods as applicable:

- a. Maintaining COLSS calculated core power less than or equal to the COLSS calculated power operating limit based on linear heat rate (when COLSS is in service); or
- b. Maintaining peak linear heat rate within its limit using any operable CPC channel (when COLSS is out of service).

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

ACTION:

With the linear heat rate limit not being maintained as indicated by:

1. COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate; or
2. Peak linear heat rate outside its limit using any operable CPC channel (when COLSS is out of service);

within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

- a. Restore the linear heat rate to within its limits within 1 hour, or
- b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limit 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 linear heat rate, as indicated on any OPERABLE Local Power Density channel, is within its limit.

4.2.1.3 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 linear heat rate.

POWER DISTRIBUTION LIMITS

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}^m

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^m) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F_{xy}^c) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

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

ACTION:

With an F_{xy}^m exceeding a corresponding F_{xy}^c , within 6 hours either:

- Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F_{xy}^m/F_{xy}^c and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F_{xy}^m/F_{xy}^c) - 1.0] \times 100\%$ is maintained; or
- Adjust the affected PLANAR RADIAL PEAKING FACTORS (F_{xy}^c) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^m) or
- Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^m) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F_{xy}^c), used in the COLSS and CPC at the following intervals:

- After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and
- At least once per 31 Effective Full Power Days.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

- a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both CEACs are operable); or
- b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the allowance shown in Figure 3.2-1 (when COLSS is in service and neither CEAC is operable); or
- c. Operating within the region of acceptable operation of Figure 3.2-2 using any operable CPC channel (when COLSS is out of service and either one or both CEACs are operable); or
- d. Operating within the region of acceptable operation of Figure 3.2-2A using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

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

ACTION:

With the DNBR not being maintained:

1. As indicated by COLSS calculated core power exceeding the appropriate COLSS calculated power operating limit; or
2. With COLSS out of service, operation outside the region of acceptable operation of Figure 3.2-2 or 3.2-2A, as applicable;

within 15 minutes initiate corrective action to increase the DNBR to within the limits and either:

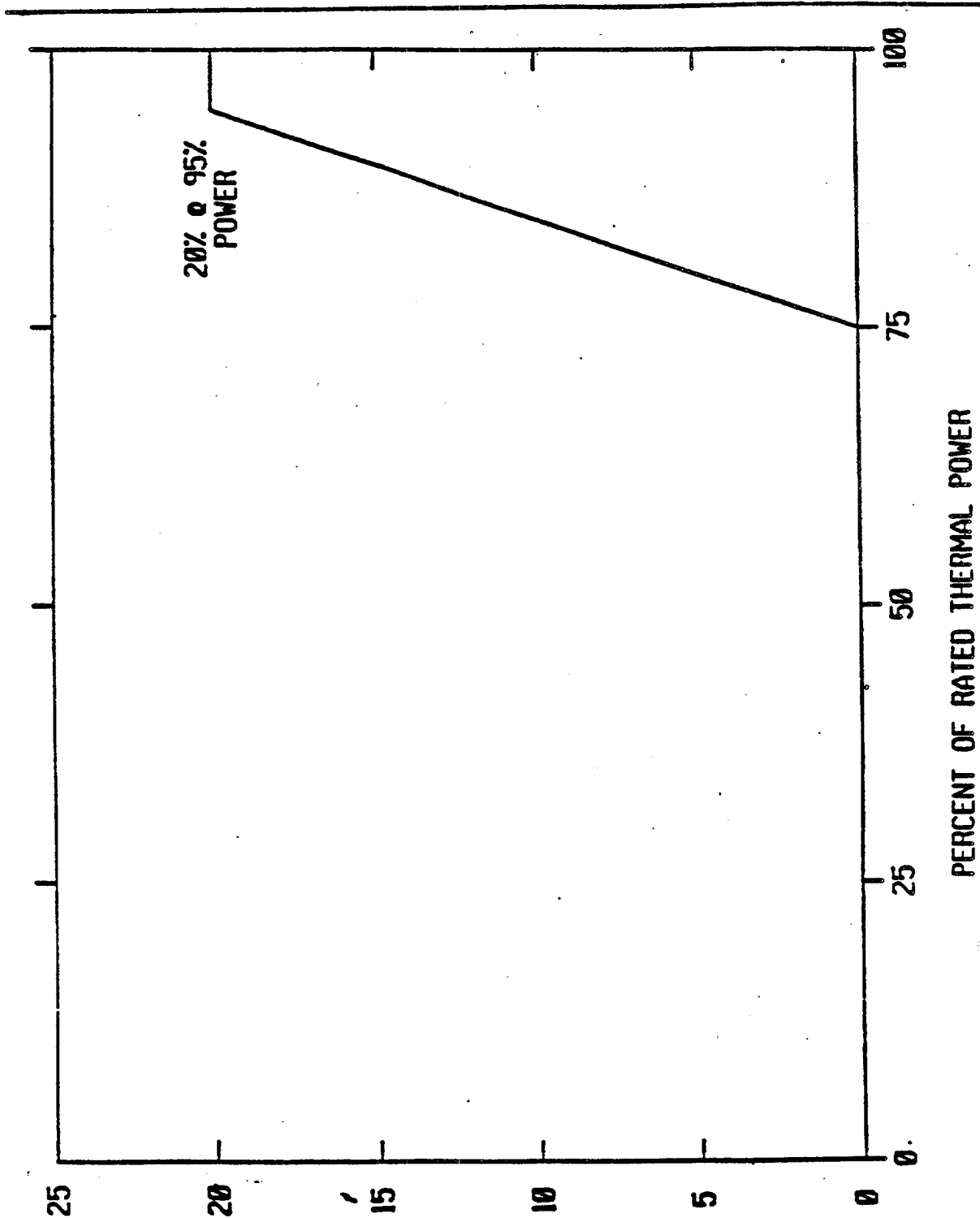
- a. Restore the DNBR to within its limits within 1 hour, or
- b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

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 any OPERABLE DNBR channel, is within the limit shown on Figure 3.2-2 or Figure 3.2-2A.

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.



COLSS DNBR POWER OPERATING LIMIT REDUCTION
(% OF RATED THERMAL POWER)

FIGURE 3.2-1

COLSS DNBR POWER OPERATING LIMIT
ALLOWANCE FOR BOTH CEACs INOPERABLE

COLSS OUT OF SERVICE DNBR LIMIT LINE

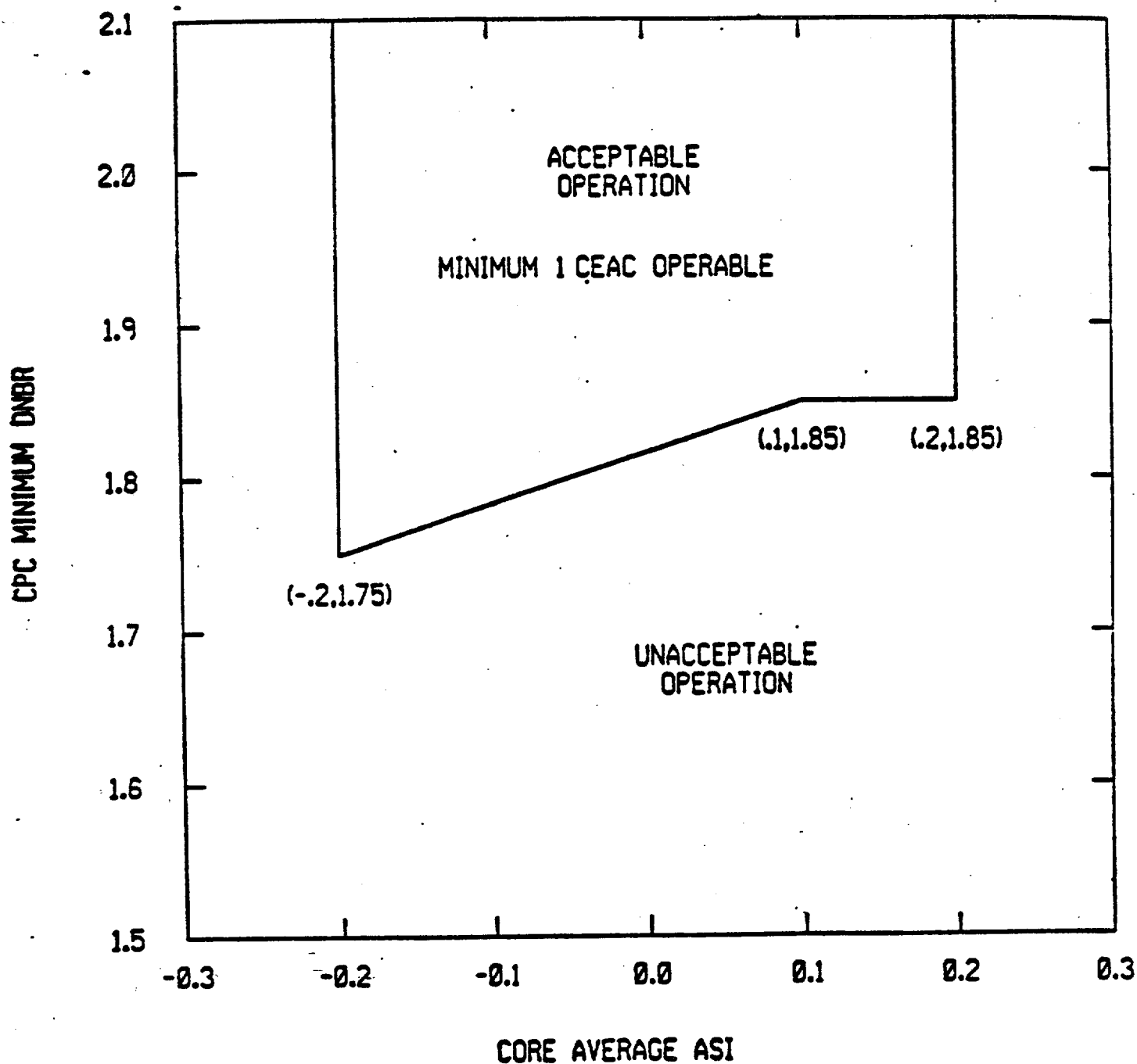


FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEACs OPERABLE)

COLSS OUT OF SERVICE DNBR LIMIT LINE

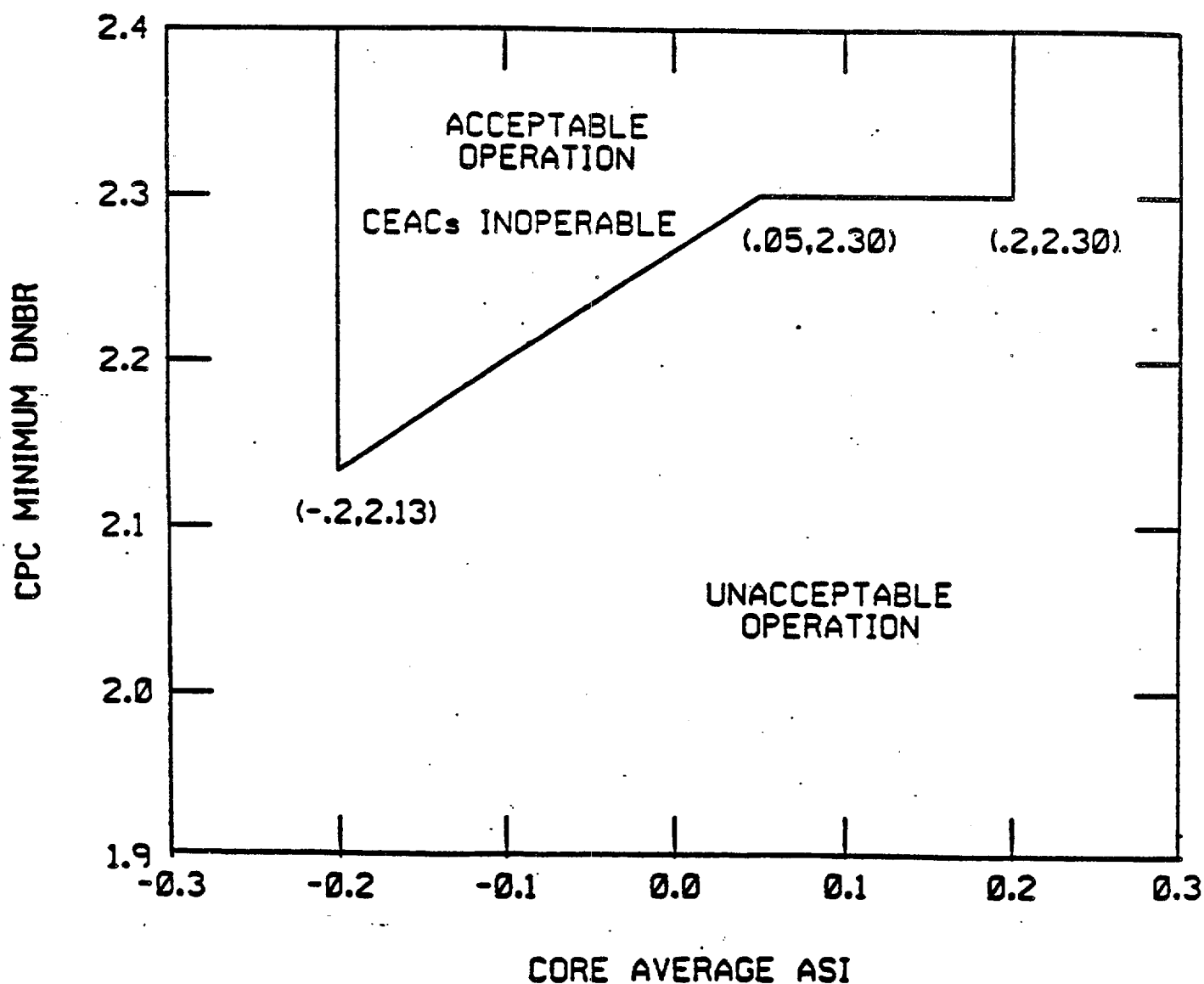


FIGURE 3.2-2A

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEACs INOPERABLE)

POWER DISTRIBUTION LIMITS

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 155.8×10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to its limit at least once per 12 hours.

POWER DISTRIBUTION LIMITS

3/4.2.7 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 outside its above limits, restore the core average 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.

POWER DISTRIBUTION LIMITS

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 2025 psia and 2300 psia.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.5

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

- | | | |
|----|--|---|
| 3. | Steam Generator Pressure - Low | - Steam Generator Pressure - Low
Steam Generator Level 1-Low (ESF)
Steam Generator Level 2-Low (ESF) |
| 4. | Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)
Steam Generator Level 1-Low (ESF)
Steam Generator Level 2-Low (ESF) |
| 5. | Core Protection Calculator | Local Power Density - High (RPS)
DNBR - Low (RPS) |

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, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

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 6.6 inches (indicated position) of all other CEAs in its group. After 7 days, operation may continue provided that the conditions of Action Item 6.b are met.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the DNBR margin required by Specification 3.2.4.b (COLSS in service) or 3.2.4.d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

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 5 may be inserted no further than 127.5 inches withdrawn.
 - b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to indicate that both CEACs are inoperable.
 - c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 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 CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), 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 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

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TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

<u>FUNCTIONAL UNIT</u>	<u>RESPONSE TIME</u>
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 1.15 seconds
2. Pressurizer Pressure - Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ 0.58 second
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. Cold Leg Temperature	≤ 0.75 second##
d. Hot Leg Temperature	≤ 0.75 second##
e. Primary Coolant Pump Shaft Speed	≤ 0.30 second#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.75 second###
g. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 0.55 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

<u>FUNCTIONAL UNIT</u>	<u>RESPONSE TIME</u>
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

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

** Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 8 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

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TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE REQUIRED</u>
I. TRIP GENERATION				
A. Process				
1. Pressurizer Pressure - High	S	R	M	1, 2
2. Pressurizer Pressure - Low	S	R	M	1, 2
3. Steam Generator Level - Low	S	R	M	1, 2
4. Steam Generator Level - High	S	R	M	1, 2
5. Steam Generator Pressure - Low	S	R	M	1, 2, 3*, 4*
6. Containment Pressure - High	S	R	M	1, 2
7. Reactor Coolant Flow - Low	S	R	M	1, 2
8. Local Power Density - High	S	D (2, 4), R (4, 5)	M, R (6)	1, 2
9. DNBR - Low	S	D (2, 4), R (4, 5) M (8), S (7)	M, R (6)	1, 2
B. Excore Neutron Flux				
1. Variable Overpower Trip	S	D (2, 4), M (3, 4) Q (4)	M	1, 2
2. Logarithmic Power Level - High	S	R (4)	M and S/U (1)	1, 2, 3, 4, 5 and *
C. Core Protection Calculator System				
1. CEA Calculators	S	R	M, R (6)	1, 2
2. Core Protection Calculators	S	D (2, 4), R (4, 5) M (8), S (7)	M (9), R (6)	1, 2

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>ESFA SYSTEM FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia ⁽¹⁾	≥ 1822 psia ⁽¹⁾
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
II. CONTAINMENT ISOLATION (CIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia ⁽¹⁾	≥ 1822 psia ⁽¹⁾
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
III. CONTAINMENT SPRAY (CSAS)		
A. Sensor/Trip Units		
Containment Pressure High - High	≤ 8.5 psig	≤ 8.9 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
IV. MAIN STEAM LINE ISOLATION (MSIS)		
A. Sensor/Trip Units		
1. Steam Generator Pressure - Low	≥ 919 psia ⁽³⁾	≥ 912 psia ⁽³⁾
2. Steam Generator Level - High	$\leq 91.0\%$ NR ⁽²⁾	$\leq 91.5\%$ NR ⁽²⁾
3. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>ESFA SYSTEM FUNCTIONAL UNIT</u>	<u>TRIP VALUES</u>	<u>ALLOWABLE VALUES</u>
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	7.4% of Span	$7.9 \geq \% \text{ of Span} \geq 6.9$
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	$\geq 25.8\% \text{ WR}^{(4)}$	$\geq 25.3\% \text{ WR}^{(4)}$
2. Steam Generator Δ Pressure - SG2 > SG1	$\leq 185 \text{ psid}$	$\leq 192 \text{ psid}$
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)		
A. Sensor/Trip Units		
1. Steam Generator #2 Level - Low	$\geq 25.8\% \text{ WR}^{(4)}$	$\geq 25.3\% \text{ WR}^{(4)}$
2. Steam Generator Δ Pressure - SG1 > SG2	$\leq 185 \text{ psid}$	$\leq 192 \text{ psid}$
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VIII. LOSS OF POWER		
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	$\geq 3250 \text{ volts}$	$\geq 3250 \text{ volts}$
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2930 to 3744 volts with a 35-second maximum time delay	2930 to 3744 volts with a 35-second maximum time delay
IX. CONTROL ROOM ESSENTIAL FILTRATION	$\leq 2 \times 10^{-5} \mu\text{Ci/cc}$	$\leq 2 \times 10^{-5} \mu\text{Ci/cc}$

3/4.10 SPECIAL TEST E. OPTIONS

3/4.10.1 SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN and K_{N-1} requirements of Specification 3.1.1.2 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2, 3* and 4*#.

ACTION:

- a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN and K_{N-1} required by Specification 3.1.1.2 are restored.
- b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 or MODE 4, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

- a. Reactor-Coolant System boron concentration,
- b. CEA position,
- c. Reactor Coolant System average temperature,
- d. Fuel burnup based on gross thermal energy generation,
- e. Xenon concentration, and
- f. Samarium concentration.

* Operation in MODE 3 and MODE 4 shall be limited to 6 consecutive hours.

Limited to low power PHYSICS TESTING at the 320°F plateau.

SPECIAL TEST EXCEPTIONS

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

- a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and
- b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

- a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or
- b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

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 noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup 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.
- c. Both reactor coolant loops and at least one reactor coolant pump in each loop are in operation.

APPLICABILITY: During startup PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER or with less than the above required reactor coolant loops in operation and circulating reactor coolant, 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 PHYSICS TESTS.

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

4.10.3.3 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

SPECIAL TEST EXCEPTIONS

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6, 3.1.3.7, and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided the limits of Specification 3.2.1 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 Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.6, 3.1.3.7, and 3.2.6 are suspended, either:

- a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, 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, 3.1.3.6, 3.1.3.7, and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6, 3.1.3.7, and/or 3.2.6 are suspended.

REACTIVITY CONTROL SYSTEMS

BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

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.

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 LCOs are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{cold} greater than or equal to 552°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.

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. Specifically, a programmed insertion schedule will be used to cycle the CEAs between the full out position ("FULL OUT" LIMIT) and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing any effects. To accommodate this programmed insertion schedule, the fully withdrawn position was redefined, in some cases, to be 144.75 inches or greater.

The establishment of LSSS and LCOs requires that the expected long- and short-term behavior of the radial peaking factors be determined. The long-term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short-term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions

REACTIVITY CONTROL SYSTEMS

BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specifications 3.1.3.6 and 3.1.3.7 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specifications 3.1.3.6 and 3.1.3.7 the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The PVNGS CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPCs will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) and CEA misoperation transients. When the COLSS is Out of Service (COOS), the monitoring function is performed via the CPC calculation of DNBR in conjunction with Technical Specification COOS Limit Lines (Figures 3.2-2 and 3.2-2A) which restricts the reactor power sufficiently to preserve the ROM.

The reduction of the CEA deviation penalties in accordance with the CEAC (Control Element Assembly Calculator) sensitivity reduction program has been performed. This task involved setting many of the inward single CEA deviation penalty factors to 1.0. An inward CEA deviation event in effect would not be accompanied by the application of the CEA deviation penalty in either the CPC DNBR and LHR (Linear Heat Rate) calculations for those CEAs with the reduced penalty factors. The protection for an inward CEA deviation event is thus accounted for separately.

REACTIVITY CONTROL SYSTEMS

BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

If an inward CEA deviation event occurs, the current CPC algorithm applies two penalty factors to each of the DNB and LHR calculations. The first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA drop. The expected margin degradation for the inward CEA deviation event for which the penalty factor has been reduced is accounted for in two ways. The ROPM reserved in COLSS is used to account for some of the margin degradation. Further, a power reduction in accordance with the curve in Figure 3.1-2A is required. In addition, the part length CEA maneuvering is restricted in accordance with Figure 3.1-5 to justify reduction of the PLR deviation penalty factors.

The technical specification permits plant operation if both CEACs are considered inoperable for safety purposes after this period.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 13.5 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, 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 linear-heat rate limit can be maintained by utilizing any operable CPC channel. The above listed uncertainty and penalty factors plus those associated with the CPC startup test acceptance criteria are also included in the CPCs.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F_{xy}^c) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^m) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. 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. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. 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. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

The AZIMUTHAL POWER TILT is equal to $(P_{\text{tilt}}/P_{\text{untilt}})-1.0$ where:

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{\text{tilt}}/P_{\text{untilt}} = 1 + T_q g \cos (\theta - \theta_0)$$

where:

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_0 is the azimuthal core location of maximum tilt

POWER DISTRIBUTION LIMITS

BASES

AZIMUTHAL POWER TILT - T_q (Continued)

$P_{\text{tilt}}/P_{\text{untilt}}$ 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.

The AZIMUTHAL POWER TILT allowance used in the CPCs is defined as the value of CPC addressable constant TR-1.0.

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. 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. The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum 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 heat flux, 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 Figures 3.2-2 and 3.2-2A 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 and penalty factors are also included in the CPCs 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 less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculations 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. In design calculations, 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.

POWER DISTRIBUTION LIMITS

BASES

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 safety analyses. The minimum value used in the safety analyses is 95% of the design flow rate (164.0×10^6 lbm/hr) or 155.8×10^6 lbm/hr. The actual RCS flow rate is determined by direct measurement and an uncertainty associated with that measurement is considered when comparing actual RCS flow rate to the minimum required value of 155.8×10^6 lbm/hr.

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.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of the core average 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.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.

INSTRUMENTATION

BASES

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time).

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

INSTRUMENTATION

BASES

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that:
(1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR. The seismic instrumentation for the site is listed in Table 3.3-7.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972. Wind speeds less than 0.6 MPH cannot be measured by the meteorological instrumentation.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the remote shutdown system ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

INSTRUMENTATION

BASES

REMOTE SHUTDOWN SYSTEM (Continued)

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control and power circuits and disconnect switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

The alternate disconnect methods or power or control circuits ensure that sufficient capability is available to permit shutdown and maintenance of cold shutdown of the facility by relying on additional operator actions at local control stations rather than at the RSP.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.4.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heat Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existence of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

INSTRUMENTATION

BASES

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION (Continued)

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If both channels are inoperable, the channels shall be restored to OPERABLE status in the nearest refueling outage. If only one channel is inoperable, it is intended that this channel be restored to OPERABLE status in a refueling outage as soon as reasonably possible.

3/4.3.3.7 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

There are two separate radioactive gaseous effluent monitoring systems: the low range effluent monitors for normal plant radioactive gaseous effluents and the high range effluent monitors for post-accident plant radioactive gaseous effluents. The low range monitors operate at all times until the concentration of radioactivity in the effluent becomes too high during post-accident conditions. The high range monitors only operate when the concentration of radioactivity in the effluent is above the setpoint in the low range monitors.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-51
ARIZONA PUBLIC SERVICE COMPANY, ET AL.
PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2
DOCKET NO. STN 50-529

1.0 INTRODUCTION

By letter dated December 3, 1987 (Ref. 1), the Arizona Public Service Company (APS) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), requested several changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-51) for the Palo Verde Nuclear Generating Station, Unit 2 (PVNGS2), relating to Cycle 2 operation for PVNGS2. In support of both the Technical Specification changes and Cycle 2 operation, the licensees submitted a Reload Analysis Report by letter dated December 2, 1987 (Ref. 2). By letter dated February 4, 1988 (Ref. 3), the licensees also provided clarifying information on the Reload Analysis Report, in response to staff questions to verify that all events affected by changes in the Moderator Temperature Coefficient were analyzed. The staff's evaluation of the reload analysis is presented in Sections 2.0 through 5.0 below. The evaluation of the specific changes to the Technical Specifications is presented in Section 6.0 below.

2.0 EVALUATION OF FUEL DESIGN

2.1 Mechanical Design

The Cycle 2 core consists of 241 fuel assemblies. One hundred and eight fresh (unirradiated) Batch D assemblies will replace 69 Batch A assemblies and 39 Batch B assemblies. The remaining 69 Batch B assemblies and all Batch C assemblies in the core during Cycle 1 will be retained.

The 108 Batch D assemblies will consist of 32 type D0 assemblies with 4.02 weight percent (w/o) and 3.57 w/o U-235 enriched fuel rods, 20 type D1 assemblies with 4.02 w/o and 3.57 w/o U-235 enriched rods and 16 burnable poison shims per assembly, 8 type D2 assemblies with 4.02

w/o and 3.57 w/o U-235 enriched rods and 16 burnable poison shims per assembly, 16 type D3 assemblies with 3.57 w/o and 3.09 w/o U-235 enriched rods and 16 burnable poison shims per assembly, four type D4 assemblies with 3.57 w/o and 3.09 w/o U-235 enriched rods and 12 burnable poison shims per assembly, and 28 type D5 assemblies with 3.57 w/o and 3.09 w/o U-235 enriched rods and 12 burnable poison shims per assembly. The mechanical design of the Batch D assemblies is identical to that of the Batch C assemblies used in Cycle 1 except for design features which were incorporated to improve fuel handling, the fabricability and quality of the fuel and the burnup capability of the poison rods. The staff, therefore, finds these modifications to be acceptable.

Attachment 5 to Reference 5 is a report entitled, "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," on work performed by Combustion Engineering (CE) for the Electric Power Research Institute (EPRI). The report presents the results from a review of interpellet gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods. The report concludes that modern CE fuel rods have a time to clad collapse in excess of any practical core residence time. The staff concurs with the conclusions of the CE report as it applies to PVNGS2 Cycle 2. This concurrence is supported by previous acceptance for PVNGS1 Cycle 2 and Waterford 3 Cycle 2 and by similar results of analyses by another fuel vendor. Therefore, the staff concludes that no further analysis of clad collapse need be performed for PVNGS2 Cycle 2.

2.2 Thermal Design

The thermal performance of Cycle 2 fuel was performed by analyzing a composite fuel pin that envelopes the peak pins of the various fuel assemblies (fuel Batches B, C, and D) in the Cycle 2 core using the NRC approved fuel performance code FATES3A. The NRC imposed grain size restriction (Ref. 13) was included and a power history was used that envelopes the power and burnup levels representative of the peak pin at each burnup interval from beginning-of-cycle (BOC) to end-of-cycle (EOC). The maximum peak pin burnup analyzed for Cycle 2 bounds the expected EOC maximum fuel rod burnup. Based on this analysis, the internal pressure in the most limiting hot rod will not reach the nominal reactor coolant system (RCS) pressure of 2250 psia. Since this satisfies the fuel rod internal gas pressure requirement of Standard Review Plan (SRP) 4.2, Section II.A.1(f), the staff finds it acceptable and concludes that the fuel rod internal pressure limits have been adequately considered for Cycle 2 operation.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The PVNGS2 Cycle 2 core consists of 241 fuel assemblies, each having a 16 by 16 fuel rod array. A general description of the core loading is given above in Section 2.1. The highest U-235 enrichment occurs in the Batch D fuel assemblies which contain fuel rods with 4.02 weight percent U-235. The PVNGS2 fuel storage facilities have been approved for storage of fuel with a maximum U-235 enrichment of 4.05 weight percent (Ref. 4).

The Cycle 2 core will use a low-leakage fuel management scheme in which the previously irradiated Batch B assemblies are placed on the core periphery. Most of the fresh Batch D assemblies are placed in the interior of the core and mixed with the previously irradiated fuel to minimize power peaking. With this loading and a Cycle 1 endpoint of 16,512 MWD/MTU, the Cycle 2 reactivity lifetime for full power operation is expected to be 16,945 MWD/MTU. The analyses presented by the licensees will accommodate a Cycle 2 burnup up to 17,500 MWD/MTU and is applicable for Cycle 1 termination burnups of between 15,744 and 17,280 MWD/MTU.

3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power densities are given in the reload analysis report for BOC, middle-of-cycle (MOC), and EOC unrodded configurations. Radial power distributions at BOC and EOC are also given for three rodded configurations allowed by the power dependent insertion limit (PDIL) at full power. These rodded configurations consist of part length CEAs, Bank 5, and Bank 5 plus the part length CEAs.

These expected values are based on ROCS code calculations with neutron cross sections generated by the DIT code (Ref. 6). Also, the use of ROCS and DIT with the MC fine-mesh module explicitly accounts for the higher power peaking which is characteristic of fuel rods adjacent to water holes. These methods have been approved by the NRC and, therefore, the calculated power distributions are acceptable.

3.3 Control Requirements

The value of the required shutdown margin varies throughout core life with the most restrictive value occurring at EOC hot zero power (HZP) conditions. This minimum shutdown margin of 6.5% delta k/k is required to control the reactivity transient resulting from the RCS cooldown associated with a steam line break accident at these conditions. For operating temperatures below 350° F, the reactivity transients resulting from any postulated accident are minimal and a 3.5% delta k/k shutdown margin provides adequate protection. Sufficient boration capability and net available CEA worth exist,

assuming a minimum worth stuck CEA and using appropriate calculational uncertainties, to meet these shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

3.4 Augmentation Factors

CE submitted a report (Ref. 5) which gave the results of a review of interpellet gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods (non-densifying fuel in prepressurized tubes). The report concluded that since the increased power peaking associated with the small interpellet gaps found in these rods is insignificant compared to other power distribution uncertainties used in the safety analyses, augmentation factors can be removed from the reload of any reactor loaded exclusively with this type of fuel. The staff accepted this conclusion for the Cycle 2 reload review of PVNGS1, the Cycle 8 reload review of Calvert Cliffs Unit 1, the Cycle 3 review of SONGS-2, and Cycle 2 review of Waterford 3. The staff finds that the conclusion is also valid for PVNGS2 Cycle 2 since the same manufacturing process is used in the fuel fabrication. The densification augmentation factors can, therefore, be eliminated for PVNGS2 Cycle 2.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 2 is performed using the approved thermal-hydraulic code TORC (Ref. 7) and the CE-1 critical heat flux (CHF) correlation (Ref. 8). The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D (Ref. 9). The CETOP-D model has been verified to predict minimum departure from nucleate boiling ratio (DNBR) conservatively relative to TORC.

The uncertainties associated with the system parameters are combined statistically using the modified statistical combination of uncertainties (SCU) methodology described in Reference 10. This SCU methodology was evaluated and approved in the safety evaluation issued with Amendment No. 24 to Facility Operating License No. NPF-41 for PVNGS1, dated October 21, 1987. Using this methodology, the engineering hot channel factors for heat flux, heat input, fuel rod pitch, and cladding diameter are combined statistically with other uncertainty factors to arrive at overall uncertainty penalty factors to be applied to the DNBR calculations performed by the core protection calculators (CPCs) and the core operating limit supervisory system (COLSS). When used with the Cycle 2 DNBR limit of 1.24, these overall uncertainty penalty factors provide assurance with a 95% confidence and a 95% probability (95/95 confidence/probability) that the hottest fuel rod will not experience DNB. The fuel rod bow penalty is incorporated directly in the DNBR limit. It has been calculated using the approved method described in Reference 11. The value used for this analysis, 1.75% DNBR, is valid for fuel assembly burnups up to 30,000 MWD/MTU. For those assemblies with average burnup in excess of 30,000 MWD/MTU, sufficient margin exists to offset rod bow penalties.

5.0 EVALUATION OF SAFETY ANALYSES

The design basis events (DBEs) considered in the safety analyses are categorized into two groups: anticipated operational occurrences (AOOs) and postulated accidents (limiting faults). All events were reviewed by the licensees to assess the need for reanalysis as a result of the new core configuration for Cycle 2. Those events for which results were not bounded by the FSAR were reanalyzed by the licensees to assure that the applicable criteria are met. The AOOs were analyzed to assure that specified acceptable fuel design limits (SAFDLs) on DNBR and fuel centerline to melt (CTM) temperature are not exceeded. This assurance may require either reactor protection system (RPS) trips or RPS trips and/or sufficient initial steady state margin.

Unless otherwise stated, the plant response to the DBEs was simulated using the same methods and computer programs which were used and approved for the reference cycle analyses. These include the CESEC III, STRIKIN II, TORC and HERMITE computer programs. For some of the reanalyzed DBEs, certain initial core parameters, such as CEA trip worth and axial shape index (ASI), were assumed to be more limiting than the calculated Cycle 2 values in order to bound future cycles. All of the events reanalyzed have results which are within NRC acceptance criteria and, therefore, are acceptable. These are discussed below.

5.1 Steam System Piping Failures Inside and Outside Containment

Steam line breaks (SLBs) inside containment may cause environmental degradation of sensor input to the core protection calculators (CPCs) and pressure measurement systems. Therefore, the only credit taken for CPC action during this event is the CPC variable overpower trip (VOPT). The required input to the VOPT includes output from the resistance temperature detectors (RTDs) and the excore neutron flux detectors. These sensors have been qualified in degraded environmental conditions for a sufficient length of time to allow their use in providing input for VOPT action for this event. The outside containment SLBs, however, are not subject to the same environmental effects on the RPS as the inside containment breaks and the full array of RPS trips, including the CPC low DNBR trip, can be credited. By crediting these RPS trips, the results of both the inside and outside containment SLB events in terms of fuel pin failure caused by the pre-trip power excursion are bounded by the reference cycle.

The hot zero power SLB post-trip return to power was also reanalyzed because of the more adverse moderator cooldown reactivity insertion curve. The effect of this more adverse reactivity insertion was accommodated for in Cycle 2 by increasing the shutdown margin required by the Technical Specifications at zero power from 6.0% delta k/k to 6.5% delta k/k. With this more restrictive requirement, the results of the SLB event initiated from zero power conditions is bounded by the reference cycle analysis. The results of the SLB event initiated

from full power conditions are also bounded by the corresponding reference cycle analysis. Therefore, the staff concludes that Cycle 2 operation is acceptable with respect to accidents resulting in breaks in the steam line.

5.2 CEA Drop Event

The single full length CEA drop event was reanalyzed to determine the initial thermal margin that must be maintained by the LCOs such that the SAFDLs will not be violated. Since the CEA position related penalty factors for downward single CEA deviations of the 4-fingered CEAs have been set equal to unity (no penalty) as part of the CPC improvement program, a reactor trip is not generated for a single 4-fingered CEA drop and, therefore, the expected margin degradation for the event is accounted for by reserving sufficient margin in the LCOs. Although this applies to part length CEAs also, only the single full length CEA drop is analyzed because it requires the maximum initial margin to be maintained by the LCOs. For 12-fingered CEA drops and CEA subgroup drops, the CEA position related penalty factors for downward deviations are still used by the CPCs, as in Cycle 1, to provide a reactor trip when necessary.

The event was initiated by dropping a full length CEA over a period of one second. The turbine load was not reduced, resulting in a power mismatch between the primary and secondary systems, which leads to a cooldown of the RCS. The largest change in power peaking was obtained by evaluating drops involving different individual CEAs into the radial rodged configurations allowed by the power dependent transient insertion limits in Figures 3.1-3 and 3.1-4 of the Technical Specifications. This resulted in a radial peaking factor increase of 8.5% before the effects of short term xenon redistribution set in.

A minimum DNBR of greater than 1.24 was obtained after 900 seconds, as determined from the 8.5% radial power peaking increase following the CEA drop plus 15 minutes of xenon redistribution at the final coolant conditions, resulting in a maximum peaking factor increase of 11.4%. If the dropped CEA has not been realigned within 10 minutes after the drop, the operator will take action to reduce power in accordance with Figure 3.1-2A of the Technical Specifications. A maximum allowable initial linear heat rate (LHR) of 18.0 kw/ft could exist as an initial condition without causing the acceptable fuel centerline melt limit of 21.0 kw/ft to be exceeded during the transient. This amount of margin is assured since the LHR LCO is based on the more limiting allowable LHR for the loss of coolant accident (LOCA) of 13.5 kw/ft. The staff, therefore, concludes that Cycle 2 meets the requirements of SRP Section 15.4.3 governing control rod misoperation.

5.3 Asymmetric Steam Generator Events

Of the four events which affect a single steam generator, the loss of load to one steam generator (LL/1SG) event is the most limiting. This event is initiated by the inadvertent closure of both main steam isolation valves which results in a loss of load to the affected steam generator. The CPC high differential cold leg temperature trip is the primary means of mitigating this transient with the steam generator low level trip providing an additional means of protection. The calculated minimum transient DNBR was greater than the DNBR SAFDL limit of 1.24. A maximum allowable LHR of 17.0 kw/ft could exist as an initial condition without exceeding the fuel CTM SAFDL of 21.0 kw/ft during the transient. This amount of margin is assured by setting the LHR LCO based on the more limiting allowable LHR for LOCA of 13.5 kw/ft. The staff concludes that the calculations contain sufficient conservatism to assure that fuel damage will not result from any asymmetric steam generator event during Cycle 2 operation.

A methodology change was made from the reference cycle analysis of this event. The change involved the application of the HERMITE computer code to model both the effect of the temperature tilt on radial power distribution and the space-time impact of the CEA scram. HERMITE has been approved for licensing applications (Ref. 12) and uses the core parameters generated by the CESEC III code (core flow, RCS inlet temperature, RCS pressure, and reactor trip time) as input to simulate the core in the two dimensions. The staff finds this improved modeling technique acceptable.

5.4 Loss of Coolant Accident (LOCA)

The emergency core cooling system (ECCS) performance evaluation for both the large and the small break LOCA must show conformance with the acceptance criteria required by 10 CFR 50.46. Since there are no significant changes to the RCS design characteristics compared to the reference cycle, the blowdown hydraulic calculations, refill/reflood hydraulics calculations, and steam cooling heat transfer coefficients of the reference cycle also apply to Cycle 2. Therefore, only fuel rod clad temperature and oxidation calculations were performed for the 1.0 double ended guillotine at pump discharge (DEG/PD) break to evaluate ECCS performance due to the Cycle 2 changes in fuel conditions. This was the limiting break size for the reference cycle and, since the hydraulics are identical, is also the limiting break size for Cycle 2.

The 1.0 DEG/PD limiting large break case resulted in a peak clad temperature (PCT) of 1960°F, a peak local clad oxidation percentage of 5.7%, and a total core wide clad oxidation percentage of less than 0.8%. These results meet the 10 CFR 50.46 acceptance criteria for peak clad temperature (2200°F), peak local clad oxidation percentage (17.0%), and core wide clad oxidation percentage (1.0%). These

results are applicable for up to 400 plugged tubes per steam generator because of the conservatively high pressure drop through the steam generators used in the analyses.

The increase in PCT for a small break LOCA, assuming 400 plugged tubes per steam generator, is much less than 100°F. Therefore, the estimated PCT for Cycle 2 is less than 1730°F and well within the 10 CFR 50.46 limit.

Based on these results, the staff concurs that both large and small break acceptable LOCA ECCS performance has been demonstrated for Cycle 2 at a peak linear heat generation rate of 13.5 kw/ft and a reactor power level of 3876 Mwt (102% of 3800 Mwt) for up to 400 plugged tubes per steam generator.

6.0 EVALUATION OF TECHNICAL SPECIFICATION CHANGES

In support of Cycle 2 operation, the licensees have requested a number of changes to the Technical Specifications. The specific changes and the staff's evaluation are presented below.

- (1) The shutdown margin versus cold leg temperature curve given in Figure 3.1-1A of Technical Specification 3.1.1.2 has been changed to increase the required shutdown margin value from 6.0% delta k/k to 6.5% delta k/k at cold leg temperatures above 500°F.

The increased shutdown margin is required to ensure that the steam line break event at hot zero power, which is the most limiting accident with regard to shutdown margin requirements for Cycle 2, is bounded by the reference cycle (Cycle 1) analysis. Sufficient CEA trip reactivity worth is available to meet the shutdown margin requirements even with the most reactive CEA assumed stuck in the fully withdrawn position. The staff, therefore, finds this change acceptable.

- (2) The moderator temperature coefficient (MTC) operating band given in Figure 3.3-1 of Technical Specification 3.1.1.3 has been made more positive at low power and more negative at high power. In addition, the x axis has been changed to core power level instead of average moderator temperature.

The MTC for Cycle 2 at 100% power has a value of 0.0 at BOC which is the same value that the Cycle 1 MTC had at 100% power and BOC. The BOC zero power value has been increased from $+0.22 \times 10^{-4}$ to $+0.5 \times 10^{-4}$ delta k/k/°F and the EOC full power value has been decreased from -3.0×10^{-4} to -3.5×10^{-4} delta k/k/°F. The licensees have reevaluated the most limiting transients and accidents which can be adversely affected by the increased MTC operating band and found them to be bounded by the reference cycle (Cycle 1) analyses. In addition, by making the MTC a dependent variable of core power only and not of inlet temperature and core power, the calculation of the limiting MTC

need only be performed once. There is no effect on the safety analysis results and the same approved methodology and computer codes are used in the calculations. Therefore, the proposed change is acceptable.

- (3) The operational pressure band of the pressurizer given in Technical Specification 3.2.8 has been changed from 1815 psia through 2370 psia to 2025 psia through 2300 psia.

The potential transients initiated at the extremes of the Cycle 1 pressure range were not analyzed for Cycle 2 and, therefore, normal operation at these extremes cannot be supported by the safety analyses. Therefore, the operation band of the pressurizer was made more restrictive, limiting the field of possible accidents and maintaining the safety margin required by the reference cycle safety analysis or the FSAR safety limits. Therefore, this change is acceptable.

- (4) Reference to the part length CEA insertion limits have been removed from Technical Specification 3.1.3.1 and a new Specification 3.1.3.7 has been added to specify the length of time for insertion and the insertion limit of the part length CEAs specifically.

The new Specification adds a more explicit LCO to clarify the allowable duration for a part length CEA to remain within the defined ranges of axial position and reduces the potential adverse consequences of a part length CEA drop or misalignment from an allowable position. The changes are, therefore, acceptable.

- (5) The response time of the DNBR-low reactor coolant pump (RCP) shaft speed trip in Technical Specification 3.3.1, Table 3.3-2, has been decreased from 0.75 seconds to 0.30 seconds.

The response time has been defined as the time from when a signal is sent down the RCP shaft speed sensor line to the CPCs to the time when the control element drive mechanism coil breakers open. Since the Cycle 2 safety analysis has taken credit for the faster response time, the change to Table 3.3-2 is necessary to ensure that PVNGS2 is operated within the safety analysis. Therefore, it is acceptable.

- (6) The DNBR limit specified in Technical Specification 2.1.1.1, Table 2.2-1 and Bases Sections 2.1.1 and 2.2.1, has been changed from 1.231 to 1.24. Also, the requirement to calculate additional rod bow penalties has been removed from Notation (5) of Table 2.2-1 and the low pressurizer pressure floor has been changed from 1861 psia to 1860 psia.

The modified SCU methodology discussed in Section 4.0 of this evaluation yields a DNBR limit of 1.24. The overall uncertainty factors determined by this modified methodology, which has been approved by the staff, continue to ensure that the COLSS power operating limit calculations and the CPC DNBR and LPD calculations will be conservative to at least a 95% probability and 95% confidence level. The 1.24 DNBR limit is, therefore, acceptable.

The rod bow penalty factor of 1.75% which has been applied to the DNBR limit compensates for the effects of rod bow for fuel assemblies with burnups up to 30,000 MWD/MTU. As discussed in Section 4.0 of this evaluation, sufficient available margin exists in assemblies with burnup greater than 30,000 MWD/MTU to offset any additional rod bow penalties. The deletion of these additional penalties from Table 2.2-1 is, therefore, acceptable.

A reevaluation of the Cycle 2 safety analysis was performed to determine how the low pressurizer pressure floor for the DNBR-low trip would change as a result of the DNBR limit change. Since the results show that a pressurizer pressure of 1860 psia instead of 1861 psia will ensure acceptable consequences in the event of a reactor trip on low-DNBR, the proposed change to the low pressurizer pressure floor is acceptable.

- (7) The CEA insertion limits given in Technical Specification 3.1.3.6 have been revised to account for changes in the reactivity worth of the CEAs due to changes in the Cycle 2 core.

Since the reactivity worth of the CEAs has changed, the consequences of the dropped and ejected CEA events are affected. The revised CEA insertion limits chosen, which were calculated by approved methods, ensure that there is sufficient margin to mitigate such events during Cycle 2. The CEA insertion limit revision is, therefore, acceptable.

- (8) The CPC penalty factors, which have been used to compensate for resistance temperature detector (RTD) response times greater than 8 seconds (but less than or equal to 13 seconds) have been removed from the Technical Specifications by modifying Table 3.3-2 and removing Table 3.3-2a.

The Cycle 2 safety analyses assume a maximum RTD response time of 8 seconds and do not include an allowance to enter CPC penalty factors to compensate for RTD response times greater than 8 seconds. Hence, the removal of the penalty factor allowances is required in order to ensure that the Cycle 2 safety analyses assumptions are met during Cycle 2 operation. Therefore, the change is acceptable.

- (9) The RCS total flow rate specified in Technical Specification 3.2.5 has been reduced from greater than or equal to 164.0×10^6 lbm/hr to greater than or equal to 155.8×10^6 lbm/hr.

This change ensures that the actual total RCS flow rate is maintained at or above the minimum value used in the Cycle 2 safety analysis and is, therefore, acceptable.

- (10) The LHR limit defined in Technical Specification 3.2.1 has been decreased from 14.0 kw/ft to 13.5 kw/ft. In addition, the amendment also delineates how LHR is to be monitored and changes the format of the Action statement.

As stated in Section 3.4, augmentation factors previously used to compensate for increased power peaking due to fuel densification were not used for the Cycle 2 safety analyses. The elimination of these augmentation factors as well as the increased fuel enrichment and different core loading pattern for Cycle 2, result in a change in the allowable LHR limit. Since the Cycle 2 safety analyses show that in the event of a LOCA, the peak fuel clad temperature will not exceed 2200°F, the decreased LHR is acceptable.

The change which delineates how LHR is to be monitored is also acceptable since, by providing more detail on the monitoring of LHR, there is added assurance that the LHR will be maintained below the specified limit.

The change in the format of the Action statement facilitates assessment of the actions required if the LHR limit should be exceeded and is, therefore, acceptable.

- (11) Technical Specification 3.2.4 has been replaced with a new format which (a) addresses the specific conditions for monitoring DNBR with or without COLSS and/or the CEACs, (b) delineates what Actions should be taken, (c) removes reference to the DNBR penalty factor table used in Technical Specification 4.2.4.4, and (d) replaces Figures 3.2-1 and 3.2-2 with new Figures 3.2-1, 3.2-2, and 3.2-2A. In addition, as a result of being incorporated into the new Technical Specification 3.2.4, references to operation with both CEACs inoperable and the graph of DNBR margin operating limit (Figure 3.3-1) have been removed from Technical Specification 3.3.1. Bases Sections 3/4.1.3 and 3/4.2.4 have also been modified due to Cycle 2 differences and the above mentioned changes.

These changes ensure operation of the reactor within the approved Cycle 2 safety analysis by modifying the figures, increase operator reliability by placing DNBR operating limits in one place, and eliminate superfluous information. The changes are, therefore, acceptable.

- (12) The refueling actuation signal trip value of the refueling water storage tank, given in Table 3.3-4 of Technical Specification 3.3.2, has been changed from $\geq 7.4\%$ to 7.4% (the midpoint) of the allowable operational values.

The change is more restrictive since it maintains the trip value at the midpoint of the allowable band and reduces the allowable trip values to a single value which was a part of the safety analysis. Therefore, this is acceptable.

- (13) A number of administrative changes have been made to Bases Sections 2.2.1, 3/4.3.1, and 3/4.3.2.

These changes were made to ensure clarity and conciseness to include updated references and to remove Cycle 1 specific information no longer needed for Cycle 2. The changes are, therefore, acceptable.

7.0 STARTUP TESTING

The licensees have presented a brief description of the low power physics tests and the power ascension testing to be performed during Cycle 2 startup. The described tests will verify that core performance is consistent with the engineering design and safety analyses. If the acceptance criteria of any of the startup physics tests are not met, an evaluation will be performed by the licensees, presented to the Plant Review Board and resolved by the Board prior to subsequent power escalation. If an unreviewed safety question is involved, the NRC will be notified.

The staff has reviewed the proposed startup test program for Cycle 2 and finds that it conforms to accepted practices and adequately supplements normal surveillance tests which are required by the plant Technical Specifications.

8.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, and thermal-hydraulics information presented in the PVNGS2 Cycle 2 reload report. The staff has also reviewed the proposed Technical Specification revisions, the startup test procedures, and the safety reanalyses. Based on the evaluations given in the preceding sections, the staff finds the proposed reload and the Technical Specification changes to be acceptable.

9.0 CONTACT WITH STATE OFFICIAL

The Arizona Radiation Regulatory Agency was advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

10.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR 20 relating to a reactor refueling. The staff has determined that this amendment involve no significant increase in the amount, and no significant change in the type, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazard consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

11.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

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Dated: May 5, 1988

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3. "Palo Verde Nuclear Generating Station (PVNGS) Unit 2 Cycle 2 Reload Questions," submitted by letter from E. E. Van Brunt, Jr. (ANPP), dated February 4, 1988.
4. Amendment No. 18 to Facility Operating License No. NPF-51, dated March 9, 1988.
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6. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, Combustion Engineering, April 1983.
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9. "CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Station, Units 2 and 3," CEN-160(S)-P, Revision 1-P, Combustion Engineering, September 1981.
10. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P, Revision 01-P, Combustion Engineering, July 1987.
11. "Fuel and Poison Rod Bowing," CENPD-225-P-A, Combustion Engineering, June 1983.
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13. "Safety Evaluation of CEN-161 (FATES3)," submitted by letter from R. A. Clark (NRC), to A. E. Lundvall, Jr. (BG&E), March 31, 1983.