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 NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
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 August 30, 1991, 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. 101 , 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 the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 28, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 101 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 the captioned amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12a	3/4 3-12a
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
B 3/4 3-1	B 3/4 3-1

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),S(7) #(4,5),M(8)	Q(11),#(6)	1, 2
15. CEA Calculators	S	#	Q,#(6)	1, 2
16. Reactor Coolant Flow-Low	S	#	Q	1, 2
17. Seismic-High	S	#	Q	1, 2
18. Loss of Load	S	N.A.	Q	1 (9)

TABLE 4.3-1 (Continued)

TABLE NOTATION

(11) - The quarterly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(12) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
b. Containment Pressure - High	S	(6)	Q	1, 2, 3
c. Pressurizer Pressure - Low	S	(6)	Q	1, 2, 3,
d. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3, 4
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3
b. Containment Pressure -- High - High	S	(6)	Q	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
b. Manual SIAS (Trip Buttons)(5)	N.A.	N.A.	(6)	1, 2, 3, 4
c. Containment Pressure - High	S	(6)	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3, 4
4. MAIN STEAM ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3
b. Steam Generator Pressure - Low	S	(6)	Q	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3
5. RECIRCULATION (RAS)				
a. Refueling Water Storage Tank - Low	S	R	Q	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3, 4
6. CONTAINMENT COOLING (CCAS)				
a. Manual CCAS (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
c. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	(6)	(6)	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	(6)	Q	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	(6)	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	R	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(5)	N.A.
b. Airborne Radiation				
i. Gaseous	S	(6)	M	1,2,3,4,6
ii. Particulate	W	(6)	M	1,2,3,4,6
iii. Iodine	W	(6)	M	6
c. Containment Area Radiation (Gamma)	S	(6)	M	1,2,3,4,6
d. Automatic Actuation Logic	N.A.	N.A.	(3),(6)	1,2,3,4,6

TABLE NOTATION

(1) Deleted.

(2) Deleted.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays exempt from testing during plant operation shall be limited to only those relays associated with plant equipment which cannot be operated during plant operation. Relays not testable during plant operation shall be tested during each COLD SHUTDOWN exceeding 24 hours unless tested during the previous 6 months.

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

(6) At least once per refueling interval.

* With irradiated fuel in the storage pool.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

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.

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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
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 August 30, 1991, 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. 90 , 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 the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 28, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 90 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 the captioned amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12a	3/4 3-12a
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
B 3/4 3-1	B 3/4 3-1

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4), S(7), #(4,5), M(8)	Q(11),#(6)	1, 2
15. CEA Calculators	S	#	Q,#(6)	1, 2
16. Reactor Coolant Flow-Low	S	#	Q	1, 2
17. Seismic-High	S	#	Q	1, 2
18. Loss of Load	S	N.A.	Q	1 (9)

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

\# - At least once per Refueling Interval.

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - Above 55% of RATED THERMAL POWER.

(10) - Deleted.

TABLE 4.3-1 (Continued)

TABLE NOTATION

(11) - The quarterly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(12) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
b. Containment Pressure - High	S	(6)	Q	1, 2, 3
c. Pressurizer Pressure - Low	S	(6)	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	Q(1)(3), SA(4)	1, 2, 3, 4
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3
b. Containment Pressure -- High - High	S	(6)	Q	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	Q(1)(3), SA(4)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
b. Manual SIAS (Trip Buttons)(5)	N.A.	N.A.	(6)	1, 2, 3, 4
c. Containment Pressure - High	S	(6)	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	Q(1)(3), SA(4)	1, 2, 3, 4
4. MAIN STEAM ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3
b. Steam Generator Pressure - Low	S	(6)	Q	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	Q(1)(3), SA(4)	1, 2, 3
5. RECIRCULATION (RAS)				
a. Refueling Water Storage Tank - Low	S	R	Q	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	Q(1)(3), SA(4)	1, 2, 3, 4
6. CONTAINMENT COOLING (CCAS)				
a. Manual CCAS (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3, 4
c. Automatic Actuation Logic	N.A.	N.A.	Q(1)(3), SA(4)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	(6)	(6)	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	(6)	Q	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	(6)	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	Q(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	(6)	N.A.
b. Airborne Radiation				
i. Gaseous	S	(6)	M	1,2,3,4,6
ii. Particulate	W	(6)	M	1,2,3,4,6
iii. Iodine	W	(6)	M	6
c. Containment Area Radiation (Gamma)	S	(6)	M	1,3,3,4,6
d. Automatic Actuation Logic	N.A.	N.A.	(3), (6)	1,2,3,4,6

TABLE NOTATION

(1) Deleted.

(2) Deleted.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays exempt from testing during plant operation shall be limited to only those relays associated with plant equipment which cannot be operated during plant operation. Relays not testable during plant operation shall be tested during each COLD SHUTDOWN exceeding 24 hours unless tested during the previous 6 months.

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

(6) At least once per Refueling Interval.

* With irradiated fuel in the storage pool.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.*

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

*Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitor 3RT-7804-1 or 3RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 3RT-7804-1 and associated sampling media shall perform the above required functions.

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 consistant 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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 90 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 August 30, 1991, 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 3/4.3.1, "Reactor Protection System Instrumentation," and TS 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation." These amendments modify the channel functional and logic units surveillance test intervals from monthly to quarterly.

2.0 EVALUATION

The proposed amendment is based on topical reports CEN-327-A and CEN-327-A Supplement 1. Both reports were prepared by Combustion Engineering for the Combustion Engineering Owners Group (CEOG). The purpose of these reports was to evaluate the safety impact and provide justification for extending the current 30 day surveillance test interval for both RPS and ESFAS instrumentation. Both reports used probability risk analysis techniques to demonstrate that the proposed surveillance interval extensions do not result in increased plant risk when compared with current technical specification requirements.

The NRC evaluation and acceptance of the topical reports is documented by a safety evaluation report (SER) that was sent to the chairman of the CEOG on November 6, 1989. The NRC found that the referenced topical reports provide an acceptable generic basis to support plant specific TS changes for extending both RPS and ESFAS channel functional test intervals from monthly to quarterly.

9204230043 920228
PDR ADOCK 05000361
P PDR

The CE analysis estimated a slight increase in RPS unavailability as a result of extending the surveillance test interval. The analysis also estimated a reduced core melt frequency based on a reduction in surveillance test induced transients. The overall effect of the proposed change on safety was determined to be negligible. The result of reduced ESFAS testing on core melt frequency was found to be similar to that for RPS.

The San Onofre Units 2 and 3 Technical Specifications in section 3/4.3.1 provide instrumentation operability and surveillance requirements for the RPS. Technical Specification 4.3.1.1 and Table 4.3-1 specify the modes and frequency for the performance of channel checks, channel functional tests and channel calibration for each RPS channel.

The staff SER for CEN-327 concluded that the CE report did not address the effects of drift in both the sensors or instrument strings. The effects of drift are plant specific and therefore should be included with each individual plant analysis. As stated in the generic SER, each licensee should confirm that they have reviewed drift information including as found and as left values for each instrument channel involved and have determined that the drift occurring in that channel will remain bounded by the setpoint methodology for the extended surveillance interval. Additionally, the licensee should maintain records of the setpoint calculations and associated data to support future staff audits.

The licensee stated that the calibration of transmitters and signal processing equipment is normally done at each refueling interval and is not affected by the proposed increase in the functional test surveillance interval. The surveillance test calibration interval for this equipment is not being changed. However, the licensee stated that an increase from monthly to quarterly for the channel functional test does affect the bistable trip units. The licensee performed an analysis of the bistable trip unit drift records including as found and as left values. The licensee concluded that the trip setpoints will be within the established pass/fail criteria when testing is performed quarterly.

The staff requested the licensee to confirm that for any proposed extension of monthly functional test intervals, the bases for the 24 month calibration surveillance interval will not be compromised. The licensee stated that the surveillance and corrective maintenance history indicates that problems are identified as a result of the shift channel checks and during routine monitoring of plant parameters. Since the monthly functional test involves the injection of simulated signals into the RPS/ESFAS logic, any failure relating to instrument calibration would not be detected by this testing methodology. However, a channel check may reveal information identifying a calibration related problem. The channel check surveillance is not being revised by the licensee and will continue to be performed once per shift.

The CEOG topical report addressed the channel functional test frequency for all the functional units referenced in Table 4.3-1 except for the manual reactor trip, reactor trip breakers, and seismic high trip. The manual reactor trip functional test is currently specified to be performed every refueling outage and is not being revised. The reactor trip breakers channel functional test interval will remain 18 months. Functional units for the reactor protection system logic, core protection calculators, and control element assembly calculators were not identified as specific dominant cut sets in CEN-327-A but were considered in the analysis. Although seismic high is not listed as a cut set for RPS, the licensee has proposed that the seismic high functional test frequency be revised from monthly to quarterly based on the similarity of bistable design with the loss of load trip function. The licensee stated that the analysis used to justify the loss of load trip function functional test is also applicable to the seismic high functional unit. This conclusion was confirmed by CE.

Table 4.3-2 specifies the functional test surveillance requirements for the ESFAS. Topical report CEN-327-A addressed all the functional units referenced in Table 4.3-2 except for the containment cooling actuation signal (CCAS), the control room isolation signal (CRIS), the toxic gas isolation signal (TGIS), the fuel handling isolation signal (FHIS), and the containment purge isolation signal (CPIS). The licensee proposed to extend the surveillance interval for CCAS from monthly to quarterly with CRIS, TGIS, FHIS, and CPIS remaining as specified in the current TS. Again, CEN-327-A does not specifically address CCAS but the licensee stated that the CCAS and the SIAS share the same type of bistable and are designed similarly. Therefore, the licensee stated that the associated analysis justifying a functional test interval extension for SIAS is also applicable to CCAS. This conclusion was confirmed by CE.

3.0 CONCLUSION

The RPS/ESFAS test interval evaluation presented in CEN-327-A and CEN-327-A Supplement 1 developed a fault tree model for the four classes of RTS and three classes of ESFAS design. Each model addressed common mode failures, operator errors, reduced redundancy, and random component failures. These models were used to evaluate the RTS and ESFAS availability based on a 30 day and 90 day test interval. The CE analysis (CEN-327-A and CEN-327 Supplement 1) concludes that there would be a slight increase in RPS unavailability as a result of extending the test interval from monthly to quarterly. The analysis also concluded that reducing the test interval would reduce the scram and core melt frequency based on the expected reduction in test induced transients/scrams. The staff found these estimates to be acceptable. The staff SER for CEN-327-A found the overall impact of reduced testing intervals on safety to be negligible. The results of the CE analysis regarding reduced ESFAS testing on core melt frequency was found to similar to RPS.

The staff SER for CEN-327-A required the licensee to evaluate the effects of drift on the proposed functional test interval extension. The licensee reviewed the drift data (as left, as found) for the affected instrumentation and determined that the projected drift is bounded by the current setpoint calculations. The evaluation results are acceptable to the staff.

The functional units not specifically referenced in CEN-327-A for surveillance interval extension but proposed by the licensee to be included in the TS amendment (seismic high and containment cooling actuation signal) utilize similar bistables and/or design when compared to functional units analyzed by CEN-327-A. The staff finds the basis for including the additional functional units acceptable.

Based on the above, the staff finds the licensee proposal to incorporate the quarterly surveillance test intervals for RPS and ESFAS instrumentation as referenced by CEN-327-A and CEN-327-A Supplement 1 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 changes 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 (56 FR 49926). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Clifford K. Doutt
Lawrence E. Kokajko

Date: February 28, 1992