SEP 21 1984

Docket Nos. 50-361
and 50-362

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Gentlemen:

Subject: Issuance of Amendment No.25 to Facility Operating License NPF-10 and Amendment No. 14 to Facility Operating License NPF-15 San Onofre Nuclear Generating Station, Units 2 and 3

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 25 to Facility Operating License No. NPF-10 and Amendment No. 14 to Facility Operating License No. NPF-15 for the San Onofre Nuclear Generating Station, Units 2 and 3, located in San Diego County, California. The amendments modify the Technical Specifications to (1) change certain ESFAS response times, (2) temporarily suspend specification 3.0.4 to allow Unit 2 to be heated up prior to the hot setting of the pressurizer code safety valve, (3) reflect the installation of additional fire protection equipment for Unit 2 and (4) correct an error relating to the groups and individuals required to review revisions and modifications to the Monthly Operating Report, Offsite Dose Calculations Manual, Process Control Program and Major Changes to Radioactive Waste Treatment Systems by the Onsite Review Committee. These amendments were requested by your letters of December 1, 1982, January 6 and 25, April 15, August 1 and December 5, 1983 and correspond to your Proposed Change Numbers 36, 56, 72 and 79.

A copy of the Safety Evaluation supporting this amendment is also enclosed.

Sincerely,

/S/

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosures:
1. Amendment No. 25 to NPF-10
2. Amendment No. 14 to NPF-15
3. Safety Evaluation

cc w/enclosures: See next page

DL:LB#3	DL:LB#3	OELD	DL:LB#3	DL:AD/L
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6/29/84	6/21/84	6/0/84	6/19/84	9/1/84

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-10

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 2 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated December 1, 1982, January 6 and 25, April 15, August 1, and December 5, 1983 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-10 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. 25, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1984

SEP 21 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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

Amendment Pages	Overleaf Pages
3/4 3-28	3/4 3-27
3/4 3-29	-
3/4 3-30	-
3/4 3-58	3/4 3-57
3/4 3-60	3/4 5-59
3/4 3-61	
3/4 3-62	-
3/4 4-7	3/4 4-8
3/4 7-5	3/4 7-4
3/4 7-31	-
3/4 7-32	-
6-19	6-20
6-24	6-23
6-25	-
6-26	-

SAFETY EVALUATION

AMENDMENT NO. 25 TO NPF-10

AMENDMENT NO. 14 TO NPF-15

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

Introduction

Southern California Edison Company, on behalf of itself and the other licensees, San Diego Gas and Electric Company, the City of Riverside, California, and The City of Anaheim, California has submitted several applications for license amendments for San Onofre Nuclear Generating Station, Units 2 and 3. The evaluations of four such requests are presented below.

A. By letters dated December 1, 1982, January 25, 1983 and December 5, 1983, the Southern California Edison Company requested changes (Proposed Change Number 36 or PCN-36) in the following ESFAS response time Technical Specifications for San Onofre Nuclear Generating Station, Units 2 and 3:

1. Table 2.2-5, Item 3.a(1) Safety Injection
2. Table 3.3-5, Item 2.b, Containment Isolation Actuation Signal
3. Table 3.3-5, Item 5, Main Steam Isolation Signal
4. Table 3.3-5, Items 8 and 9, Emergency Feedwater Actuation Signal

B. By letter dated January 6, 1983, Southern California Edison Company requested a change to the San Onofre Unit 2 Technical Specifications to temporarily suspend Technical Specification 3.0.4 for up to 18 hours to allow the plant to be heated up prior to the hot setting of the pressurizer code safety valve (PCN-56).

C. By letter dated April 15, 1983, Southern California Edison Company requested a change to the San Onofre 2 Technical Specifications 3/4.3.3.7 Fire Detection Instrumentation and 3/4.7.8.3 Spray/and/or Sprinkler Systems to reflect the installation of additional fire protection equipment (PCN-72).

D. By letter dated August 1, 1983, Southern California Edison Company requested a change to the San Onofre Units 2 & 3 Technical Specifications 6.9.1.10, 6.13, 6.14 and 6.15 to correct an error relating to the groups and individuals required to review revisions and modifications to the Monthly Operating Report, Offsite Dose Calculations Manual, Process Control Program and Major Changes to Radioactive Waste Treatment Systems by the Onsite Review Committee (PCN-79).

Evaluation

A. ESFAS Response Times (PCN-36)

(1) Table 3.3-5, Item 2.a(1), Safety Injection, is revised to include an additional item: (c) Charging Pumps. The revised table also includes a response time of 31.2 seconds for the Charging Pumps. This item is being added because charging flow is required on pressurizer pressure low (only) to augment High Pressure Safety Injection (HPSI) flow for the small break LOCA. Because it is used to augment HPSI flow, the charging pump response time is the same as the response time for high pressure safety injection.

(2) Response time requirements for the main feedwater backup isolation valves (HV 1105, HV 1106, HV 4047, and HV 4051) are added to Table 3.3-5, Item 3.b, CIAS. The main feedwater backup isolation valves are required to isolate main feedwater in the event of a main steam or feedline break inside containment concurrent with a single failure of a main feedwater isolation valve (MFWIV). The response time requirement for the backup isolation valves is the same as that for the MFWIV's.

(3) Table 3.3-5, Item 5, Main Steam Isolation Signal (MSIS), is revised by the addition of response time requirements which apply to individual classes of valves actuated by a MSIS. Specifically added are Steam, Blowdown, Sample, and Drain Isolation Valves and Auxiliary Feedwater Isolation Valves. The response times listed correspond to those assumed in the accident analysis.

(4) The response time for Emergency Feedwater Actuation Signal (EFAS), Table 3.3-5 Items 8 and 9, are increased to the analyzed limits for auxiliary feedwater delivery. The allowed response time for non-LOCA events, bounded by the loss of normal feedwater event, is changed from 30.9 seconds for the steam/DC auxiliary feedwater train and 40.9 seconds for the AC train, to 42.7 seconds for each train. The response time for events which require AFW with SIAS is bounded by the coincident loss of normal AC power event at 53 seconds. For these cases the response time is changed from 50.9 to 52.7 seconds.

(5) An additional surveillance requirement is added to Specification 4.7.1.2.1.a which requires the licensee to verify that the AFW piping is full. This change is required to support the EFAS response time relaxation described in (4) above. The AFW lines are long enough that system transport time could result in unacceptable delivery time, if less than completely filled, even though the pumps and valves meet the revised response time requirements.

Items (1), (2), (3), and (5) above have been reviewed and found to be acceptable because they provide additional assurance that the accident analyses in the FSAR (upon which the staff's SER was based) are valid. Specifically, item (1) provides additional assurance that charging pump flow will be available within the time assumed in the accident analyses; item (2) provides additional assurance that main feedwater isolation will occur within the time assumed in the accident analyses; item (3) provides additional assurance that main steam isolation will occur within the time assumed in the accident analyses; and item (5) provides additional assurance that auxiliary feedwater delivery time will not exceed the time assumed in the accident analyses. Item (4) is a change in the allowed EFAS response times. These times are defined as the interval between the auxiliary feedwater system initiation signal (low steam generator water level) and the time that auxiliary feedwater reaches the steam generators(s). The revised response times are based upon the licensee's safety analyses in Chapter 15 of the San Onofre 2 and 3 FSAR. The limiting response times specified in the FSAR and the revised technical specification are 42.7 seconds during the loss of normal feedwater event and 52.7 seconds during the loss of normal A/C power event.

The revised limits restrict the response time of active AFW system components to values that do not exceed the values assumed in the FSAR accident analyses. The changes also require that the system remain filled to eliminate fluid transport time in order to ensure that overall auxiliary feedwater system response time is within the limits of existing safety analyses. In summary, we find the changes to be acceptable because they do not exceed the values assumed in the FSAR safety analyses which were previously reviewed and found to be acceptable, as described in the staff's Safety Evaluation Report on San Onofre 2 and 3 (NUREG-0712).

B. Temporary Suspension of Technical Specification 3.0.4 (PCN-56

The amendment changes Technical Specification 3.4.2 by the addition of a statement which allows the provisions of Technical Specification 3.0.4 to be temporarily suspended for up to 18 hours under certain operating conditions. As previously worded, Technical Specification 3.4.2 requires that all pressurizer code safety valves be operable with a lift setting of 2500 PSIA ± 1% when the plant is in Operating Modes 1, 2 and 3. The Technical Specification also requires that the lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperature and pressure. Further, Technical Specification 3.0.4 requires that the conditions of 3.4.2 for Modes 1, 2 and 3 must be met before entry into those modes. Thus, as previously worded, Technical Specifications 3.4.2 and 3.0.4 required the pressurizer code safety valves to be set at nominal operating temperatures of approximately 550°F, but do not allow the temperature to exceed 350°F (the upper limit of Mode 4) if the valve lift setting is not correct. However, if the plant is in Modes 4, 5, or 6, and the pressurizer code safety valve lift setting is not correct for any reason (such as repair or replacement), the Technical Specifications as previously written would prevent the plant from ever entering Mode 3 or above. To allow plant operation in this event the licensees have proposed that Technical Specification 3.4.2 be amended to allow the provisions of Technical Specification 3.0.4 to be suspended for up to 18 hours to allow the pressurizer code safety valves to be set under hot conditions, provided that a preliminary cold setting has been made prior to heatup. This amendment makes the requested change.

The above change is already in effect in the San Onofre Unit 3 Technical Specifications.

We find that the proposed temporary relief from the requirements of Technical Specification 3.0.4 is acceptable, because of the limited time involved (18 hours), and because setting the lift setpoint in the cold condition will approximate the hot setting. The change meets current staff criteria as included in the latest revision of the Standard Technical Specifications. Also, such relief has been approved by the NRC for other operating plants such as San Onofre Unit 3. Based on the above, we find the proposed change to be acceptable.

C. Fire Protection Equipment (PCN-72)

The proposed amendment would change Technical Specifications 3/4.3.3.7 FIRE DETECTION INSTRUMENTATION, and 3/4.7.8.2 SPRAY/AND/OR SPRINKLER SYSTEMS to reflect the installation of additional fire protection equipment in the plant: namely, (1) fire detectors to fire zones 11, 28, 45, 62, 72 and the Technical Support Center and (2) a deluge water spray system to the Auxiliary Feedwater Pump Room. These changes were implemented in accordance with commitments made as a result of License condition 2.C.(14)c of the San Onofre Nuclear Generating Station, Unit 2 Operating License.

The fire protection equipment covered by the proposed technical specification meets the staff's fire protection criteria and enhances the fire protection capability of the plant. The revised Technical Specifications represent an additional limitation, restriction or control on the facility. Therefore, the staff finds the proposed change to be acceptable.

D. Onsite Review Committee Review (PCN-79)

The proposed amendments would revise Technical Specifications 6.9.1.10, 6.13, 6.14 and 6.15 to correct an error relating to the review of revisions and modifications to the Monthly Operating Report (6.9.1.10), Offsite Dose Calculation Manual (6.14), Process Control Program (6.13) and Major Changes to Radioactive Waste Treatment Systems (6.15) by the Onsite Review Committee (OSRC). Specifically, references in these sections to review by the OSRC are replaced by references to review in accordance with Technical Specification 6.5.2. Technical review and control of activities at San Onofre 2 and 3 is normally implemented in accordance with Technical Specification 6.5.2. Reference to OSRC review is inconsistent with the provisons of Technical Specification 6.5.2 and was an administrative oversight in the initial issue of Technical Specification Administrative Controls. Reviews of changes to Technical Specifications 6.9.1.10, 6.13, 6.14 and 6.15 are performed by qualified individuals/organizations in accordance with Technical Specification 6.5.2.9 and are not a responsibility of the OSRC. In this instance, the proposed changes make the Technical Specifications more consistent throughout. Accordingly, we find the proposed changes to be acceptable.

Contact With State Official

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

Environmental Consideration

These amendments involve changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendments involve no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration and there has been no public comment on such findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 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 these amendments.

Conclusion

Based upon our evaluation of the proposed changes to the San Onofre Units 2 and 3 Technical Specifications, we have concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

Dated: September 21, 1984

DL:LB#3	DL:LB#3	DL:LB#3
JLee/yt	HRood	GWKnighton
9/17/84	9/12/84	9/14/84

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. NPF-15

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 3 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated December 1, 1982, January 25, 1983, August 1 and December 5, 1983 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

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

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

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"			10			
	Cable Tray Areas Elev 45'			9			
	Cable Tray Areas Elev 30'			4			
	Elevator Machinery Room			1			
	Combustible Oil Area Two steam generator rooms				32		
	Charcoal Filter Area Elev 45'	2					
2	Penetration Elev 63'6"			12			
4	New Fuel Storage Area and Spent Fuel Pool Areas						
	Spent Fuel Pool		4				
	New Fuel Pool		3				
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2	24		
	Cable Riser Gallery Rm 449			3	24		
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank Rms 511A		None				
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A	1		1			
	Emgy. A.C. Unit Rm 302-Train B	1		1			
10	Penetration Elev 45'			6			

*The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
11	S.E.B. Roof and Main Steam Relief Valves			2 (Note 1)			
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3	42		
	Cable Riser Gallery Rm 315			3	40		
13A	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 50'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50'						
	Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3		4	
	Train B			3		4	
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2	9 (Note 2)	6	
23	Fuel Handling Bldg Elev 30'						
	Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'	2			8 (Note 1)		

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
29	Control Building Elev 30'						
	Cable Riser Gallery Rm 236			3	51		
	Cable Riser Gallery Rm 224			3	52		
30	Electrical Tunnel Elev 30'6"			13	50		
31	Control Building Elev 30'			29			
32A	Control Building Elev 30'						
	Fan Room Rm 219 & Corridor Rm 221	2		1			
32B	Control Building Elev 30'						
	Fan Room Rm 233 & Corridor Rm 234	2		1			
34	Radwaste Elev 9' & 24'						
	Secondary Radwaste Tank Rms 126A,B & 127A,B	None					
35	Radwaste Elev 9' & 24'						
	Spent Resin Tank Rms 125A,B	None					
36	Fuel Handling Building Elev 17'6"						
	Spent Fuel Pool Pump Rm 107			2			
37	Radwaste Elev 24'						
	Letdown Heat Exchanger Rms 209A,B	None					
38	Radwaste Elev 24'						
	Letdown Control Valve Rms 218A,B	None					
39	Radwaste Elev 24'						
	Filter Crvd Tank Rm 216	None					
40	Radwaste Elev 9' & 24'						
	Primary Radwaste Tank Rms 211A,D	None					
41	Control Building Elev 9'						
	Cable Spreading Rm 111A			17	36		
	Cable Spreading Rm 111B			14	36		
42	Control Building Elev 9'						
	Cable Riser Gallery Rm 110			6	44		
	Cable Riser Gallery Rm 112			6	39		

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
43	Control Building Elev 9'						
	Emgy. Chiller Rm 115			2			
	Emgy. Chiller Rm 117			2			
44	Intake Structure						
	Pump Rm T2-106			4			
	Pump Rm T3-106			4			
45	Penetration Area Elev 9' & 15'						
	Piping Penetration Area 15'			6 (Note 1)			
48	Safety Equipment Building 9'						
	CCW HX and Piping Rm 022-025	None					
50	Radwaste Elev 9'						
	Charging Pump Rms 106A-F			6			
51	Radwaste Elev 9'						
	Boric Acid Makeup Tank Rms 105A-D	None					
53	Electrical Tunnel Elev 9'6", 11'6", (-) 2'6"			21	54		
54	Safety Eqpmt Bldg Elev 15'6" & 8'						
	Shutdown HX Rms 003, 004, 016, 018	None					
55	Safety Eqpmt Bldg Elev 8'						
	Chemical Storage Tank Rm 019			1			
56	Safety Eqpmt Bldg Elev 8'						
	Component Cooling Water Surge Tank Rms 020, 021	None					
57	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 005			1			
58	Radwaste Elev 37'						
	Reactor Trip System Rms 308A-D, 309-A-C			9			
59	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 001			1			

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
60	Safety Eqpmt Bldg Elev 15'6" Pump Rm 015			1			
61	Safety Eqpmt Bldg Elev 15'6" Component Cooling Water Pump Rms 006, 007, 008			3			
62	Radwaste Elev 50' Volume Control Valve Rooms			2 (Note 1)			
63	Control Building Elev 50' Corridor			12			
64	Control Building Elev 50' Vital Power Distribution Rms 310A-H			8			
65	Control Building Elev 50' Battery Rms 306B-J			8			
66	Control Building Elev 50' Evacuation Rm 311			1			
67	Radwaste Elev 63'6" Cable Riser Gallery Rm 506A Cable Riser Gallery Rm 506B			2 2	4 4		
68	Penetration 9' - 63'6" Cable Riser Shaft			1	21		
69	Safety Eqpmt Bldg Elev 5'3" Salt Water Cooling Piping Rm 010	None					
70	Radwaste Elev 24' Duct Shaft Rms 222A,B	None					
72	Control Building Elev 70' Corridor 401			4 (Note 1)			
75	Refueling Water Storage Tank T-005	None					
76	Refueling Water Storage Tank T-006	None					

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
78	Control Building Elev 9'						
	Corridor Rm 105			4			
79	Control Building Elev 50'						
	ESF Switchgear Rm 302A			2			
	ESF Switchgear Rm 302B			2			
80	Radwaste Elev 37' & 50'						
	Duct Shaft Rms	None					
81	Radwaste Elev 63'6"						
	Duct Shaft Rms 527A,B	None					
83	Salt Water Cooling Tunnel			6*			
84	Safety Eqpmt Bldg Elev 8'						
	HVAC Rm 017			3			
Technical Support Center (TSC) 5				1 (note 1)			

*3 in UNIT 2, 3 in UNIT 3

Notes

1. On completion of DCP 2-403E
2. On completion of DCP 2-122M

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 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. 14, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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

Amendment Pages	Overleaf Pages
3/4 3-28	3/4 3-27
3/4 3-29	3/4 3-30
3/4 7-5	3/4 7-6
6-20	6-19
6-25	-
6-26	-
6-27	-

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(c) Charging Pumps	31.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (NOTE 4b)	23.2*
(c) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (NOTE 4b)	23.0*
(c) Emergency Cooling Fans	21.0*
b. CIAS	
(1) Containment Isolation	10.9* (NOTE 2)
(2) Main Feedwater Backup Isolation (HV1105, HV1106, HV4047, HV4051)	10.9
(3) CCW Valves (Note 4a)	20.9
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump for entry into MODE 3.

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T-121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that the AFW piping is full of water by venting the accessible discharge piping high points.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an EFAS test signal.

2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted in accordance with 6.5.2.

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Exposure Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.#

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

    b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

    c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and acceptance pursuant to 6.5.2.

**Measurement made at 18" from source of radioactivity.

#The PCP shall be submitted and approved prior to shipment of "wet" solid radioactive waste.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and acceptance by the OSRC.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and solid)

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was performed pursuant to 6.5.2. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and acceptance pursuant to 6.5.2.

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted in accordance with 6.5.2.

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the NRC Regional Administrator, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents that exceed the limits of Specifications 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

l. Failure of one or more pressurizer safety valves.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRC and the NSG.

l. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following approved plant radiation protection procedures for entry into high radiation areas.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Exposure Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.#

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

    b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

    c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and acceptance pursuant to 6.5.2.

**Measurement made at 18" from source of radioactivity.
#The PCP shall be submitted and approved prior to shipment of "wet" solid radioactive waste.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and acceptance pursuant to 6.5.2.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and solid)

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was performed pursuant to 6.5.2. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and acceptance pursuant to 6.5.2.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(c) Charging Pumps	31.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (Note 4b)	23.2*
(c) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (Note 4b)	23.0*
(c) Emergency Cooling Fans	21.0*
b. CIAS	
(1) Containment Isolation	10.9* (NOTE 2)
(2) Main Feedwater Backup Isolation (HV1105, HV1106, HV4047, HV4051)	10.9
(3) CCW Valves (Note 4a)	20.9
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route a fire hose to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose and related valves shall be mounted above the hose and at the inoperable hose station.

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

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7-5

Safety Related Spray and/or Sprinkler Systems

Hazard	Location	No. of Systems	System Type
Reactor Coolant Pumps	Containment	4	Deluge-Water Spray
R.R. Tunnel	Fuel Hand. Bldg.	1	Wet Pipe
Truck Ramp	Radwaste Bldg.	1	Wet Pipe
Cable Tunnel	Section 1	1	Deluge-Water Spray
Cable Tunnel	Section 2	1	Deluge-Water Spray
Cable Tunnel	Section 3	1	Deluge-Water Spray
Cable Tunnel	Section 4	1	Deluge-Water Spray
Cable Tunnel	Section 5	1	Deluge-Water Spray
Cable Tunnel	Section 6	1	Deluge-Water Spray
Cable Tunnel	Section 7	1	Deluge-Water Spray
Cable Tunnel	Section 8	1	Deluge-Water Spray
Cable Tunnel	Section 9	1	Deluge-Water Spray
Cable Tunnel	Section 10	1	Deluge-Water Spray
Cable Tunnel Riser	Fuel Hand. Bldg.	1	Deluge-Water Spray
Cable Gallery	Radwaste Bldg.	2*	Deluge-Water Spray
Cable Risers El. 9 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 30 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 50 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 70 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Spreading Room	Control Bldg.	4*	Deluge-Water Spray
Emergency A.C. Unit - Train A	Fuel Handling Bldg.	1**	Deluge-Water Spray
Emergency A.C. Unit - Train B	Fuel Handling Bldg.	1**	Deluge-Water Spray
Diesel Generator	DG Building	2	Pre-action Sprinkler
HVAC Room 309A; Corridor 303	Control Bldg. 50'	1	Wet Pipe
Auxiliary Feedwater Pump Room	Tank Bldg. 30'	1 1#	Pre-action Sprinkler Deluge-Water Spray
Fan Room 233 and Corridor 234	Control Bldg. 30'	1	Wet Pipe
Salt Water Cooling Pumps and Salt Water Cooling Tunnel	Intake Structure	1	Wet Pipe
CCW Heat Exchangers and Piping Room; A/C Room 017	Safety Equipment Bldg.	1	Wet Pipe
Corridor 401	Control Bldg. 70'	1	Wet Pipe
Corridor 105	Control Bldg. 9'	1	Wet Pipe

*One half of these systems are designated Unit 3, but are required to be OPERABLE for Unit 2 operation.
**Charcoal filter deluge systems are manually actuated.
#On Completion of DCP 2-122M.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that the AFW piping is full of water by venting the accessible discharge piping high points.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an EFAS test signal.

2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

CONDENSATE STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CSTs) shall be OPERABLE with a contained volume of at least 144,000 gallons* in T-121 and 280,000 gallons in T-120.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tanks inoperable, within 4 hours either restore the CSTs to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits.

*Prior to first achieving 100% power, the minimum volume required to be contained in T-121 is that shown on Figure 3.7-1 corresponding to the maximum power level achieved to date.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 900 cubic feet and at least two groups of pressurizer heaters powered from the 1E busses, each having a capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually energizing the heaters.

4.4.3.3 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

3/4.4.2 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The provisions of Specification 3.0.4 may be suspended for one valve at a time for up to 18 hours for entry into and during operation in MODE 3 for the purpose of setting the pressurizer code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV5054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kV Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs.
4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.
4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.
5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to A.C. motor-operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-actuated valves except MSIVs, MFIVs, and CCW Valves 2HV-6211 and 2HV-6216.

4a. CCW noncritical loop isolation Valves 2HV-6212, 2HV-6213, 2HV-6218, and 2HV-6219 close.

4b. Containment emergency cooler CCW isolation Valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

Table 3.3-5 (Continued)

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(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (Steam/DC train)	30.9 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

ISSUANCE OF AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

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