October 4, 1988

Docket Nos.: 50-361
and 50-362

Mr. Kenneth P. Baskin
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Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas and Electric Company
101 Ash Street
Post Office Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: REISSUANCE OF AMENDMENTS 64 AND 53 TO FACILITY OPERATING LICENSES NPF-10 AND NPF-15, RESPECTIVELY (TAC NOS. 54737, 54738, 65547, 65548, 66816, 66817, 66821, 66822, 66823, 66826 AND 66287)

On August 3, 1988, the Commission issued the above amendments. Through an error in reproduction, not all the Technical Specifications pages were printed. Therefore, we have reissued those amendments in their entirety.

We have also found that a symbol was omitted in Item 15 of Table 3.3-1. The revised amendments correct this error also. We apologize for any inconvenience caused by these mistakes.

Sincerely,

original signed by

Donald E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
As stated

cc: See next page

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

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

August 3, 1988

Docket Nos.: 50-361 and 50-362

Mr. Kenneth P. Baskin
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Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas & Electric Company
101 Ash Street
Post Office Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NPF-10 AND AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NPF-15 SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 (TAC NOS. 54737/54738/65547/65548/66816/66817/66820/66821/66822/66823/66826/66827)

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 64 to Facility Operating License No. NPF-10 and Amendment No. 53 to Facility Operating License No. NPF-15 for the San Onofre Nuclear Generating Station, Units 2 and 3, located in San Diego County, California.

These amendments revise a number of Technical Specifications (TS), and are in partial response to your applications for amendments designated as PCN-197, 231, 233, 234, 239 and 155, dated May 12, December 14, and November 4, 1987 and April 27, 1984. The Technical Specifications that are changed by each PCN are as follows: PCN-197 - TS 3.1.3.6, "Regulating CEA Insertion Limits," TS 3.1.3.7, "Part Length CEA Insertion Limits," TS 3.10.2, "Group Height, Insertion and Power Distribution Limits," and Bases 3/4.1.3, "Movable Control Assemblies"; PCN-231 - Figure 3.1-2 of TS 3/4.1.3.6, "Regulating CEA Insertion Limits"; PCN-233 - TS 3/4.10.2, "Group Height, Insertion and Power Distribution Limits," and Tables 2.2-1 and 3.3-1; PCN-234 - TS 3/4.5.1, "Safety Injection Tanks"; PCN-239 - TS 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation"; and PCN-155 - TS 3/4.4.10, "Reactor Coolant Gas Vent System."

A copy of the related safety evaluation supporting this amendment is enclosed. The enclosed Notice of Issuance will, which relates to approval of PCN 231, 233 and 234, be forwarded to the Office of the Federal Register for publication. Approval of PCN-155, 197 and 239 will be included in a separate Notice of Issuance in the Commission's bi-weekly Federal Register Notice.

Proposed Change Number 232 concerning TS 3/4.3.3 "Monitoring Instrumentation," has been denied, as indicated in our letter to you dated April 19, 1988.

8808310199 2pp.

A copy of the related Notice of Denial is enclosed for your information. The Notice has been forwarded to the Office of the Federal Register for publication.

The remaining undispositioned changes proposed in your letters dated May 12 and December 14, 1987 require further staff review which is currently underway.

Sincerely,

Donald E. Hickman, Project Manager
Project Directorate V
Division of Reactor Project - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 64 to NPF-10
2. Amendment No. 53 to NPF-15
3. Safety Evaluation
4. Environmental Assessment
5. Notice of Issuance
6. Notice of Denial

cc: See next page

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
License No. NPF-10

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 2 (the facility) filed by the Southern California Edison Company (SCE) on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees), dated April 27, 1984 and May 12, November 4, and December 14, 1987 (as supplemented April 14 and May 6, 1988), comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

8810310208 S 9pp.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this amendment and Paragraph 2.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 64, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The changes in Technical Specifications are to become effective within 30 days of issuance of the amendment, with the exception of the change to Technical Specification 3/4.4.10, which is to become effective prior to startup from the Cycle 5 Refueling Outage. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensees shall adhere to the Technical Specifications existing at the time. The period of time during changeover shall be minimized.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 64

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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

Amendment Pages	Overleaf Pages
v	--
xi	xii
2-4	2-3
3/4 1-22	3/4 1-21
3/4 1-23	--
3/4 1-24	--
3/4 1-25	--
3/4 1-26	--
3/4 3-3	--
3/4 3-4	--
3/4 3-6	3/4 3-5
3/4 3-28	3/4 3-27
3/4 4-35	--
3/4 4-36	--
3/4 5-1	3/4 5-2
3/4 10-2	3/4 10-1
B3/4 1-5	--
B3/4 4-9	--

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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	COLD SHUTDOWN - Loops Filled	3/4 4-5
	COLD SHUTDOWN - Loops Not Filled	3/4 4-6
3/4.4.2	SAFETY VALVES - OPERATING	3/4 4-7
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3/4.4.5	REACTOR COOLANT SYSTEM LEAKAGE	
	LEAKAGE DETECTION SYSTEMS	3/4 4-16
	OPERATIONAL LEAKAGE	3/4 4-17
3/4.4.6	CHEMISTRY	3/4 4-20
3/4.4.7	SPECIFIC ACTIVITY	3/4 4-23
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3/4.4.10	REACTOR COOLANT GAS VENT SYSTEM	3/4 4-35
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3/4.5.1	SAFETY INJECTION TANKS	3/4 5-1
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3/4.5.3	ECCS SUBSYSTEMS - T_{avg} < 350°F	3/4 5-7
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SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION..... B 3/4 4-1

3/4.4.2 SAFETY VALVES..... B 3/4 4-1

3/4.4.3 PRESSURIZER..... B 3/4 4-2

3/4.4.4 STEAM GENERATORS..... B 3/4 4-2

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE..... B 3/4 4-4

3/4.4.6 CHEMISTRY..... B 3/4 4-4

3/4.4.7 SPECIFIC ACTIVITY..... B 3/4 4-5

3/4.4.8 PRESSURE/TEMPERATURE LIMITS..... B 3/4 4-6

3/4.4.9 STRUCTURAL INTEGRITY..... B 3/4 4-9

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM..... B 3/4 4-9

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS..... B 3/4 5-1

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS..... B 3/4 5-1

3/4.5.4 REFUELING WATER TANK..... B 3/4 5-2

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3/4.6.3	CONTAINMENT ISOLATION VALVES	B 3/4 6-4
3/4.6.4	COMBUSTIBLE GAS CONTROL	B 3/4 6-5

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥24.23% (4)
9. Local Power Density - High (5)	≤ 21.0 kw/ft	≤ 21.0 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≥ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

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

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

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

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

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grids is 1.31. The bypass setpoint may be changed during testing pursuant to Special Test Exception 3.10.2.

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

FLOOR is the minimum value of the trip setpoint.

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

(7) Acceleration, horizontal/vertical, g.

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

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

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

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6

a. When COLSS is in-service, the regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2. The CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

   1. Less than or equal to 4 hours per 24 hour interval,

   2. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day Interval, and

   3. Less than or equal to 14 Effective Full Power Days per 365 Effective Full Power Day Interval.

b. When COLSS is out-of-service, the regulating CEA groups shall be limited to the Short Term Steady State Insertion Limit shown on Figure 3.1-2. The CEA insertion between the Long Term Steady State Insertion Limits and the Short Term Steady State Insertion Limits is restricted to:

   1. Less than or equal to 4 hours per 24 hour interval,

   2. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day Interval, and

   3. Less than or equal to 14 Effective Full Power Days per 365 Effective Full Power Day Interval.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

When COLSS is in service and

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups 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 CEA group position using the above figure.

---

*See Special Test Exceptions 3.10.2 and 3.10.4.

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

ACTION: (Continued)

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

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

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

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 365 EFPD interval, either:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

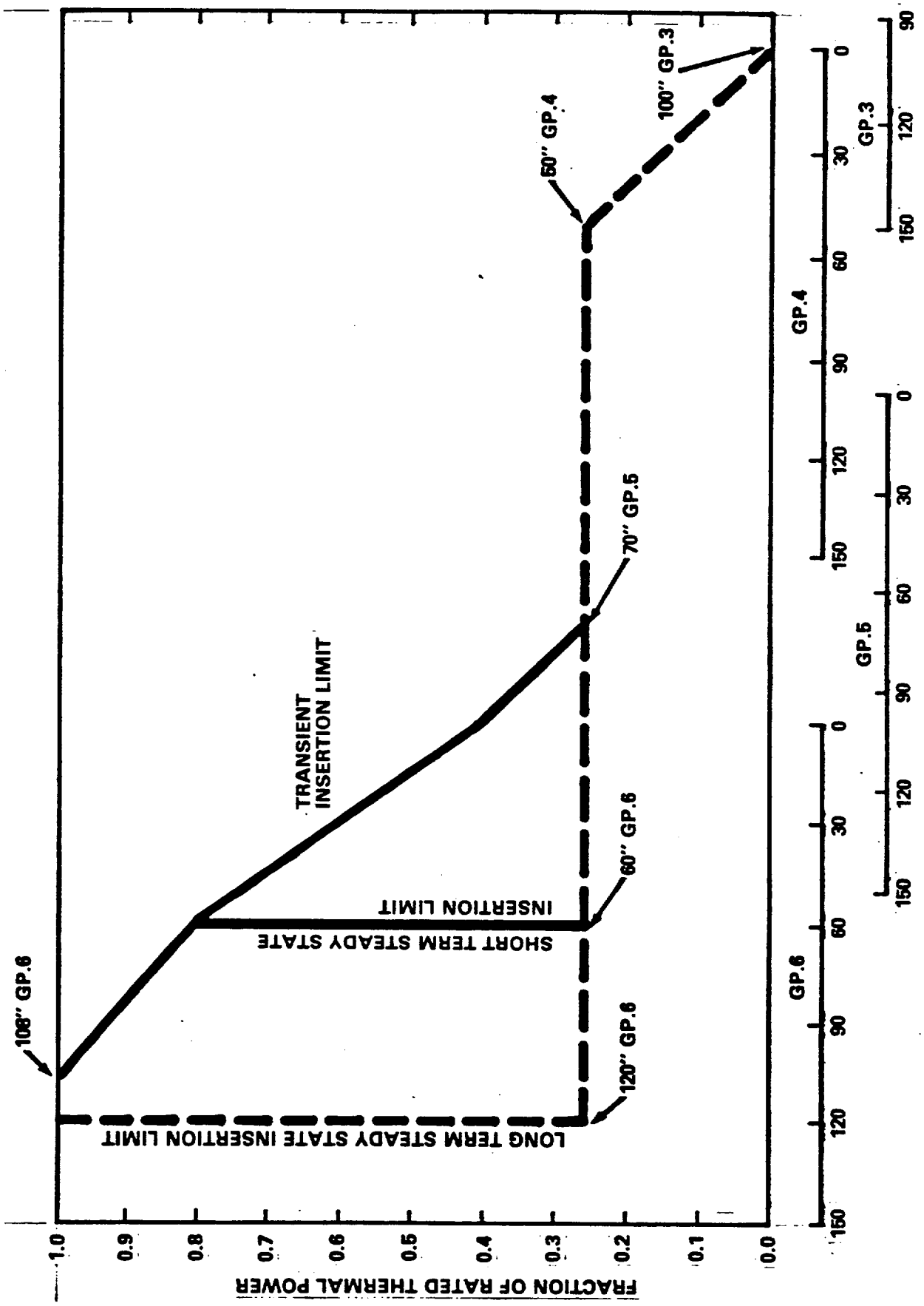

FIGURE 3.1-2
CEA INSERTION LIMITS

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The part length CEA group shall be limited to the insertion limits shown on Figure 3.1-3 with PLCEA insertion between the Long Term Steady State Insertion Limit and the Transient Insertion Limit restricted to:

a. ≤ 7 Effective Full Power Days per 30 Effective Full Power Day Interval, and

b. ≤ 14 Effective Full Power Days per 365 Effective Full Power Day Interval.

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

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 groups to within the limit, 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 3.1-3.

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 365 EFPD interval, either:

   1. Restore the part length groups to within the Long Term Steady State Insertion Limit within two hours, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA groups shall be determined to be within the Transient Insertion Limit at least once per 12 hours. The accumulated time during which the part length CEA groups are inserted beyond the Long Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours.

*See Special Test Exception 3.10.2.

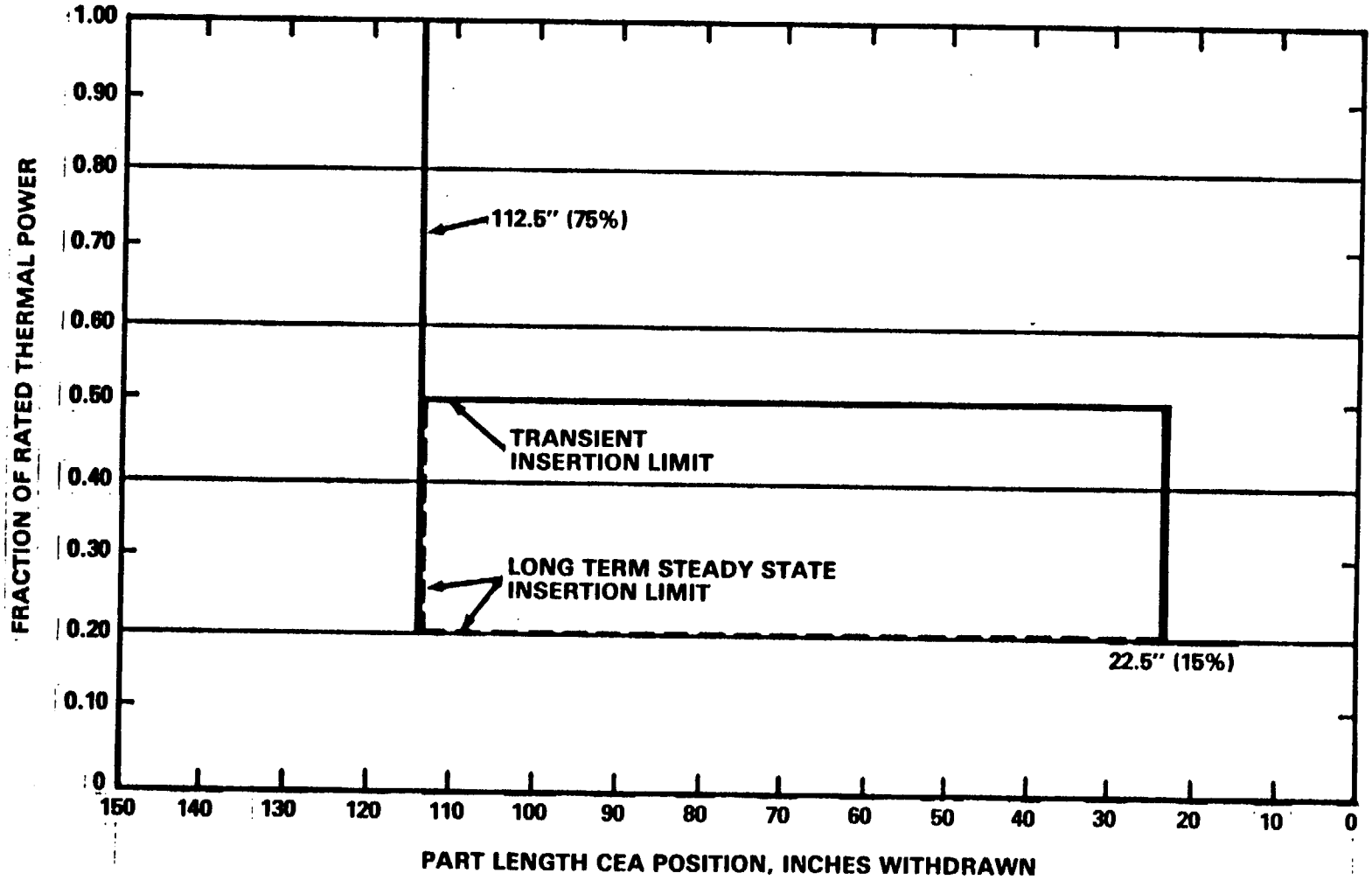

Figure 3.1-3 PART 1 CEA INSERTION LIMIT vs. THERMAL POWER

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)(e)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)(e)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2	2#, 3#
				3*, 4*, 5*	7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	7A
14. Core Protection Calculators	4	2(c)(d)(e)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6#, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) 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.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia.

(c) 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. During testing pursuant to Special Test Exception 3.10.2 or 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

ACTION STATEMENTS

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

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

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

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

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

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that Action 6.b is met.* If the exemption to Specification 3.0.4 is used, Action 6.b must be 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 Specification 3.2.4.d (COLSS out of service) is satisfied.

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

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
1. Manual	
a. SIAS	
Safety Injection	Not Applicable
Control Room Isolation	Not Applicable
Containment Isolation (3)	Not Applicable
Containment Emergency Cooling	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable
f. CCAS	
Containment Emergency Cooling	Not Applicable
g. EFAS	
Auxiliary Feedwater	Not Applicable
h. CRIS	
Control Room Isolation	Not Applicable
i. TGIS	
Toxic Gas Isolation	Not Applicable
j. FHIS	
Fuel Handling Building Isolation	Not Applicable
k. CPIS	
Containment Purge Isolation	Not Applicable

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(c) Charging Pumps	31.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (Note 4b)	23.2*
(c) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (Note 4b)	23.0*
(c) Emergency Cooling Fans	21.0*
b. CIAS	
(1) Containment Isolation	10.9* (NOTE 2)
(2) Main Feedwater Isolation and Backup Isolation Valves (HV 4048, HV 4052, HV 1105, HV 1106, HV 4047, HV 4051)	10.9
(3) CCW Valves (Note 4a)	20.9
(4) Mainsteam Isolation Valves (HV 8204, HV 8205)	8.9
(5) Minipurge Isolation Valves	5.9
4. Containment Pressure - High-High	
CSAS	
Containment Spray	23.0*

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10 The Reactor Coolant Gas Vent System shall be OPERABLE with:

a. At least one of valves 2HV0296A or 2HV0296B capable of being powered from an emergency bus and providing a vent path from the reactor vessel head; and,

b. At least one of valves 2HV0297A or 2HV0297B capable of being powered from an emergency bus and providing a vent path from the pressurizer steam space; and,

c. At least one of valves 2HV0298, capable of being powered from an emergency bus and providing a vent path to the containment atmosphere, or 2HV0299, capable of being powered from an emergency bus and providing a vent path to the quench tank; and

d. Valves 2HV0296A, 2HV0296B, 2HV0297A, 2HV0297B, 2HV0299 and 2HV0298 all closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any of valves 2HV0296A, 2HV0296B, 2HV0297A or 2HV0297B inoperable, operation may continue provided that:

   i) power is removed from the inoperable valve(s) within 4 hours; and,

   ii) valves 2HV0299 and 2HV0298 are maintained closed and power is removed within 4 hours; and,

   iii) the inoperable valve(s) is restored to OPERABLE status during the next COLD SHUTDOWN.

b. With any of valves 2HV0299 or 2HV0298 inoperable, operation may continue provided that:

   i) power is removed from the inoperable valve(s) within 4 hours; and,

   ii) valves 2HV0296A, 2HV0296B, 2HV0297A and 2HV0297B are all maintained closed and power is removed within 4 hours; and

LIMITING CONDITION FOR OPERATION

iii) the inoperable valve(s) is restored to OPERABLE status during the next COLD SHUTDOWN.

c. The provisions of 3.0.4 are not applicable for entry into MODES 3, 2 and 1.

SURVEILLANCE REQUIREMENTS

4.4.10 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1850 and 2800 ppm of boron, and

d. A nitrogen cover-pressure of between 615 and 655 psia.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure greater than or equal to 715 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days by verifying the fuses removed from each safety injection tank vent valve.

d. At least once per 31 days when the RCS pressure is above 715 psia, by verifying that the isolation valve operator breakers are padlocked in the open position.

e. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. Before an actual or simulated RCS pressure signal exceeds 715 psia, and

   2. Upon receipt of an SIAS test signal.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

MOVABLE CONTROL ASSEMBLIES (Continued)

The establishment of LSSS and LCOs require 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 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 Specification 3.1.3.6 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 Specification 3.1.3.6 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 Part Length CEA Insertion Limits of Specification 3.1.3.7 ensure that adverse power shapes and rapid local power changes which affect radial peaking factors and DNB considerations do not occur as a result of a part length CEA group covering the same axial segment of the fuel assemblies for an extended period of time during operation.

The CEA fully withdrawn position is defined to be greater than or equal to 145 inches. The extreme limits of CEA travel, fully withdrawn and fully inserted, may be described as the upper electrical limit and lower electrical limit respectively.

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertant safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

Reactor coolant system gas vents are provided to exhaust noncondensible gases from the primary system that could inhibit natural circulation core cooling following a non-design bases accident. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The design redundancy of the Reactor Coolant Gas Vent System serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant Gas Vent System are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. NPF-15

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 3 (the facility) filed by the Southern California Edison Company (SCE) on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees), dated April 27, 1984 and May 12, November 4, and December 14, 1987 (as supplemented April 14 and May 6, 1988), comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this amendment and Paragraph-2.C(2) of Facility Operating License No. NPF-15 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 53, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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

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TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 21.0 kw/ft	≤ 21.0 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

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

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

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

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

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10-4% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10-4% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grid is 1.31. The bypass setpoint may be changed during testing pursuant to Special Test Exception 3.10.2.

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

FLOOR is the minimum value of the trip setpoint.

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

(7) Acceleration, horizontal/vertical, g.

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

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

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

LIMITING CONDITION FOR OPERATION

3.1.3.6

a. When COLSS is in-service, the regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2. The CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

   1. Less than or equal to 4 hours per 24 hour interval,
   2. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and
   3. Less than or equal to 14 Effective Full Power Days per 365 Effective Full Power Day interval.

a. When COLSS is out-of-service, the regulating CEA groups shall be limited to the Short Term Steady State Insertion Limit shown on Figure 3.1-2. The CEA insertion between the Long Term Steady State Insertion Limits and the Short Term Steady State Insertion Limits is restricted to:

   1. Less than or equal to 4 hours per 24 hour interval,
   2. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and
   3. Less than or equal to 14 Effective Full Power Days per 365 Effective Full Power Day interval.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

When COLSS is in service and

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups 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 CEA group position using the above figure.

---

*See Special Test Exceptions 3.10.2 and 3.10.4.

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

ACTION: (Continued)

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

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

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

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 365 EFPD interval, either:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

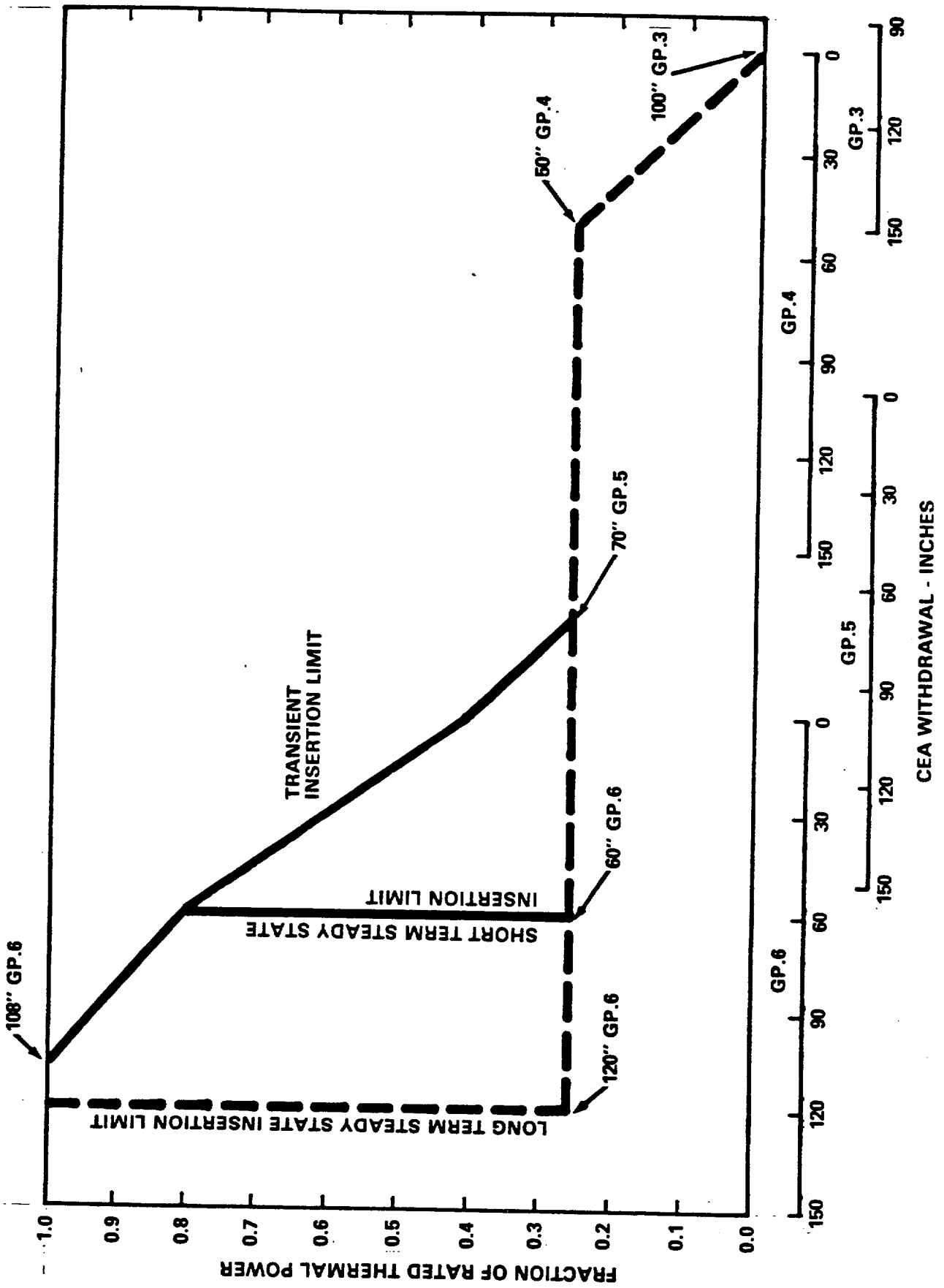

FIGURE 3.1-2
CEA INSERTION LIMITS VS FRACTION OF ALLOWABLE THERMAL POWER

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The part length CEA group shall be limited to the insertion limits shown on Figure 3.1-3 with PLCEA insertion between the Long Term Steady State Insertion Limit and the Transient Insertion Limit restricted to:

a. ≤ 7 Effective Full Power Days per 30 Effective Full Power Day interval, and

b. ≤ 14 Effective Full Power Days per 365 Effective Full Power Day interval.

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

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 groups to within the limit, 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 3.1-3.

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 365 EFPD interval, either:

   1. Restore the part length groups to within the Long Term Steady State Insertion Limit within two hours, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 4 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. The accumulated time during which the part length CEA groups are inserted beyond the Long Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours.

*See Special Test Exception 3.10.2.

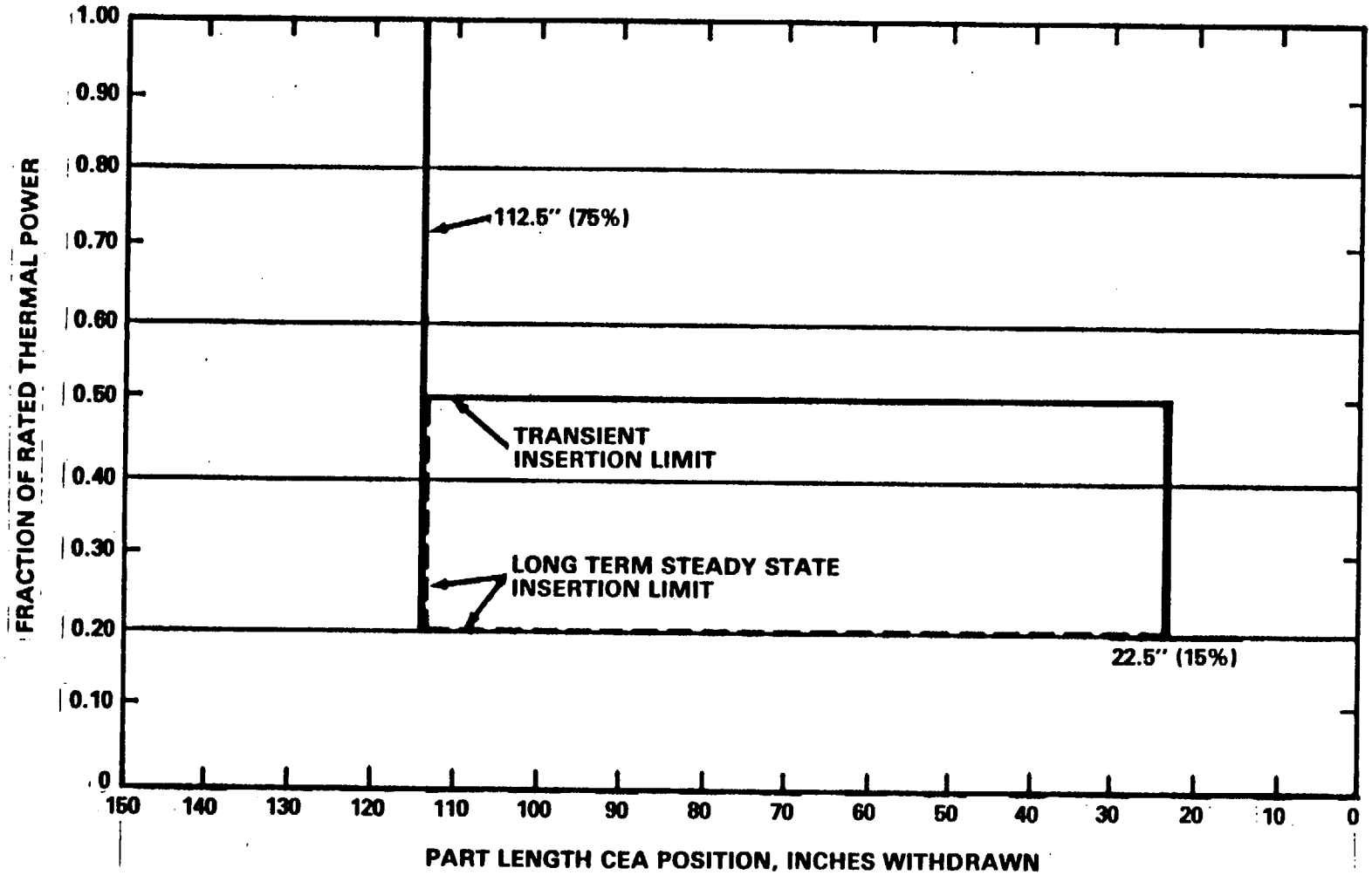

FIGURE 3.1-3 Part Length CEA Insertion Limit Vs. Thermal Power

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)(e)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)(e)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2	2#, 3#
				3*, 4*, 5*	7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	7A
14. Core Protection Calculators	4	2(c)(d)(e)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6#, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) 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.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia.

(c) 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. During testing pursuant to Special Test Exception 3.10.2 or 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

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

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

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

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that ACTION 6.b is met.* If the exemption to Specification 3.0.4 is used, Action 6.b must be met.

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

1. Within 1 hour the DNBR margin reqired by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

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

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
1. Manual	
a. SIAS	
Safety Injection	Not Applicable
Control Room Isolation	Not Applicable
Containment Isolation (3)	Not Applicable
Containment Emergency Cooling	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable
f. CCAS	
Containment Emergency Cooling	Not Applicable
g. EFAS	
Auxiliary Feedwater	Not Applicable
h. CRIS	
Control Room Isolation	Not Applicable
i. TGIS	
Toxic Gas Isolation	Not Applicable
j. FHIS	
Fuel Handling Building Isolation	Not Applicable
k. CPIS	
Containment Purge Isolation	Not Applicable

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(c) Charging Pumps	31.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (NOTE 4b)	23.2*
(c) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (NOTE 4b)	23.0*
(c) Emergency Cooling Fans	21.0*
b. CIAS	
(1) Containment Isolation	10.9* (NOTE 2)
(2) Main Feedwater Backup Isolation and Backup Isolation Valves (HV 4048, HV 4052, HV 1105, HV 1106, HV 4047, HV 4051)	10.9
(3) CCW Valves (Note 4a)	20.9
(4) Mainsteam Isolation Valves (HV 8204, HV 8205)	8.9
(5) Minipurge Isolation Valves	5.9
4. Containment Pressure - High-High	
CSAS	
Containment Spray	23.0*

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10 The Reactor Coolant Gas Vent System shall be OPERABLE with:

a. At least one of valves 3HV0296A or 3HV0296B capable of being powered from an emergency bus and providing a vent path from the reactor vessel head; and,

b. At least one of valves 3HV0297A or 3HV0297B capable of being powered from an emergency bus and providing a vent path from the pressurizer steam space; and,

c. At least one of valves 3HV0298, capable of being powered from an emergency bus and providing a vent path to the containment atmosphere, or 3HV0299, capable of being powered from an emergency bus and providing a vent path to the quench tank; and

d. Valves 3HV0296A, 3HV0296B, 3HV0297A, 3HV0297B, 3HV0299 and 3HV0298 all closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any of valves 3HV0296A, 3HV0296B, 3HV0297A, or 3HV0297B inoperable, operation may continue provided that:

i) power is removed from the inoperable valve(s) within 4 hours; and,

ii) valves 3HV0299 and 3HV0298 are maintained closed and power is removed within 4 hours; and,

iii) the inoperable valve(s) is restored to OPERABLE status during the next COLD SHUTDOWN.

b. With any of valves 3HV0299 or 3HV0298 inoperable, operation may continue provided that:

i) power is removed from the inoperable valve(s) within 4 hours; and,

ii) valves 3HV0296A, 3HV0296B, 3HV0297A and 3HV0297B are all maintained closed and power is removed within 4 hours; and

LIMITING CONDITION FOR OPERATION

iii) the inoperable valve(s) is restored to OPERABLE status during the next COLD SHUTDOWN.

c. The provisions of 3.0.4 are not applicable for entry into MODES 3, 2 and 1.

SURVEILLANCE REQUIREMENTS

4.4.10 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed.

b. A contained borated water volume of between 1680 and 1807 cubic feet.

c. Between 1850 and 2800 ppm of boron, and

d. A nitrogen cover-pressure of between 615 and 655 psia.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure greater than or equal to 715 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days by verifying the fuses removed from each safety injection tank vent valve.

d. At least once per 31 days when the RCS pressure is above 715 psia, by verifying that the isolation valve operator breakers are padlocked in the open position.

e. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. Before an actual or simulated RCS pressure signal exceeds 715 psia, and

2. Upon receipt of an SIAS test signal.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

MOVABLE CONTROL ASSEMBLIES (Continued)

The establishment of LSSS and LCOs require 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 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 Specification 3.1.3.6 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 Specification 3.1.3.6 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 Part Length CEA Insertion Limits of Specification 3.1.3.7 ensure that adverse power shapes and rapid local power changes which affect radial peaking factors and DNB considerations do not occur as a result of a part length CEA group covering the same axial segment of the fuel assemblies for an extended period of time during operation.

The CEA fully withdrawn position is defined to be greater than or equal to 145 inches. The extreme limits of CEA travel, fully withdrawn and fully inserted, may be described as the upper electrical limit and lower electrical limit respectively.

TABLE B 3/4.4-1 (Continued)

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
205-03	C-6831-1	A508CL1	Inlet Nozzle Forging S/E	-20	12	40	124
205-03	C-6831-2	A508CL1	Inlet Nozzle Forging S/E	-20	12	40	124
205-03	C-6831-3	A508CL1	Inlet Nozzle Forging S/E	-20	-15	50	114
205-03	C-6831-4	A508CL1	Inlet Nozzle Forging S/E	-20	-15	50	114
205-07	C-6832-1	A508CL1	Outlet Nozzle Forging S/E	-20	-20	0	159
205-07	C-6832-2	A508CL1	Outlet Nozzle Forging S/E	-20	-20	0	152
231-01	C-6833-1	A533GRBCL1	Closure Head Peel	-40	20	NA	NA
231-01	C-6834-1	A533GRBCL1	Closure Head Peel	-30	10	NA	NA
231-02	C-6835-1	A533GRBCL1	Closure Head Dome	-40	10	NA	NA

NA = Not Available

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or an RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs is less than or equal to 285°F. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

Reactor coolant system gas vents are provided to exhaust noncondensible gases from the primary system that could inhibit natural circulation core cooling following a non-design bases accident. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The design redundancy of the Reactor Coolant Gas Vent System serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant Gas Vent System are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL.

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

Southern California Edison Company (SCE), on behalf of itself and the other licensees, San Diego Gas and Electric Company, The City of Riverside, California, and The City of Anaheim, California, has submitted a number of applications for license amendments for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The NRC staff's evaluation of six of these applications (referred to as Proposed Change Numbers 197, 231, 233, 234, 239 and 155) is described below.

2.0 DISCUSSION

PCN-197

By letter dated May 12, 1987 the licensees submitted proposed change PCN-197 to the SONGS Units 2 and 3 Technical Specifications. The proposed changes would revise Technical Specification 3.1.3.6, "Regulating CEA Insertion Limits," 3.1.3.7, "Part Length CEA Insertion Limits," and 3.10.2, "Group Height, Insertion and Power Distribution Limits," as well as Bases 3/4.1.3, "Movable Control Assemblies."

Technical Specification 3.1.3.6 currently provides restrictions on control element assembly (CEA) insertion limits to periods less than or equal to 14 effective full power days (EFPD) per calendar year. The proposed change would replace "calendar year" with "365 EFPD interval." Since calendar year may be interpreted as the period from January through December, which is not the intent of the specified interval, the proposed change provides greater clarification of the intended interval restriction. The staff finds this nomenclature change acceptable.

The part length CEA insertion limits of Technical Specification 3.1.3.7 are intended to ensure that adverse power shapes and rapid local power changes which affect radial peaking factors and departure from nucleate boiling (DNB) considerations do not occur as a result of a part length

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CEA group covering the same axial segment of the fuel assemblies for an extended period of time during operation. However, the Specification does not clearly specify the allowable duration within the transient insertion limit nor does it clearly address operation within the long term steady state insertion limit. The long term steady state limit is based on the expected variation of the steady state radial peaking factors with burnup. It serves to limit the behavior of the radial peaking factors within acceptable bounds determined from analyses. The transient insertion limit aids in ensuring that the minimum shutdown margin is maintained and that the potential effects of a CEA ejection accident are limited to acceptable levels. Long term operation at the transient insertion limit 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 proposed change would restrict the part length CEA group insertion to the insertion limits of Figure 3.1-3 with insertion between the long term steady state insertion limit and the transient insertion limit restricted to intervals less than or equal to 7 EFPD per 30 EFPD interval and intervals less than or equal to 14 EFPD per 365 EFPD interval. This is acceptable since it clarifies the intent of Specification 3.1.3.7 to apply the 7 EFPD per 30 EFPD interval only to insertions beyond the long term steady state insertion limit.

The proposed change would also revise the actions to be taken if part length CEA groups are inserted beyond the transient insertion limit or between the long term steady state insertion limit and the transient insertion limit for excessive periods of time. The staff finds these revised Action Statements acceptable.

The applicability of Specification 3.1.3.7 would be revised from Modes 1 and 2 to Mode 1 above 20% of rated power. The staff finds this proposed change acceptable since plant operation is limited to no greater than 5% of rated power in Mode 2, and below 20% of rated power part length CEA insertion is unrestricted.

In addition the proposed change to Specification 4.1.3.7 would revise the Surveillance Requirement so that part length CEA group position is determined every 12 hours to be within the transient insertion limit and the accumulated time for insertion beyond the long term steady state insertion limit but within the transient insertion limit is determined every 24 hours. The staff considers these surveillance intervals adequate to allow the appropriate proposed actions to be taken in sufficient time. The proposed Surveillance Requirement changes are, therefore, acceptable.

The proposed change would also revise Specification 3.10.2 to allow suspension of the insertion limits of Specification 3.1.3.7 during special physics tests. This is acceptable since the addition of power dependent insertion limits for the part length CEAs makes Specification 3.1.3.7 similar to Specification 3.1.3.6 which has previously been included as a special test exception in Specification 3.10.2.

Finally, the proposed change would revise the Bases to Specification 3/4.1.3 to clarify the extreme limits of CEA travel (fully withdrawn and fully inserted CEA positions). The terms "upper electrical limit" and "lower electrical limit" are used to describe the fully withdrawn and fully inserted CEA position. Furthermore, the withdrawn position would be defined as greater than or equal to 145 inches. This change is acceptable since it is administrative in nature and provides clarification in terminology which should avoid misunderstandings.

PCN-231

By letter dated December 14, 1987 the licensees submitted proposed change PCN-231 which would revise Figure 3.1-2 of Technical Specification 3/4.1.3.6, "Regulating CEA Insertion Limits" to relax the control element assembly (CEA) power dependent insertion limits (PDIL) at power levels of 25% of rated thermal power or less. The licensee has requested this change in order to preclude the need to borate prior to startup to assure that the estimated critical CEA position is within the zero power CEA insertion limits. This will also help reduce the amount of waste water generated during startup.

The proposed change would allow more CEAs to be inserted at power levels below 25% of rated thermal power. This results in higher reactivity insertion rates in the event of a CEA initiated reactivity accident. The greater CEA insertion at low power also results in a decrease in shutdown margin. Therefore, the licensee reevaluated the three limiting events that are affected by individual CEA worth and required shutdown margin. These are the CEA withdrawal at low power, the CEA ejection at zero power, and the steam line break at zero power.

The revised PDIL results in a maximum reactivity insertion rate of 1.7×10^{-4} delta k/k/sec during a CEA withdrawal event at low power compared to 1.1×10^{-4} delta k/k/sec reported in the Cycle 3 reload analysis report. However, for the Cycle 3 reload, the licensee performed a parametric study on reactivity insertion rate in order to maximize peak reactor coolant system pressure. This resulted in an intermediate insertion rate (less than 1.1×10^{-4} delta k/k/sec) producing the most adverse results. The licensee has verified that this remains true even for the requested PDIL revision and, therefore, the previous Cycle 3 analysis for the CEA withdrawal event remains bounding.

The CEA ejection event at zero power is initiated by a higher ejected CEA worth than for Cycle 3 due to the revised PDIL. However, the Cycle 1 reference analysis allowed greater CEA insertion at zero power and, therefore, an even higher ejected CEA worth. The CEA ejection event with the proposed revised PDIL is, therefore, bounded by the reference analysis of Cycle 1.

The most restrictive shutdown margin requirement is based on the postulated steam line break event at no load operating temperature and the resulting

uncontrolled reactor coolant system cooldown. The reference analysis (Cycle 1) assumed a shutdown margin of 5.15% delta k/k for this event. The proposed revised PDIL results in a greater calculated shutdown margin than 5.15% delta k/k and, thus, is bounded by the reference analysis.

In summary, although the proposed change allows more CEA insertion at lower power relative to current Cycle 3 allowances, it is still less than was allowed for Cycle 1. A reevaluation of the limiting events which are adversely affected by greater CEA insertion have shown that the previous reference analysis results are still bounding. The proposed relaxation of the PDIL at 25% of rated thermal power and below is therefore acceptable.

PCN-233

By letter dated December 14, 1987 the licensees submitted proposed change PCN-233 which would revise Technical Specification 3/4.10.2, "Group Height, Insertion and Power Distribution Limits," and Tables 2.2-1 and 3.3-1 for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. The first part of the proposed change would reference footnote 5 of Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," and footnote C of Table 3.3-1, "Reactor Protective Instrumentation," in the body of Special Test Exception 3.10.2. This part of the change would also modify (1) footnote 5 of Table 2.2-1 to indicate that the bypass setpoint for core protection calculator (CPC) generated trips may be changed during testing pursuant to Special Test Exception 3.10.2, and (2) footnote C of Table 3.3-1 to indicate that the trip may be manually bypassed below 5% of rated thermal power during testing pursuant to Special Test Exception 3.10.2. The second part of the proposed change would modify Surveillance 4.10.2.2 to reference Specification 4.2.1.2. The final change would modify Action 6 of Table 3.3-1 by inserting an exemption to Specification 3.0.4 in the event of inoperability of one or both control element assembly calculators (CEACs).

CEA bank reactivity worth measurements which are performed during low power physics testing routinely generate abnormal CEA configurations which could cause a CPC trip. Therefore, the bistable setpoint for these CPC trips (local power density and DNBR) must be raised during physics testing as currently allowed by Special Test Exception 3.10.3. However, since this specification test exception was developed for partial reactor coolant system flow conditions which are not applicable during normal physics testing, the more appropriate test exception would be 3.10.2. One of the proposed changes, therefore, allows resetting of the plant trip bistable from 10^{-4}% to 5% power in Special Test Exception 3.10.2 to allow performance of the tests that are now accomplished pursuant to Exception 3.10.3. This change does not alter the physics test program and is acceptable to the staff.

Another proposed change would reference Surveillance Specification 4.2.1.2 in Surveillance 4.10.2.2. Since 4.2.1.2 specifies conditions within which the linear heat rate is to be determined, it is the appropriate surveillance to reference. The change, therefore, ensures that the intended surveillance will be performed and is acceptable.

The final portion of the proposed change would modify Action 6 of Table 3.3-1 by adding a Specification 3.0.4 exemption when one or both CEACs are inoperable. An exemption to Specification 3.0.4 allows plant startup in situations where a specified system or component is in an inoperable condition. Such exemptions are normally authorized in cases where system or component inoperability would be allowed indefinitely under the provisions of the Action requirements. Since operation of the plant with both CEACs inoperable is currently allowed to continue indefinitely under the conditions of Action 6.b, the staff finds the proposed change to be consistent with the generic intent of the exemption and is therefore acceptable.

PCN-234

By letter dated December 14, 1987 the licensees submitted proposed change PCN-234 which would revise Technical Specification 3/4.5.1, "Safety Injection Tanks". The existing Limiting Condition for Operation (LCO) 3.5.1.d requires that each reactor coolant system safety injection tank be Operable with a nitrogen cover-pressure of between 600 and 625 psig. This requirement of the LCO ensures that a sufficient volume of borated water will be immediately forced into the reactor core through the cold legs of the Reactor Coolant System (RCS) in the event that the RCS pressure falls below the pressure of the safety injection tanks (SITs). This surge of water into the core provides the initial cooling mechanism during large pipe ruptures within the reactor coolant pressure boundary. The proposed change would revise the required upper limit of the nitrogen cover-pressure from 625 psig to 640 psig. This change would prevent possible violation of the pressure limit in the other SITs due to inleakage from the common fill header when one of the tanks is being filled to maintain its pressure within limits. SCE has evaluated the effect of this higher SIT pressure upon the previously evaluated large break loss of coolant accident (LOCA). By letters dated April 14 and May 6, 1988 SCE provided the results of this evaluation. The higher pressure would cause an additional 160 cubic feet of water to flow out the break but there would still be sufficient water remaining in the SITs after blowdown is complete to fill the downcomer. Also, the higher pressure would result in core reflood commencing sooner. The net effect is to reduce the peak fuel clad temperature. The licensee has determined that a SIT pressure in excess of 1000 psig would be necessary to cause an unacceptable amount of flow loss during the blowdown period so that insufficient liquid would remain to fill the downcomer. Therefore, the proposed increase in the upper limit of the nitrogen cover-pressure is acceptable because its effect is bounded by the previous accident analysis.

In addition, the proposed change would revise the units of pressure from pounds per square inch gauge (psig) to pounds per square inch absolute (psia). This will make the units of measurement consistent with other units on the control room panel. This change is acceptable because it is administrative in nature and provides consistency with other pressure measurements displayed in the control room.

Finally, the proposed change would delete the Unit 3 Cycle 2 specific lower SIT boron concentration requirement from the Limiting Condition for Operation. This change is acceptable because this requirement only applied to Cycle 2 operation of Unit 3, which has been completed. Therefore, this specification is no longer applicable.

PCN-239

By letter dated November 4, 1987 the licensees submitted proposed change PCN-239 which would revise Technical Specification (TS) 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation" to correct an editorial discrepancy in Table 3.3-5 of the specification. Table 3.3-5, "Engineered Safety Features Response Times," specifies the Main Steam Isolation Valve (MSIV) response time in two locations, one associated with the Main Steam Isolation Signal (MSIS) and the other associated with the Containment Isolation Actuation Signal (CIAS). Amendment 46 for Unit 2 and Amendment 35 for Unit 3, issued on May 16, 1986, approved an increase in the MSIV response time from 6.9 to 8.9 seconds but only in association with the MSIS signal. The proposed change would increase the MSIV response time in association with the CIAS signal from 6.9 seconds to 8.9 seconds to make it consistent with that of the MSIS signal. The safety evaluation supporting the MSIS signal also applies to the CIAS response time. This change will provide consistency in Table 3.3-5 and it will not affect plant equipment or operation nor will it affect previously analyzed accidents. Therefore, this change is acceptable.

PCN-155

By letter dated April 27, 1984 the licensees submitted proposed change PCN-155 which would add Section 3/4.4.10, "Reactor Coolant Gas Vent System" (RCGVS) and its bases to the Technical Specifications. This proposed change was subsequently revised to correct typographical errors by letter dated June 3, 1988. The primary safety concern with the RCGVS is the ability to isolate the system. The proposed Technical Specification would maintain the system isolated and, in the event of a malfunction of any valve in the system, would remove power from the redundant isolation valve(s) in that vent path(s). The ACTION statement provides acceptable time limits for returning the system to operability since the RCGVS is not used for normal plant operations or for mitigation of any design basis accident. The exception to Section 3.0.4 of the Technical Specifications is acceptable for the same reason. Therefore, the staff finds the proposed Section 3/4.4.10 to be acceptable.

3.0 SUMMARY OF STAFF EVALUATION

The staff has reviewed PCN 197, 231, 233, 234, 239 and 155. We find these changes to be acceptable as proposed and clarified by the licensee's submittals referenced herein.

4.0 CONTACT WITH STATE OFFICIAL

The NRC staff has advised the Chief of the Radiological Health Branch, State Department of Health Services, State of California, of the proposed determination of no significant hazards consideration. No comments were received.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, an environmental assessment related to PCN-231, 233 and 234 has been published (53 FR 29291) in the Federal Register on August 3, 1988. The Commission has determined that the issuance of this amendment for those changes will not have a significant effect on the quality of the human environment.

The remaining changes, PCN-197, 239 and 155 covered by these amendments, involve changes in the installation or use of facility components located within the restricted area. The staff has determined that those portions of the amendments involve no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued proposed findings that the changes involve no significant hazards consideration, and there has been no public comment on such findings. Accordingly, those portions of the amendments meet 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 to be prepared in connection with the issuance of this portion of these amendments.

6.0 CONCLUSION

Based upon our evaluation of the proposed changes to the San Onofre Units 2 and 3 Technical Specification, we have concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Kopp and D. Hickman

Dated: August 3, 1988

UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL.

DOCKET NOS. 50-361 AND 50-362

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The United States Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. NPF-10 and NPF-15, issued to Southern California Edison Company, San Diego Gas and Electric Company, The City of Riverside, California and The City of Anaheim, California (the licensees), for operation of the San Onofre Nuclear Generating Station, Units 2 and 3, located in San Diego County, California.

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action:

The proposed amendments would incorporate proposed changes, identified as PCN-231, 233 and 234, as described below:

Proposed Change PCN-231 is a request to revise Technical Specification 3/4.1.3.6, "Regulating CEA Insertion Limits." The proposed change would revise Figure 3.1-2, relaxing the CEA insertion limits at low power levels to increase operating flexibility and to reduce the volume of radioactive waste water.

Proposed Change PCN-233 is a request to revise Technical Specification 3/4.10.2, "Group Height, Insertion and Power Distribution Limits," and Tables 2.2-1 and 3.3-1. The proposed change would modify the existing Technical Specification to allow use of the correct Special Test Exception during the measurement of various CEA reactivity worths at low power levels. Tables 2.2-1 and 3.3-1 would incorporate this Special Test Exception in Footnotes (5)

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and (C), respectively. Additionally, the proposed change would specify, in Table 3.3-1, an exemption from the requirements of Specification 3.0.4 if the CEACs are inoperable. PCN-233 would also correct a reference error in Surveillance Requirement 4.10.2.2.

Proposed Change PCN-234 is a request to revise Technical Specification 3/4.5.1, "Safety Injection Tanks." The proposed change would increase the upper limit on SIT cover gas pressure from 625 to 640 psig and change the designated units from psig to psia.

The Need for the Proposed Action:

The proposed changes would provide operational flexibility, clarify and correct the Technical Specifications, and reduce the volume of radioactive waste water.

Environmental Impacts of the Proposed Action:

The Commission has completed its evaluation of the proposed revisions to the Technical Specifications and has concluded that the proposed changes provide reasonable assurance that the facility can be operated safely. The proposed changes do not increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in the allowable individual or cumulative occupational radiation exposure. Changing the insertion limits will decrease the volume of radioactive waste water and changing the SIT cover gas pressure will result in core reflood after a large break LOCA commencing sooner. Accordingly, the Commission concludes that this proposed action would result in no significant adverse radiological environmental impact.

With regard to potential nonradiological impacts, the proposed change to the Technical Specifications involves systems located within the restricted

area as defined in 10 CFR Part 20. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed amendments.

The Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on March 31, 1988 (53 FR 10452). No request for hearing or petition for leave to intervene was filed following these notices.

Alternative to the Proposed Action:

Since the Commission concluded that there is no significant adverse environmental effect that would result from the proposed action, alternatives with equal or greater environmental impacts need not be evaluated.

The principal alternative would be to deny the requested amendments. Denial of the request would not reduce environmental impacts of plant operation and in fact would prevent a reduction in the volume of radioactive waste water generated at the facility.

Alternative Use of Resources:

This action does not involve the use of resources not previously considered in the Final Environmental Statement Related to the Operation of the San Onofre Nuclear Generating Station, Units 2 and 3, dated April 1981.

Agencies and Persons Consulted:

The NRC staff has reviewed the licensee's request and did not consult other agencies or persons.

FINDING OF NO SIGNIFICANT IMPACT

The Commission has determined not to prepare an environmental impact statement for the proposed license amendment.

Based upon this environmental assessment, we conclude that the proposed action will not have a significant adverse effect on the quality of the human environment.

For further details with respect to this action, see the application for amendments dated December 14, 1987 and the supplementary information provided by letters dated April 14 and May 6, 1988, which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C., and at the General Library, University of California at Irvine, Irvine, California 92713.

Dated at Rockville, Maryland, this 29th day of July, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

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

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL.

DOCKET NOS. 50-361 AND 50-362

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 64 to Facility Operating License No. NPF-10 and Amendment No. 53 to Facility Operating License No. NPF-15, issued to Southern California Edison Company, San Diego Gas and Electric Company, The City of Riverside, California and The City of Anaheim, California (the licensees), which revised the Technical Specifications for operation of the San Onofre Nuclear Generating Station, Units 2 and 3, located in San Diego County, California.

The amendment was effective as of the date of issuance.

These amendments revise a number of Technical Specifications (TS), and are in partial response to applications for amendments designated as PCN-231, 233 and 234. The Technical Specifications that are changed by each PCN are as follows: PCN-231 - Figure 3.1-2 of TS 3/4.1.3.6, "Regulating CEA Insertion Limits"; PCN-233 - TS 3/4.10.2, "Group Height, Insertion and Power Distribution Limits," and Tables 2.2-1 and 3.3-1; and PCN-234 - TS 3/4.5.1, "Safety Injection Tanks."

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

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Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on March 31, 1988 (53 FR 10452). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined that an environmental impact statement will not be prepared and that issuance of the amendments will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the applications for amendments dated December 14, 1987 and supplemental letters dated April 14 and May 6, 1988, (2) Amendment No. 64 to License No. NPF-10 and Amendment No. 53 License No. NPF-15, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street N.W., and at the General Library, University of California, P.O. Box 19557, Irvine, California 92713. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this 3rd day of August 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

D. E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL

DOCKET NOS. 50-361/362

NOTICE OF PARTIAL DENIAL OF AMENDMENTS TO

FACILITY OPERATING LICENSES AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied part of a request by Southern California Edison Company, San Diego Gas and Electric Company, The City of Riverside, California and the City of Anaheim, California (licensees) for amendments to Facility Operating License Nos. NPF-10 and NPF-15, issued to the licensees, for operation of the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, located in San Diego County, California. The Notice of Consideration of Issuance of Amendments was published in the FEDERAL REGISTER on March 31, 1988 (53 FR 10452).

The amendments, as proposed by the licensees, would revise Technical Specification (TS) Surveillance Requirement 4.3.3.2(a) by changing the frequency of performance of the Incore Detection System Channel Check from 7 days to 31 Effective Full Power Days. The proposed change would allow verification of incore detector operability to be performed in conjunction with other routine surveillances.

The staff finds that there are no plant-unique characteristics of SONGS 2 or 3 which would justify deviation from the Incore Detection System Operability Requirements specified in the Standard Technical Specifications. In addition, the staff does not consider the reported error introduced by a

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single soft (undetected) detector failure to be insignificant. The degree of compromise to measurement quality which is acceptable is a judgment which should be considered generically.

By September 9, 1988, the licensees may demand a hearing with respect to the denial described above and any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for a hearing or petition for leave to intervene must be filed with the Secretary for the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Charles R. Kocher, Esq., Southern California Edison Company, 2244 Walnut Grove Avenue, P.O. Box 800, Rosemead, California 91770, and Orrick, Herrington & Sutcliff, Attn: David R. Pigott, Esq., 600 Montgomery Street, San Francisco, California 94111.

For further details with respect to this action, see (1) the application for amendments dated December 14, 1987, and (2) the Commission's Notification of Denial forwarded to the licensees by letter dated April 19, 1988, which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the General Library, University of California at Irvine, Irvine, California 92713. A copy of Item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission,

Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 3rd day of August 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald E. Hickman

Donald E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects