July 14, 1995

Mr. Harold B. Ray
Executive Vice President
Southern California Edison Company
P.O. Box 128
San Clemente, California 92674-0128

SUBJECT: ISSUANCE OF AMENDMENT FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2 (TAC NO. M87775) AND UNIT NO. 3 (TAC NO. M87776)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. NPF-10 and Amendment No. 109 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated September 15, 1993, as supplemented by letter dated September 6, 1994.

These amendments revise TS Table 2.2-1 "Reactor Protective Instrumentation Trip Setpoint Limits," Table 3.3-1, "Reactor Protective Instrumentation," Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," and Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values," and the associated Bases. The revisions to the notes in these tables change the pressure at which the Low Pressurizer Pressure (LPP) trip bypass shall be automatically removed to a consistent value of "before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia)." In addition, the wording of the notes is revised to make the notes more consistent with each other.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

Mel B. Fields, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000361
P PDR

Docket Nos. 50-361
and 50-362

Enclosures:
1. Amendment No. 120 to NPF-10
2. Amendment No. 109 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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WBateman	ACRS, (4), T2E26
MFields	CGrimes, 011E22
LHurley, RIV	RJones
KPerkins, WCFO (4)	

DOCUMENT NAME: S087775.AMD

MLO

OFC	PDIV-2/LA	PDIV-2/PM	NRR:SRXB	OGC
NAME	EPeyton	MFields:ye	RJones	
DATE	6/21/95	6/ /95	6/5/95	6/13/95

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Executive Vice President
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Original Signed By

Mel B. Fields, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
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Docket Nos. 50-361
and 50-362

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WBateman	ACRS, (4), T2E26
MFields	CGrimes, O11E22
LHurley, RIV	RJones
KPerkins, WCFO (4)	

DOCUMENT NAME: SO87775.AMD

MLO

OFC	PDIV-2/LA	PDIV-2/PM	NRR:SRXB	OGC
NAME	EPeyton	MFields:ye	RJones	
DATE	6/27/95	6/ /95	6/5/95	6/13/95

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 14, 1995

Mr. Harold B. Ray
Executive Vice President
Southern California Edison Company
P.O. Box 128
San Clemente, California 92674-0128

SUBJECT: ISSUANCE OF AMENDMENT FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2 (TAC NO. M87775) AND UNIT NO. 3 (TAC NO. M87776)

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Sincerely,

Mel B. Fields

Mel B. Fields, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-361
and 50-362

Enclosures:
1. Amendment No. 120 to NPF-10
2. Amendment No. 109 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

cc w/encls:
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Mayor
City of San Clemente
100 Avenida Presidio
San Clemente, California 92672

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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 NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-10

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

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated September 15, 1993, as supplemented by letter dated September 6, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 120, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mel B. Fields

Mel B. Fields, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 14, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-4	2-4
B 2-3	B 2-3
3/4 3-4	3/4 3-4
3/4 3-19	3/4 3-19
3/4 3-26	3/4 3-26
B 3/4 3-1a	B 3/4 3-1a

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	≤ 111.0% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.83% of RATED THERMAL POWER	≤ 0.93% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2375 psia	≤ 2385 psia
5. Pressurizer Pressure - Low (2)	≥ 1740 psia	≥ 1700 psia
6. Containment Pressure - High	≤ 3.1 psig	≤ 3.4 psig
7. Steam Generator Pressure - Low (3)	≥ 741 psia	≥ 729 psia
8. Steam Generator Level - Low	≥ 21% (4)	≥ 20.0% (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)	≤ 7.25 psid (6)(8)
12. Steam Generator Level - High	≤ 89% (4)	≤ 89.7% (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. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 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.

BASES

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA, or certain intermediate steam line breaks. This trip initiates a reactor trip at a linear power level of less than or equal to 111.0% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to 0.93% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2385 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1700 psia. This trip's setpoint may be manually decreased, to a minimum value of 300 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia). The ≤ 472 psia value represents an allowable value which includes margin to account for instrument loop uncertainties and ensures the 500 psia analytical limit will not be exceeded.

BASES

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to safety injection actuation.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

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

Local Power Density-High

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

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

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

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

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 source range neutron flux monitors may be used in Modes 3, 4, and 5 with the reactor trip circuit breakers open or the Control Element Assembly (CEA) Drive System not capable of CEA withdrawal.

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) Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 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 3.3-3 (Continued)

TABLE NOTATION

(a) Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia).

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Actuated equipment only; does not result in CIAS.

(d) Applicability for SDVS is Modes 1, 2, 3, and 4 when the diesel generator circuit breaker is open.

\# The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

** With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - 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.

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.	Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	(8)	(8)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	(6)(7)	(6)(7)
ii. Particulate	(6)(7)	(6)(7)
iii. Iodine	(6)(7)	(6)(7)
c. Containment Area Radiation (Gamma)	$\leq$ 325 mR/hr (MODES 1-4) $\leq$ 2.4 mR/hr (MODE 6)	$\leq$ 340 mR/hr (MODES 1-4) $\leq$ 2.5 mR/hr (MODE 6)
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia.

(2) 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.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Deleted.

(5) Actuated equipment only; does not result in CIAS.

(6) The trip setpoint shall be set sufficiently high to prevent spurious alarms/trips yet sufficiently low to assure an alarm/trip should an inadvertent release occur.

(7) Prior to the completion of DCP 53N, the setpoints for Containment Airborne Radiation Monitor 2RT-7804-1 shall be determined by the ODCM.

(8) The trip setpoint shall be set sufficiently high to prevent spurious alarm/trips yet sufficiently low to assure an alarm/trip should a fuel handling accident occur.

---

*Variable setpoints are for use only during <u>normal</u>, <u>controlled</u> plant heatups and cooldowns.

**Above normal background.

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

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

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

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

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The quarterly frequency for the CHANNEL FUNCTIONAL TESTS for these systems is based on the analyses presented in the NRC approved topical report, CEN-327, "RPS/ESFAS Extended Test Interval Evaluation," as supplemented.

The measurement of response time at the specified frequencies provides assurance that the reactor protective and ESF actuation associated with each channel is completed within the time limit assumed in the accident analyses.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

No credit was taken in the analyses for those channels with response times indicated as not applicable.

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

The Pressurizer Pressure-Low trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia). The ≤ 472 psia value represents an allowable value which includes margin to account for instrument loop uncertainties and ensures the 500 psia analytical limit will not be exceeded.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 109
License No. NPF-15

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

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated September 15, 1993, as supplemented by letter dated September 6, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 109, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mel B. Fields

Mel B. Fields, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 14, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-4	2-4
B 2-3	B 2-3
3/4 3-4	3/4 3-4
3/4 3-19	3/4 3-19
3/4 3-26	3/4 3-26
B 3/4 3-1a	B 3/4 3-1a

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.0% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.83% of RATED THERMAL POWER	$\leq$ 0.93% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2375 psia	$\leq$ 2385 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1740 psia	$\geq$ 1700 psia
6. Containment Pressure - High	$\leq$ 3.1 psig	$\leq$ 3.4 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 741 psia	$\geq$ 729 psia
8. Steam Generator Level - Low	$\geq$ 21.0% (4)	$\geq$ 20.0% (4)
9. Local Power Density - High (5)	$\leq$ 21.0 kw/ft	$\leq$ 21.0 kw/ft
10. DNBR - Low	$\geq$ 1.31 (5)	$\geq$ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.22 psid/sec (6)(8)	$\leq$ 0.231 psid/sec (6)(8)
b) Floor	$\geq$ 13.2 psid (6)(8)	$\geq$ 12.1 psid (6)(8)
c) Step	$\leq$ 6.82 psid (6)(8)	$\leq$ 7.25 psid (6)(8)
12. Steam Generator Level - High	$\leq$ 89% (4)	$\leq$ 89.7% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

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

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 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.

BASES

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA, or certain intermediate steam line breaks. This trip initiates a reactor trip at a linear power level of less than or equal to 111.0% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to 0.93% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2385 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1700 psia. This trip's setpoint may be manually decreased, to a minimum value of 300 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia). The ≤ 472 psia value represents an allowable value which includes margin to account for instrument loop uncertainties and ensures the 500 psia analytical limit will not be exceeded.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to safety injection actuation.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

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

Local Power Density-High

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

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

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

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

TABL' '-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 source range neutron flux monitors may be used in Modes 3, 4, and 5 with the reactor trip circuit breakers open or the Control Element Assembly (CEA) Drive System not capable of CEA withdrawal.

#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) Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 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 3.3-3 (Continued)

TABLE NOTATION

(a) Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia).

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Actuated equipment only; does not result in CIAS.

(d) Applicability for SDVS is Modes 1, 2, 3, and 4 when the diesel generator circuit breaker is open.

The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

** With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - 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.

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. Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3. Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, 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 inoperable channel in the tripped condition within 1 hour.

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.	Containment Pressure Circuit	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3.	Steam Generator Level - Low	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

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 9 are satisfied.

ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	(8)	(8)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	(6)(7)	(6)(7)
ii. Particulate	(6)(7)	(6)(7)
iii. Iodine	(6)(7)	(6)(7)
c. Containment Area Radiation (Gamma)	$\leq$ 325 mR/hr (MODES 1-4) $\leq$ 2.4 mR/hr (Mode 6)	$\leq$ 340 mR/hr (MODES 1-4) $\leq$ 2.5 mR/hr (MODE 6)
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia).

(2) 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.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Deleted.

(5) Actuated equipment only; does not result in CIAS.

(6) The trip setpoint shall be set sufficiently high to prevent spurious alarms/trips yet sufficiently low to assure an alarm/trip should an inadvertent release occur.

(7) Prior to the completion of DCP 53N, the setpoints for Containment Airborne Radiation Monitor 3RT-7804-1 shall be determined by the ODCM.

(8) The trip setpoint shall be set sufficiently high to prevent spurious alarm/trips yet sufficiently low to assure an alarm/trip should a fuel handling accident occur.

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* Variable setpoints are for use only during normal, controlled plant heatups and cooldowns.

**Above normal background.

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

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

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

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

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The quarterly frequency for the CHANNEL FUNCTIONAL TESTS for these systems is based on the analyses presented in the NRC approved topical report, CEN-327, "RPS/ESFAS Extended Test Interval Evaluation," as supplemented.

The measurement of response time at the specified frequencies provides assurance that the reactor protective and ESF actuation associated with each channel is completed within the time limit assumed in the accident analyses.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

No credit was taken in the analyses for those channels with response times indicated as not applicable.

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

The Pressurizer Pressure-Low trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is ≤ 472 psia). The ≤ 472 psia value represents an allowable value which includes margin to account for instrument loop uncertainties and ensures the 500 psia analytical limit will not be exceeded.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated September 15, 1993, as supplemented by letter dated September 6, 1994, Southern California Edison Company, et al. (SCE or the licensee) submitted a request for changes to the Technical Specifications (TS) for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3. The proposed changes would revise TS Table 2.2-1 "Reactor Protective Instrumentation Trip Setpoint Limits," Table 3.3-1, "Reactor Protective Instrumentation," Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," and Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values," and the associated Bases. The revisions to the notes in these tables change the pressure at which the low pressurizer pressure (LPP) trip bypass shall be automatically removed to a consistant value of "before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is 472 psia)." In addition, the wording of the notes is revised to make the notes more consistent with each other.

The September 6, 1994, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination which was published in the Federal Register on September 29, 1993 (58 FR 50975).

2.0 DISCUSSION

One of the functions of the reactor protection system (RPS) is to initiate a reactor trip whenever the pressurizer pressure falls below the trip setpoint. The engineered safety feature actuation system (ESFAS) instrumentation initiates a safety injection actuation signal (SIAS) whenever the pressurizer pressure falls below the safety injection setpoint. The reactor trip and the safety injection actuation functions are provided to mitigate the consequences of an accident such as a main steam line break (MSLB) or a loss of coolant accident (LOCA).

The LPP trip setpoint may be decreased in a manner prescribed by the TS, with the minimum allowed setpoint equal to 300 psia. Bypass of the LPP trip and actuation of safety injection, namely RPS/ESFAS bypass, is provided to allow for systems testing at low pressure and to allow heatup and cooldown without generating an undesired safeguard action. This bypass may be manually initiated when pressurizer pressure drops below the bypass permissive setpoint and is automatically removed when pressurizer pressure rises above the bypass permissive setpoint.

The existing Note (2) of Table 2.2-1, in part, states that the LPP trip may be manually bypassed below 400 psia and that the bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia. Note (b) of Table 3.3-1, Note (a) of Table 3.3-3, and Note (1) of Table 3.3-4, in part, state that the LPP trip may be manually bypassed below 400 psia and that the bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia. As noted, there is an inconsistency in the existing tables associated with the pressure at which the automatic removal of the LPP trip bypass becomes effective.

The licensee proposes to remove this inconsistency by revising the TS to state that the LPP trip bypass shall be automatically removed at a pressurizer pressure equal to or less than 472 psia. The pressurizer pressure at which the LLP trip may be manually bypassed, below 400 psia, is not being changed by this TS amendment request. The new value of 472 psia is an allowable value which includes total loop uncertainties for the instruments used in the LPP trip circuit and ensures the analytical limit of 500 psia will not be exceeded. The licensee verified that the analytical limit of 500 psia was acceptable by evaluating all accident scenarios where the pressurizer low pressure setpoint is relied upon to mitigate the consequences of the accident.

In addition to removing inconsistencies in the TS, increasing the value at which the LPP trip bypass must be automatically removed reduces the possibility of inadvertent actuation of the safety injection system during plant shutdown. This is because the minimum allowed LPP setpoint is 300 psia and, due to an instrument hysteresis of approximately 75 psi, the operators currently have to reduce RCS pressure to 325 psia before the instruments will permit LPP bypass. This small margin between the LPP trip (300 psia) and the LPP bypass (325 psia) presents an increased chance of inadvertent actuation of the safety injection system. Increasing the pressure at which the LPP trip bypass must be automatically removed from 400 psia to 472 psia will allow the operators to bypass the LPP trip at a pressurizer pressure much closer to the 400 psia TS limit during plant shutdown.

Due to the proposed TS changes, the appropriate sections of the Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the changes. These sections include Sections 7.2.1.1.1.6, "Low Pressurizer Pressure," 7.2.1.1.5, "Bypasses," Table 7.2-2, "Reactor Protective System Bypasses,"

Table 7.2-7, "Plant Protection System Failure Mode and Effects Analysis," 7.3.1.1.1, "Safety Injection System," and Table 7.3-4, "ESFAS Bypasses."

3.0 EVALUATION

General Design Criterion (GDC) 20, "Protection system functions," of Appendix A to 10 CFR Part 50 states that the protection system shall be designed to (1) initiate automatically the operation of appropriate systems including reactivity control systems, to assure that specified fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) sense accident conditions and to initiate the operation of systems and components important to safety. The licensee evaluated the new LPP setpoint to assure that this regulatory requirement was complied with.

The postulated accidents that are affected most by the proposed change are the MSLB and LOCA. The RPS/ESFAS bypass, if in effect, would prevent a reactor trip and safety injection actuation on low pressurizer pressure, to mitigate the consequences of an MSLB or LOCA occurring at these conditions. The consequences of an unmitigated accident could include a potential return to criticality and subsequent approach to the specified acceptable fuel design limits and the potential for exceeding ECCS acceptance criteria.

The higher value (472 psia as opposed to 400 psia) at which automatic removal of LPP trip manual bypass becomes effective has been evaluated by the licensee. An MSLB outside containment with a RCS pressure equal to or less than 500 psia does not require safety injection actuation to mitigate the consequences of the event. At a RCS pressure of 500 psia, the maximum RCS temperature would be less than 467°F (saturation temperature at 500 psia). In the event of an MSLB at this condition, the total positive reactivity that would be added would be approximately 4.3 percent $\Delta\rho$ as compared to 3.9 percent $\Delta\rho$ that would be added if an MSLB were to occur at 400 psia. This 4.3 percent $\Delta\rho$ reactivity is the sum of Doppler and moderator reactivity additions and includes uncertainties. The shutdown margin required by the TS in Mode 3 is 5.15 percent $\Delta\rho$. Therefore, the shutdown margin is more than sufficient to offset the reactivity insertion due to an MSLB at 500 psia and preclude a return to criticality.

An MSLB inside containment or a LOCA would result in automatic SIAS generated by the ESFAS high containment pressure signal. The high containment pressure actuation of SIAS is maintained during all modes of plant operation. The high containment pressure SIAS setpoint trip is 3.4 psig (3.7 psig allowable value). As indicated above, an MSLB initiated at 500 psia does not require automatic SIAS to prevent a return to criticality. However, automatic SIAS on high containment pressure would occur for all but the smallest MSLB to provide mitigation for the MSLB inside containment.

The licensee further indicated that the consequences of a LOCA are not sensitive to the initial RCS pressure assumed (either 400 psia, 472 psia, or 500 psia). At 3.4 psig containment pressure, SIAS would actuate the safety injection equipment that is required to be operable by the TS to mitigate the event. The rate at which containment pressure increases to the SIAS setpoint following a LOCA is not sensitive to the initial RCS pressure. Automatic SIAS on high containment pressure would be expected for all LOCAs except for very small break LOCAs at the small end of the break spectrum. Containment analysis of the smallest break (0.01 ft^2) in the UFSAR indicates that automatic SIAS actuation on high containment pressure would occur prior to core uncovery. For smaller break sizes, for which a containment high pressure signal may not be generated, the time available would be sufficient to credit manual SIAS initiation to mitigate the event.

The licensee has evaluated the consequences of other accidents occurring with the RCS pressure at 500 psia without an automatic SIAS, and determined that these events were not limiting. These events included increased main steam flow, steam generator tube rupture, inadvertent opening of a pressurizer safety valve, primary sample or instrument line break, and loss of normal feedwater.

The staff has reviewed the analyses and assumptions provided by the licensee to justify the new LPP bypass setpoint. The staff has determined that the appropriate accident senerios were considered and that the analytical methodology used to calculate the change in safety limits was correctly utilized by the licensee. The staff concludes that these analyses demonstrate that the new bypass setpoint complies with the regulatory requirements contained in GDC 20. The staff therefore finds the proposed new bypass setpoint for the LPP to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 50975). Accordingly, 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 be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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