October 21, 1987

Docket No.: STN 50-528

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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:

SUBJECT: ISSUANCE OF AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. NPF-41 FOR THE PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1 (TAC NOS. 65460, 65461, 65462 AND 65691 THROUGH 65706)

The Commission has issued the subject Amendment, which is enclosed, to the Facility Operating License for Palo Verde Nuclear Generating Station, Unit 1. The Amendment consists of changes to the Technical Specifications (Appendix A to the license) in response to your application transmitted by letter dated June 29, 1987, as supplemented by letters dated June 29, July 13, August 20 (two letters), September 4 and October 1, 1987.

The Amendment revises several portions of the Technical Specifications to incorporate changes in support of Cycle 2 operation for Palo Verde, Unit 1. One of the proposed changes to the Technical Specifications in your amendment request, involving the removal of the numerical values for the axial shape index limits, has not been granted. This request, which is not required for restart of Palo Verde Unit 1, is similar to requests from other licensees and is currently being reviewed on a generic basis by the staff. After the results of the staff's generic review become available, you may resubmit your request consistent with the resultant staff findings.

Page 3/4 3-41 of the Technical Specifications was revised in Amendment No. 21, which was issued September 4, 1987. This page is being reissued at this time, along with its overleaf page (3/4 3-42), in order to correctly represent the approval granted in Amendment No. 21.

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,

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

Enclosures:

1. Amendment No. 24 to NPF-41
2. Safety Evaluation
3. Pages 3/4 3-41 and 3-42

cc: See next page

See previous concurrence

*DRSP/PDV	*DRSP/PDV	*OGC-Beth	DRSP/[illegible]/PDV
EALicitra:cd	JLee		GWKnighton
10/6/87	10/6/87	10/14/87	10/21/87

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

Palo Verde

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. NPF-41

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

   A. The application for amendment, dated June 29, 1987, as supplemented by letters dated June 29, July 13, August 20 (two letters), September 4 and October 1, 1987, 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-41 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 24, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

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

Enclosure:
Changes to the Technical Specifications

Date of Issuance: October 21, 1987

October 21, 1987

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. NPF-41

DOCKET NO. STN 50-528

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	-
3/4 1-23	-
3/4 1-24	-
3/4 1-25	3/4 1-26
3/4 1-31	-
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
3/4 3-7	-
3/4 3-8	-
3/4 3-9	-
3/4 3-10	-
3/4 3-11	-
3/4 3-12	-
3/4 3-13	3/4 3-14
3/4 3-26	3/4 3-25

ENCLOSURE TO LICENSE AMENDMENT CONTINUATION

Amendment Pages	Overleaf Pages
B 3/4 1-6	B 3/4 1-5
B 3/4 1-7	-
B 3/4 2-1	B 3/4 2-2
B 3/4 2-3	-
B 3/4 2-4	-
B 3/4 3-1	-
B 3/4 3-2	-
5-5	5-6

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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.

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

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
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	≥ 44.2% (4)	≥ 43.7% (4)
4. Steam Generator Level - High	≤ 91.0% (9)	≤ 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

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
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.

2.1.1 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

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:

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.

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.

	Parameter	Limiting Value
a.	RCS Cold Leg Temperature-Low	$\geq$ 470°F
b.	RCS Cold Leg Temperature-High	$\leq$ 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	$\geq$ 1860 psia
f.	Pressurizer Pressure-High	$\leq$ 2388 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 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 excesssive 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.

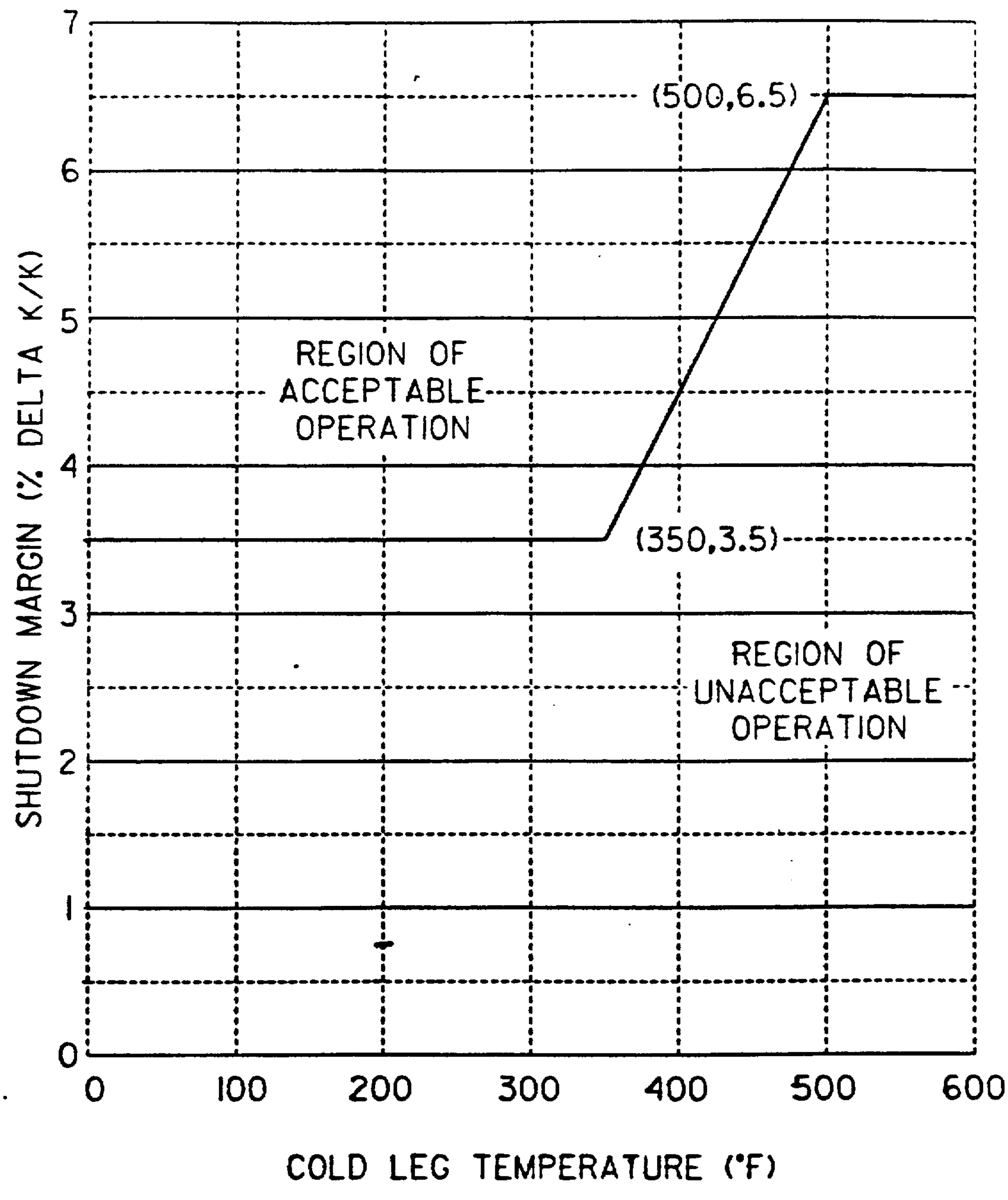


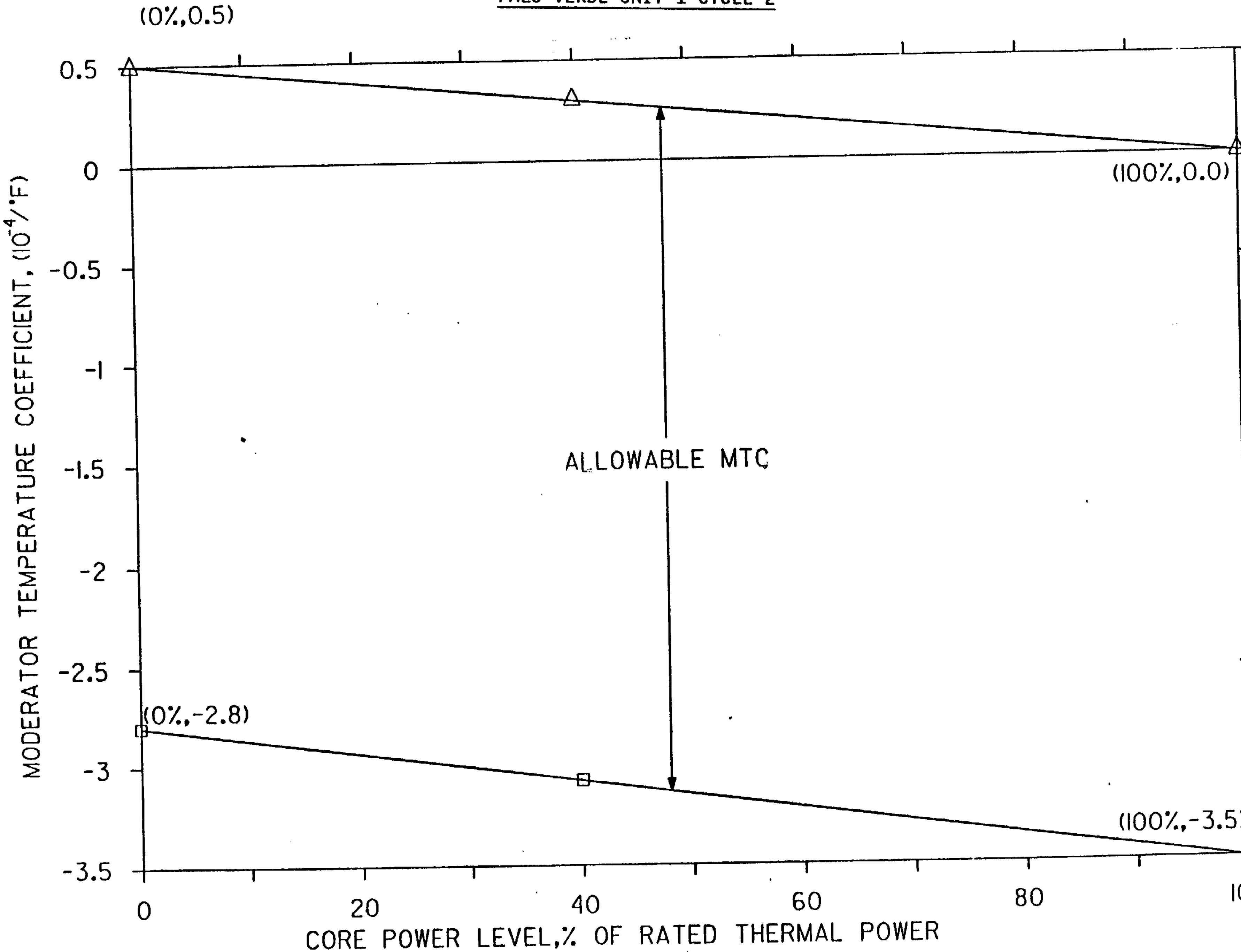
FIGURE 3.1-1A

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.

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 3.1.1.1 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 3.1.1.1 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.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 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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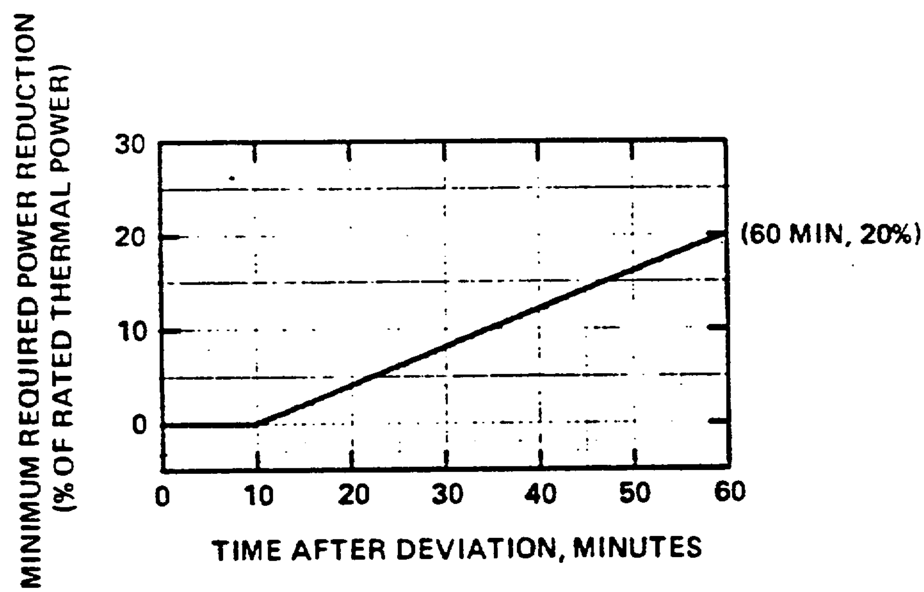

*WHEN CORE POWER IS REDUCED TO 55% OF RATED THERMAL POWER PER THIS LIMIT CURVE, FURTHER REDUCTION IS NOT REQUIRED

FIGURE 3.1-2A

CORE POWER LIMIT AFTER CEA DEVIATION*

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.

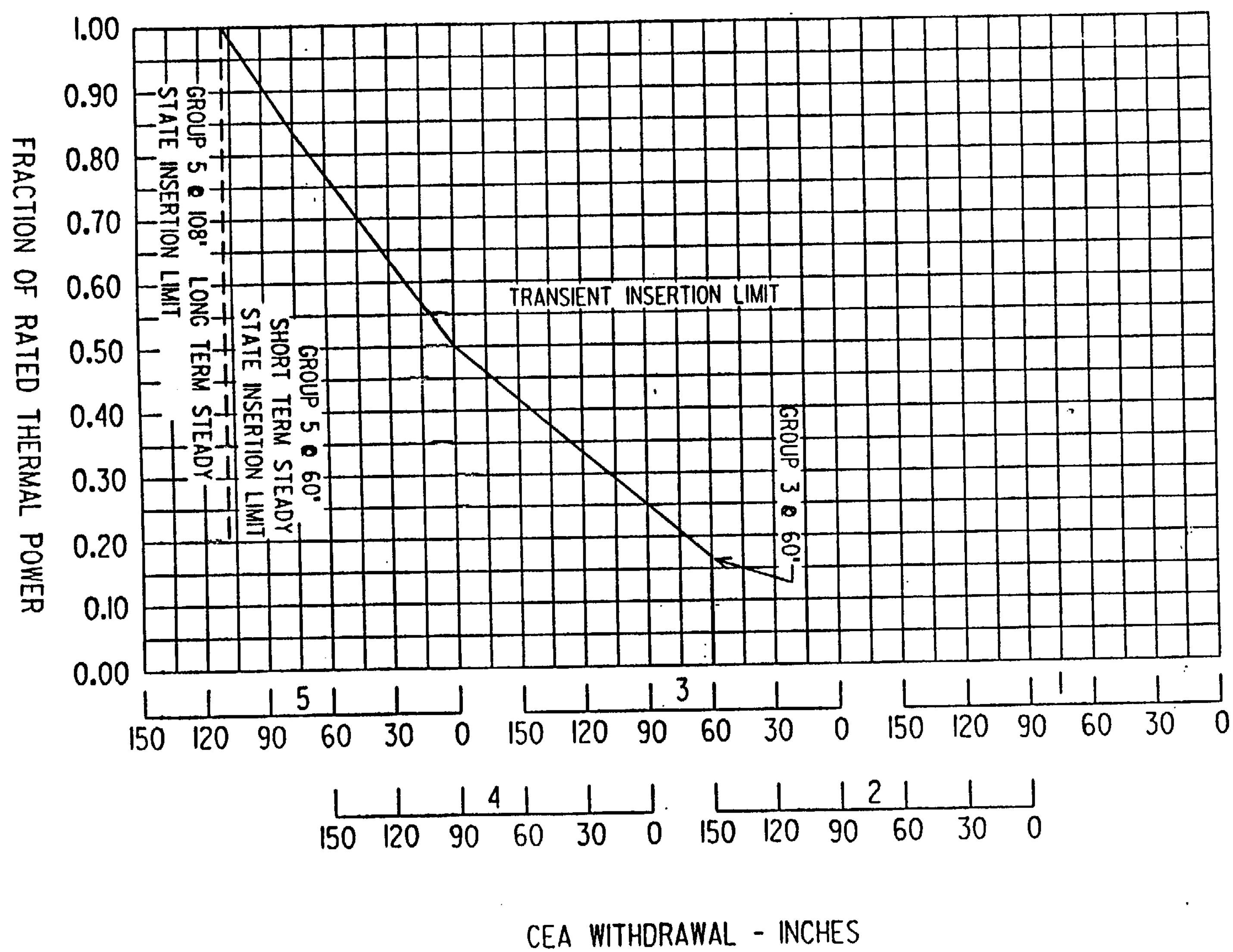

FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS IN SERVICE)

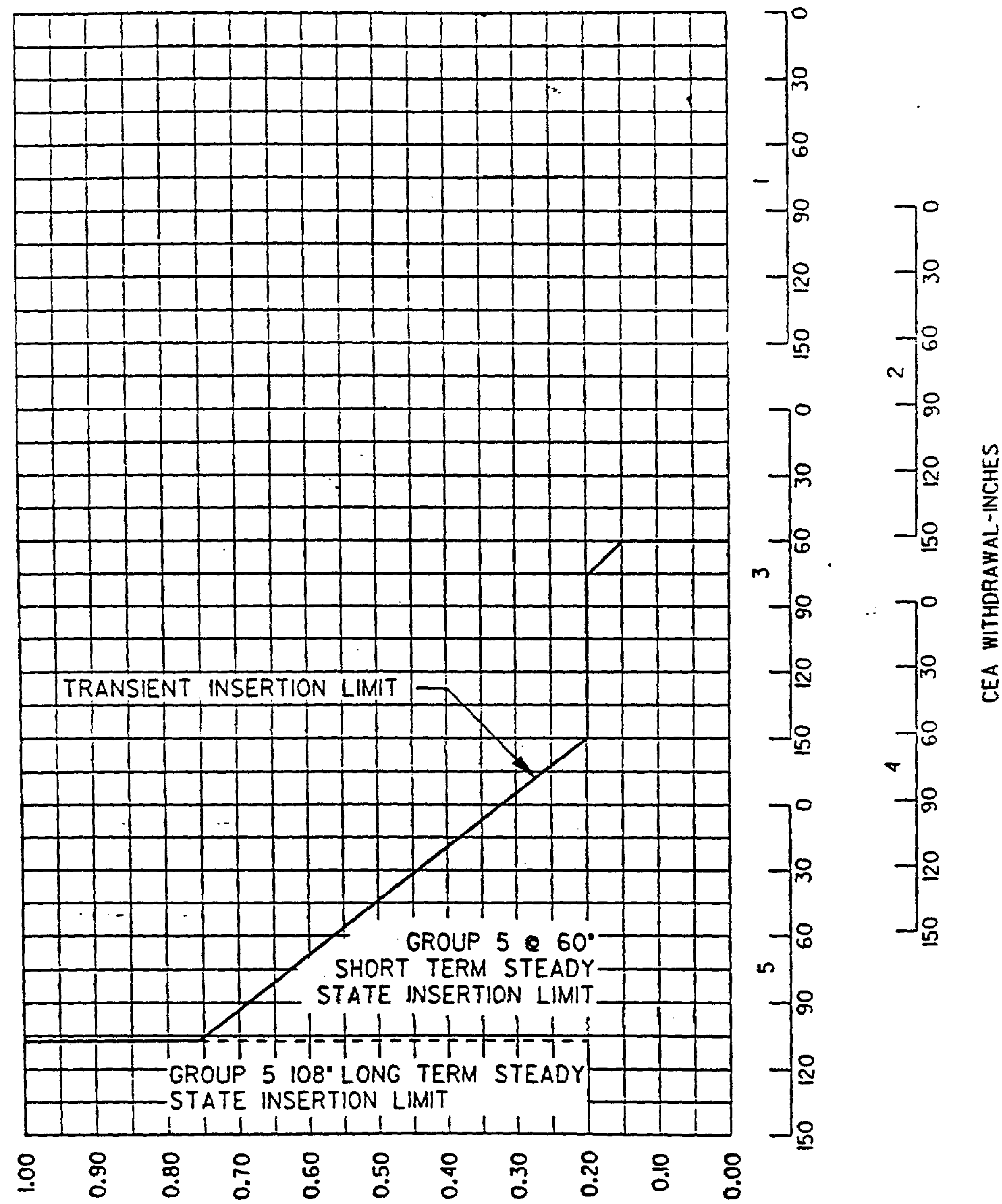

FIGURE 3.1-4

<u>CEA INSERTION LIMITS VS THERMAL POWER</u>
(COLSS OUT OF SERVICE)

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. $\leq$ 7 EFPD per 30 EFPD interval, and

b. $\leq$ 14 EFPD per calender 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.

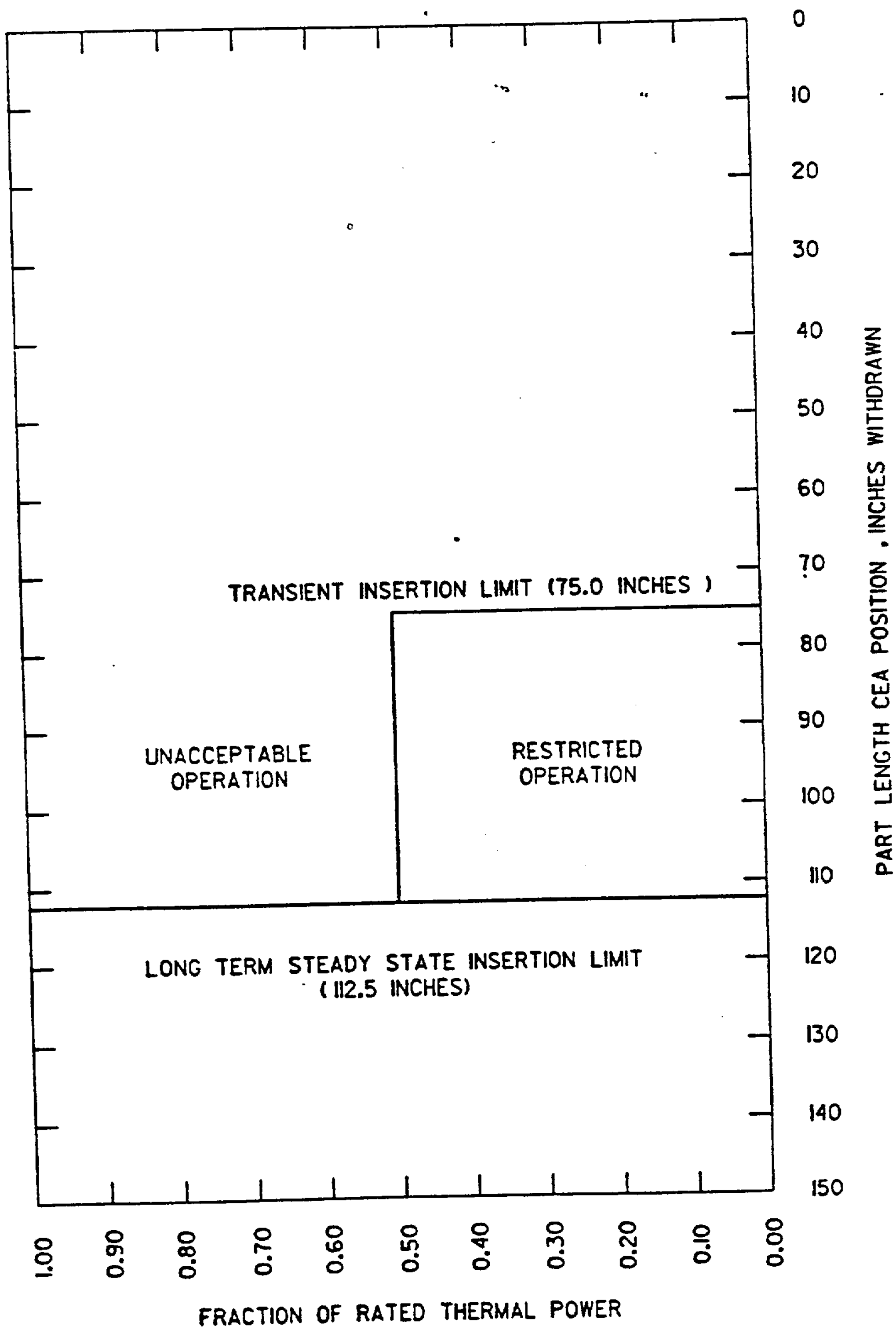

FIGURE 3.1-5

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 neat 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}

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) 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^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. 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^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

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

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 Effective Full Power Days.

*See Special Test Exception 3.10.2.

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 bein 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.

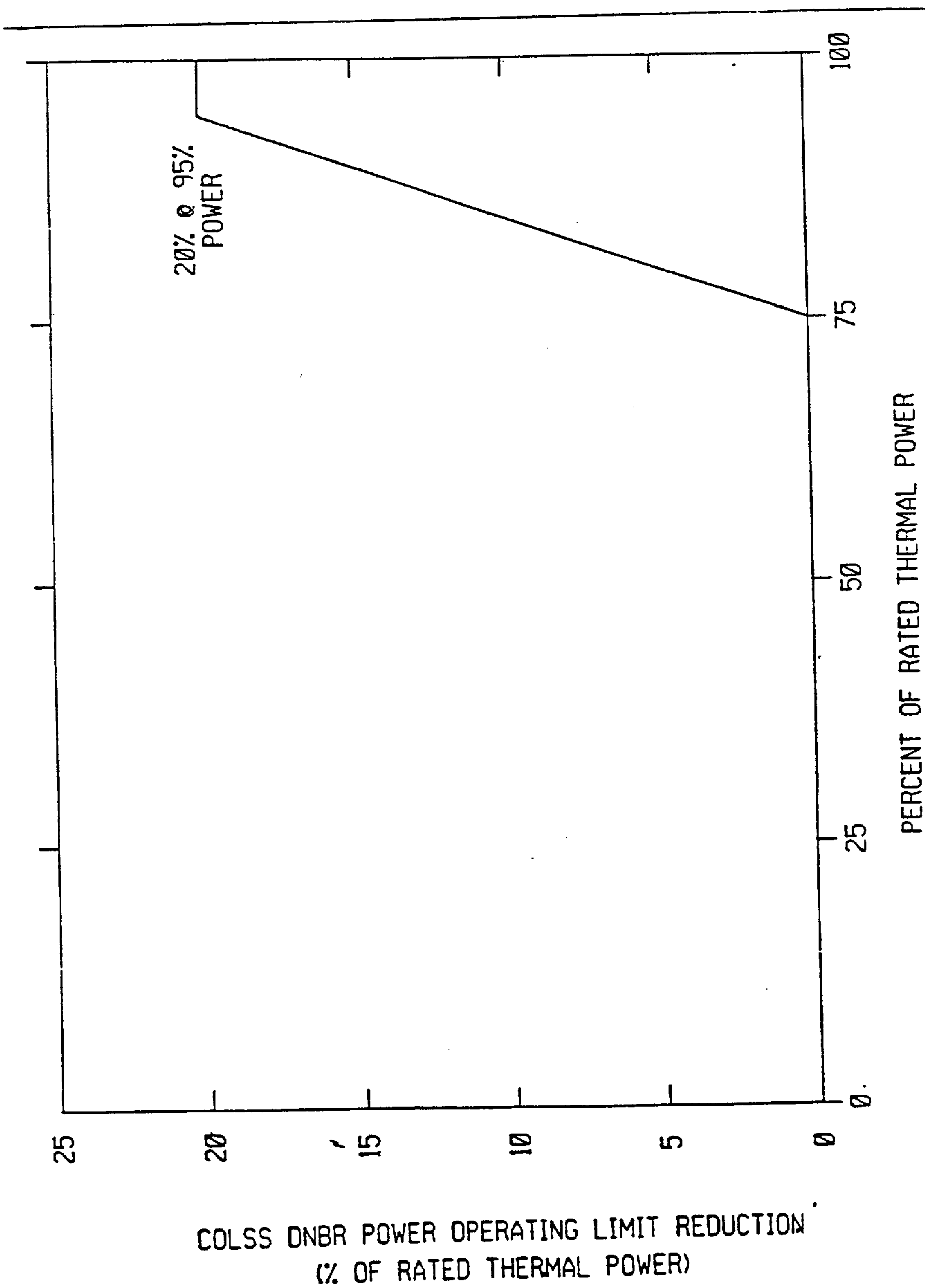

FIGURE 3.2-1

COLSS DNBR POWER OPERATING LIMIT
ALLOWANCE FOR BOTH CEACs INOPERABLE

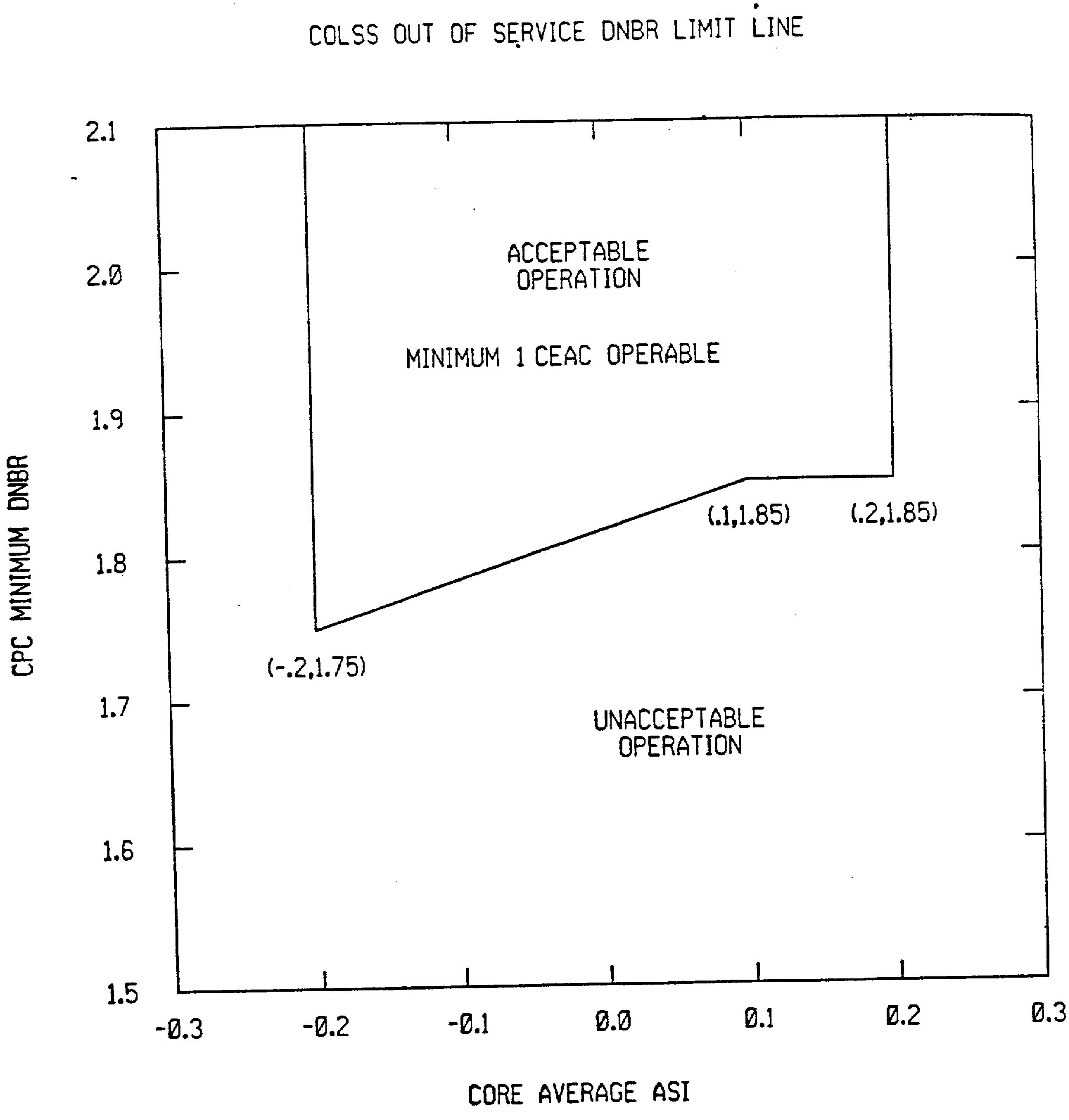

FIGURE 3.2-2

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

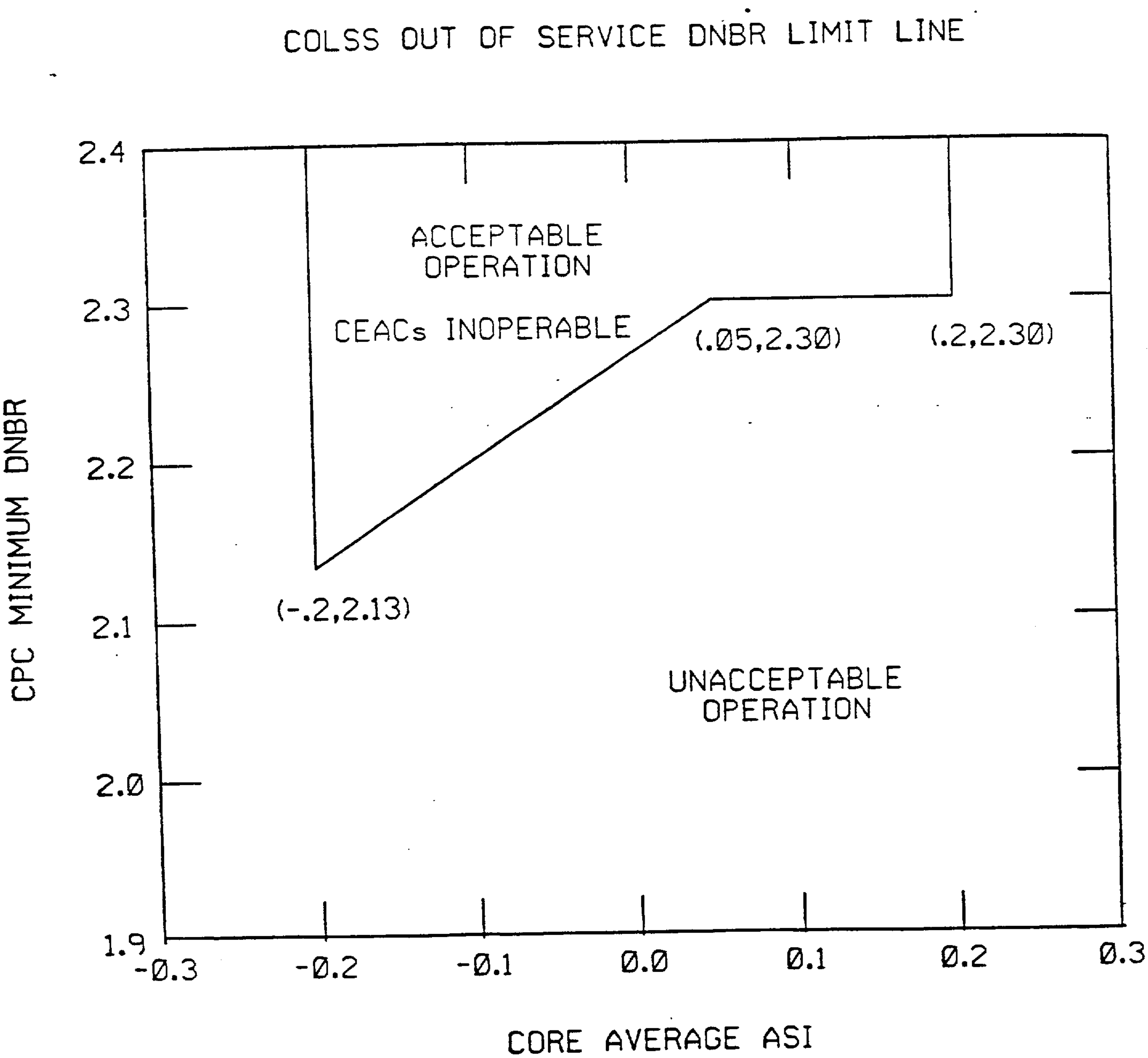

FIGURE 3.2-2A

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

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.

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
$-0.28 \leq ASI \leq 0.28$

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

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.

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)

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)

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 be indicated that both CEAC's 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

FUNCTIONAL UNIT	RESPONSE TIME
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

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	$\leq$ 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

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
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

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	$\leq$ 3.0 psig	$\leq$ 3.2 psig
2. Pressurizer Pressure - Low	$\geq$ 1837 psia(1)	$\geq$ 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	$\leq$ 3.0 psig	$\leq$ 3.2 psig
2. Pressurizer Pressure - Low	$\geq$ 1837 psia(1)	$\geq$ 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	$\leq$ 8.5 psig	$\leq$ 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	$\geq$ 919 psia(3)	$\geq$ 912 psia(3)
2. Steam Generator Level - High	$\leq$ 91.0% NR(2)	$\leq$ 91.5% NR(2)
3. Containment Pressure - High	$\leq$ 3.0 psig	$\leq$ 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

ESFA SYSTEM FUNCTIONAL UNIT	TRIP VALUES	ALLOWABLE VALUES
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	7.4% of Span	7.9 $\geq$ % 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% WR(4)	$\geq$ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG2 > SG1	$\leq$ 185 psid	$\leq$ 192 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% WR(4)	$\geq$ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG1 > SG2	$\leq$ 185 psid	$\leq$ 192 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 volts	$\geq$ 3250 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	$\leq 2 \times 10^{-5}$ μCi/cc	$\leq 2 \times 10^{-5}$ μCi/cc

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

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 and 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 restrict the reactor power sufficiently to preserve the ROPM.

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 DNB 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.

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.

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 $(P_{tilt}/P_{untilt})-1.0$ 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:

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

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

AZIMUTHAL POWER TILT - T_q (Continued)

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

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) are 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.

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.

3/4.3.1 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.

Any modifications which are made to the core protection calculator software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revision 3-P-A and Supplement 1-P, Revision 3-P-A or another NRC approved procedure on CPC software modifications.

CPC modifications which result in a) an unreviewed safety questions, b) a Technical Specification change, or c) methodology not previously approved by the NRC, including additions or deletions to addressable constants or modifications to the approved constant limit values, will require NRC approval prior to implementation.

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.

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

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.

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).

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 241 fuel assemblies with each fuel assembly containing 236 fuel rods or burnable poison rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of approximately 1950 grams uranium. Each burnable poison rod shall have a nominal active poison length of 136 inches. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.05 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 76 full-length and 13 part-length control element assemblies.

5 4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

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

b. For a pressure of 2500 psia, and

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

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,900 + 300/-0 cubic feet at a nominal T_{avg} of 593°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. NPF-41

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. STN 50-528

1.0 INTRODUCTION

By letter dated June 29, 1987 (Ref. 1), as supplemented by letters dated August 20, 1987 (Ref. 4) and October 1, 1987, 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 NPF-41) for the Palo Verde Nuclear Generating Station, Unit 1 (PVNGS1), relating to Cycle 2 operation for PVNGS1. In support of both the Technical Specification changes and Cycle 2 operation, the licensees submitted (1) a Reload Analysis Report by letter dated June 29, 1987 (Ref. 2), as supplemented by letters dated July 13 and August 20, 1987 (Ref. 5 and 6) and September 4, 1987, and (2) a report concerning a modified version for a Statistical Combination of Uncertainties (SCU), dated July 1987 (Ref. 3). The staff's evaluation of the SCU report and the reload analysis is presented in Sections 2.0 through 6.0 below. The evaluation of the specific changes to the Technical Specifications is presented in Section 7.0 below.

2.0 EVALUATION OF FUEL DESIGN

2.1 Mechanical Design

The Cycle 2 core consists of 241 fuel assemblies. Eighty fresh (unirradiated) Batch D assemblies will replace 69 Batch A assemblies and 11 Batch B assemblies. The remaining 97 Batch B assemblies and all Batch C assemblies in the core during Cycle 1 will be retained.

The 80 Batch D assemblies will consist of 36 type DO assemblies with 4.05 weight percent (w/o) and 3.36 w/o U-235 enriched fuel rods, 28 type D* assemblies with 3.36 w/o and 2.78 w/o U-235 enriched rods and eight burnable poison shims per assembly, 12 type DX assemblies with 3.36 w/o and 2.78 w/o U-235 enriched rods and eight burnable poison shims per assembly, and four type D/ assemblies with 3.36 w/o and 2.78 w/o U-235 enriched rods and eight 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 the fabricability and quality of the fuel and the burnup capability of the poison rods. The staff, therefore, finds these modifications to be acceptable.

The licensees have also evaluated the criticality effects of storage of the higher enriched Batch D fuel assemblies in the PVNGS1 fuel storage facilities and have shown that the NRC acceptance criterion of k_{eff} no greater than 0.95 is met for all normal and abnormal conditions. This evaluation is discussed in more detail in Section 3.1 below. The staff, therefore, concludes that the Batch D fuel assemblies are acceptable for use during Cycle 2.

Attachment 5 to Reference 7 is a report entitled, "Evaluation of Inter-pellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," regarding 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 PVNGS1 Cycle 2. This concurrence is supported by similar results of analyses by another fuel vendor and was previously accepted for Waterford 3 Cycle 2. Therefore, the staff concludes that no further analysis of clad collapse need be performed for PVNGS1 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. 12) 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 PVNGS1 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 4.05 weight percent U-235.

The storage facility for new fuel at PVNGS1 provides dry storage for a maximum of 90 fuel assemblies (more than one-third of a core load) and includes the fuel assembly storage racks and the concrete storage vault that contains the storage racks. The fuel assemblies are normally stored in a dry environment in these racks.

The spent fuel storage facility provides underwater storage for a maximum of 1329 fuel assemblies or approximately 5-1/3 full core loads. The facility is designed to store spent fuel assemblies in three different arrays depending on the mode chosen. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells. By properly placing L-inserts and blocking devices in the fuel storage cells, the fuel may be stored in a checkerboard mode for a maximum of 665 fuel assemblies. The blocking devices are placed in appropriate cells in order to prevent the inadvertent placement of a fuel assembly in other than the prescribed spacing. By inserting neutron poison boxes in the rack cells, the fuel may be stored in a high density mode for a maximum of 1329 fuel assemblies. These two storage modes may also be utilized simultaneously within an individual storage rack resulting in a third mixed mode of storage. The spent fuel assemblies are normally stored in borated water.

By letter dated July 13, 1987 (Ref. 5), the licensees submitted a letter from C. Ferguson (CE) to Paul F. Crawley (ANPP), V-CE-34683, dated May 27, 1987 documenting the review of analyses performed to support PVNGS Units 1, 2 and 3 fuel enrichments up to 4.30 weight percent U-235. This letter verifies that the original analyses of the new fuel, spent fuel (except as noted below) and intermediate racks, as well as the fuel elevator, fuel upender and transfer machine were all performed for 4.30 weight percent U-235 fuel. Since the k_{eff} values based on the storage of 4.30 weight percent U-235 fuel meet the NRC acceptance criteria of no greater than 0.95 for fully flooded (unborated) conditions and 0.98 for optimum moderation conditions, the PVNGS fuel storage facilities are acceptable for the storage of 4.05 weight percent U-235 fuel.

One exception is the analysis of the spent fuel racks with neutron poison (boron) boxes in the cells. This high density mode was analyzed for a maximum enrichment of 4.0 weight percent U-235. Therefore, if a future decision is made to use this mode for storage of higher enriched fuel, an analysis will be required for this higher enrichment.

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

17,280 MWD/MTU, the Cycle 2 reactivity lifetime for full power operation is expected to be 13,056 MWD/MTU. The analyses presented by the licensees will accommodate a Cycle 2 burnup up to 13,098 MWD/MTU and are applicable for Cycle 1 termination burnups of between 16,512 and 19,085 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. 8). 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. 7) 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 8 reload review of Calvert Cliffs Unit 1, the Cycle 3 review of SONGS-2, and the Cycle 2 review of Waterford 3. The staff finds that the conclusion is also valid for PVNGS1 Cycle 2 since the same manufacturing process is used in the fuel fabrication. The densification augmentation factors can, therefore, be eliminated for PVNGS1 Cycle 2.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 2 is performed by using the approved thermal-hydraulic code TORC (Ref. 9) and the CE-1 critical heat flux (CHF) correlation (Ref. 10). The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D (Ref. 11). The CETOP-D model has been verified to predict the 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 3, which is evaluated and approved below in Section 5.0 of this evaluation. 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 13. 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 MODIFIED STATISTICAL COMBINATION OF UNCERTAINTIES (SCU)

The licensees requested NRC review and approval of the topical report, "Modified Statistical Combination of Uncertainties," CEN-356(V)-P, Revision 01-P, July 1987 (Ref. 3). This report describes changes to the methodology for statistically combining uncertainties to obtain overall uncertainty factors. The overall uncertainty factors are used to determine the limiting safety system setting (LSSS) and limiting condition for operation (LCO) for the PVNGS COLSS and CPC system.

The existing SCU method treats uncertainties in two groups. The uncertainties in one group (system parameter uncertainties) include engineering factors, critical heat flux (CHF) correlation uncertainties and code modeling uncertainties which are statistically combined to generate a DNBR probability density function. The 95/95 probability/confidence level limit of this function is then used as the setpoint analysis minimum DNBR. The uncertainties in the other group (state parameter uncertainties) include measured state parameter, COLSS and CPC algorithm, radial peaking factor measurement, simulator model, computer processing and startup measurement uncertainties. These uncertainties are also statistically combined to determine the CPC and COLSS overall uncertainty factors.

Although the uncertainties within each group are combined statistically and a 95/95 probability/confidence level generated for each group, the resultant uncertainties of the two groups are effectively combined in a deterministic manner due to the separate application of the two uncertainty limits. The proposed modified SCU methodology would statistically combine uncertainty components which were previously applied deterministically. In addition, the statistical treatment of several uncertainty components would be modified so that the overall uncertainty factors can be calculated and applied as a function of burnup, axial shape index (ASI), and power in COLSS and CPC.

The staff has reviewed the uncertainties and the uncertainty treatment procedure described for the proposed modified SCU methodology and has determined that the resultant penalties applied to the COLSS power operating limit and the CPCS DNBR and local power density (LPD) calculations adequately incorporate all uncertainties at the 95/95 probability/confidence level. The analytical methods reviewed show that a DNBR limit of 1.24 with the uncertainty penalties derived in the report provides a 95/95 probability/confidence level assurance against DNB occurring during steady state operation or anticipated operational occurrences at the Palo Verde Nuclear Generating Station. The proposed methodology is, therefore, acceptable for use with the Palo Verde Nuclear Generating Station digital monitoring and protection systems.

6.0 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 APS 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 APS 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.

6.1 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 nct 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 excurson 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 is 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.

6.2 Loss of Forced Reactor Coolant Flow

The loss of coolant flow (LOF) event was reanalyzed by APS due to the change in the CPC trip. Rather than using the low DNBR trip, the LOF event for Cycle 2 was analyzed with a CPC trip based on low reactor coolant pump (RCP) shaft speed, which is initiated when the shaft speed drops to 95% of its initial speed. The analysis also used a more rapid coastdown than the reference analysis. The results show that this event initiated from the Technical Specification LCOs in conjunction with the low RCP shaft speed trip will not exceed the minimum DNBR limit of 1.24. As in the reference analysis, no credit is taken for the slight RCS pressure increase in computing this minimum DNBR. The acceptance criteria stated in SRP Section 15.3.1, therefore, are met and the staff concludes that the results of the LOF event for Cycle 2 are acceptable.

6.3 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 rodded configurations allowed by the PDIL 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. Since there is no trip assumed, the peak will stabilize at this asymptotic value after a few minutes as the secondary side continues to demand 100% power.

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 by then, the operator will take action to reduce power in accordance with Figure 3.1-1A of the Technical Specifications. A maximum allowable initial 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.

6.4 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 LCC 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. 14) 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 to be acceptable.

6.5 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 1925°F, a peak local clad oxidation percentage of 4.6%, 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 for both large and small break LOCAs, acceptable 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.

7.0 EVALUATION OF TECHNICAL SPECIFICATION CHANGES

In support of Cycle 2 operation, the licensees have requested a number of changes to the Technical Specifications (Ref. 1 and 4). The specific changes and the staff's evaluation are presented below.

(1) The maximum allowed enrichment of reload fuel specified in Technical Specification 5.3.1 has been increased from 4.0 to 4.05 weight percent U-235.

The slight increase in allowed reload fuel enrichment has been properly accounted for in the Cycle 2 reload analysis and results in acceptable consequences. The effect of the higher enrichment on the storage of fuel assemblies has been evaluated in Section 3.1 above and found to be acceptable.

(2) 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.

(3) 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 and 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 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 $+0.22\times10^{-4}$ to $+0.5\times10^{-4}$ delta k/k/°F. APS has reevaluated the most limiting transients and accidents which can be adversely affected by a positive MTC and has found them to be bounded by the reference cycle (Cycle 1) analysis. 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.

(4) 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 operational band of the pressurizer was made more restrictive, thus 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.

(5) 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.

(6) 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 is 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 PVNGS1 is operated within the safety analysis. Therefore, it is acceptable.

(7) 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 changed from 1861 psia to 1860 psia.

The modified SCU methodology discussed above in Section 5.0 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.

(8) 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 the changes in 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.

(9) 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.

(10) 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.

(11) 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 revised Specification also delineates how LHR is to be monitored and changes the format of the Action statement.

As stated above in Section 3.4 of this evaluation, 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.

(12) 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 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 analyses by modifying the figures, increase operator reliability by placing DNBR operating limits in one place, and eliminate superfluous information. The changes are, therefore, acceptable.

(13) 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% 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 change is acceptable.

(14) 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.

8.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, and thermal-hydraulics information presented in the PVNGS1 Cycle 2 reload report. The staff has also reviewed the proposed Technical Specification revisions, the SCU modification, 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 has been advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

10.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents 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 hazards consideration, and there has been no public comment on such findings. 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 need 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.

Principal Contributor: L. Kopp

Dated: October 21, 1987

REFERENCES

1. Reload Technical Specification Amendment, submitted by letter from J. G. Haynes (ANPP) dated June 29, 1987.

2. Reload Analysis Report for Palo Verde Nuclear Generating Station Unit 1 Cycle 2, submitted by letter from J. G. Haynes (ANPP) dated June 29, 1987.

3. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P, Revision 01-P, Combustion Engineering, July 1987.

4. Revision to Reload Technical Specification Amendment - Attachment 2, Shutdown Margin, submitted by letter from J. G. Haynes (ANPP) dated August 20, 1987.

5. Letter from J. G. Haynes (ANPP) dated July 13, 1987.

6. Response to NRC Questions Regarding the Unit 1 Cycle 2 Reload Submittal, submitted by letter from J. G. Haynes (ANPP) dated August 20, 1987.

7. "CEPAN Method of Analyzing Creep Collapse of Oval Cladding, Volume 5: Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," EPRI NP-3966-CCM, April 1985.

8. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, Combustion Engineering, April 1983.

9. "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P, Combustion Engineering, July 1975.

10. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spaces Grids, Part 1, Uniform Axial Power Distribution," CENPD-162-P-A, Combustion Engineering, April 1975.

11. "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.

12. "Safety Evaluation of CEN-161 (FATES3)," submitted by letter from R. A. Clark (NRC), to A. E. Lundvall, Jr. (BG&E), March 31, 1983.

13. "Fuel and Poison Rod Bowing," CENPD-225-P-A, Combustion Engineering, June 1983.

14. "HERMITE Space-Time Kinetics," CENPD-188-A, Combustion Engineering, July 1976.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and 75% of all detectors, with at least one detector in each quadrant at each level; and

b. A minimum of six tilt estimates, with at least one at each of three levels.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of three OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if the system has just been returned to OPERABLE status or if 7 days or more have elapsed since last use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.02g) shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.