July 17, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2 (TAC NO. M84516) AND UNIT NO. 3 (TAC NO. M84517)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. NPF-10 and Amendment No. 110 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated September 3, 1992.

These amendments will revise TS 3/4.4.8, "Pressure/Temperature Limits-Reactor Coolant System" and their associated Bases by removing the reactor vessel material surveillance capsule withdrawal schedules in accordance with the guidance in Generic Letter (GL) 91-01, "Removal of the Schedule for Withdrawal of Reactor Vessel Material Specimens from Technical Specifications." These amendments make editorial changes.

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

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

Docket Nos. 50-361
and 50-362

Enclosures:
1. Amendment No. 121 to NPF-10
2. Amendment No. 110 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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CGrimes, O11E22	ACRS (4), T2E26
Region IV	MFields
KPerkins, RIV/WCFO	TPolich
JStrosnider	

DOCUMENT NAME: SO84516.AMD

OFC	PDIV-2	PDIV-2	EMCB for KRW JRS	OGC NLO
NAME	EPeyton	MFields:ye	JStrosnider	MZOBLER
DATE	7/6/95	7/15/95	7/7/95	7/11/95

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July 17, 19

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Executive Vice President
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TPolich

DOCUMENT NAME: SO84516.AMD

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DATE	7/6/95	7/6/95	7/7/95	7/11/95

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

July 17, 1995

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

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Mel B Fields

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

Docket Nos. 50-361
and 50-362

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3. Safety Evaluation

cc w/encls: See next page

cc w/encls:
Mr. R. W. Krieger, Vice President
Southern California Edison Compan
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, California 92674-0128

Chairman, Board of Supervisors
County of San Diego
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c/o U.S. Nuclear Regulatory Commission
Post Office Box 4329
San Clemente, California 92674

Mayor
City of San Clemente
100 Avenida Presidio
San Clemente, California 92672

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. NPF-10

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

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated September 3, 1992, 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. 121, 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

Mel B. Fields

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 17, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 121 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 Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
XIX	XIX
3/4 4-27a	--
3/4 4-28	3/4 4-28
B 3/4 4-7	B 3/4 4-7

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LIST OF TABLES

LIST OF TABLES

REACTOR COOLANT SYSTEM

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 and.Figure 3.4-3 during heatup, cooldown, criticality, boltup, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 10°F in any one hour period with RC cold leg temperature less than 112°F. A maximum heatup of 30°F in any one hour period with RC cold leg temperature less than 163°F. A maximum heat-up of 60°F in any one hour period with RC cold leg temperature greater than 163°F.

b. A maximum cooldown of 10°F in any one hour period with RC cold leg temperatures less than 103°F. A maximum cooldown of 30°F in any one hour period with RC cold leg temperatures less than 145°F. A maximum cooldown of 100°F in any one hour period with RC temperature greater than 145°F.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

d. A minimum temperature of 86°F to tension reactor vessel head bolts.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY with-in the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. Recalculate the Adjusted Reference Temperature based on the greater of the following:

a. The mean value of shift in reference temperature for plate C-6404-3*, or

b. The predicted shift in reference temperature for weld seams 3-203A or 3-203B as determined by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

*The most limiting material in the reactor vessel in accordance with the new Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988, has changed, and is plate C-6404-3. Calculative procedures provided in the new guide should be used to obtain the mean values of shift in RT_{NDT} of plate C-6404-3. Calculations are based on the actual shift in reference temperature as determined by impact testing on the existing plate C-6404-2 surveillance material.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F/hr or cooldown rate of up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period, and they include adjustments for possible errors in the pressure and temperature sensing instruments.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than I Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper and nickel content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is maintained in the FSAR. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel taking into account the location of the sample closer to the core than the vessel wall by means of the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
215-01	C-6403-1	A533GRBCL-1	Upper Shell Plate	40	15	35	130
215-01	C-6403-2	"	"	0	20	25	133
215-01	C-6403-3	"	"	-10	20	45	131
215-03	C-6404-1	"	Intermediate Shell Plate	-30	10	50	145
215-03	C-6404-2	"	"	-20	20	50	155
215-03	C-6404-3	"	"	-20	10	50	131
215-02	C-6404-4	"	Lower Shell Plate	-10	-5	25	124
215-02	C-6404-5	"	"	-20	10	25	134
215-02	C-6404-6	"	"	-10	-20	0	151
238-02	C-6401	A508C1-2	Vessel Flange Forging	-10	-70	-35	148
209-02	C-6402	"	Closure Head Flange Forging	-10	-90	-40	142
205-02	C-6410-1	"	Inlet Nozzle Forging	20	-40	-35	130
205-02	C-6410-2	"	"	0	-20	-5	135
205-02	C-6410-3	"	"	0	-15	-15	140
205-02	C-6410-4	"	"	0	-65	-50	140
205-06	C-6411-1	"	Outlet Nozzle Forging	-100	-30	-10	140
205-06	C6411-2	"	"	0	-35	-10	140
232-01	C-6424	A533GRBCL-1	Bottom Head Torus	-50	-20	10	122
232-02	C-6425	"	Bottom Head Dome	-50	-30	-20	136
205-03	C-6428-1	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174
205-03	C-6428-2	"	"	-30	-70	-50	174
205-03	C-6428-3	"	"	-30	-70	-50	174
205-03	C-6428-4	"	"	-30	-70	-50	174
205-07	C-6429-1	"	Outlet Nozzle Ext. Forging	-30	-40	-25	229
205-07	C-6429-1	"	"	-30	-40	-25	229
231-02	C-6430-1	A533GRBCL-1	Closure Head Peels	+10	20	55	118
231-02	C-6431-1	"	"	-20	10	50	100
231-02	C-6432-1	"	"	-10	-15	45	115
231-02	C-6432	"	Closure Head Dome	-10	-15	45	115

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-15

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

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated September 3, 1992, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 110, 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

Mel B. Fields

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 17, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 110 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
XIX	XIX
3/4 4-28a	--
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LIST OF TABLES

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4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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a. The actual shift in reference temperature for plate C-6802-1 as determined by impact testing, or

b. The predicted shift in reference temperature for weld seams 2-203A, 2-203B, or 2-203C as determined by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

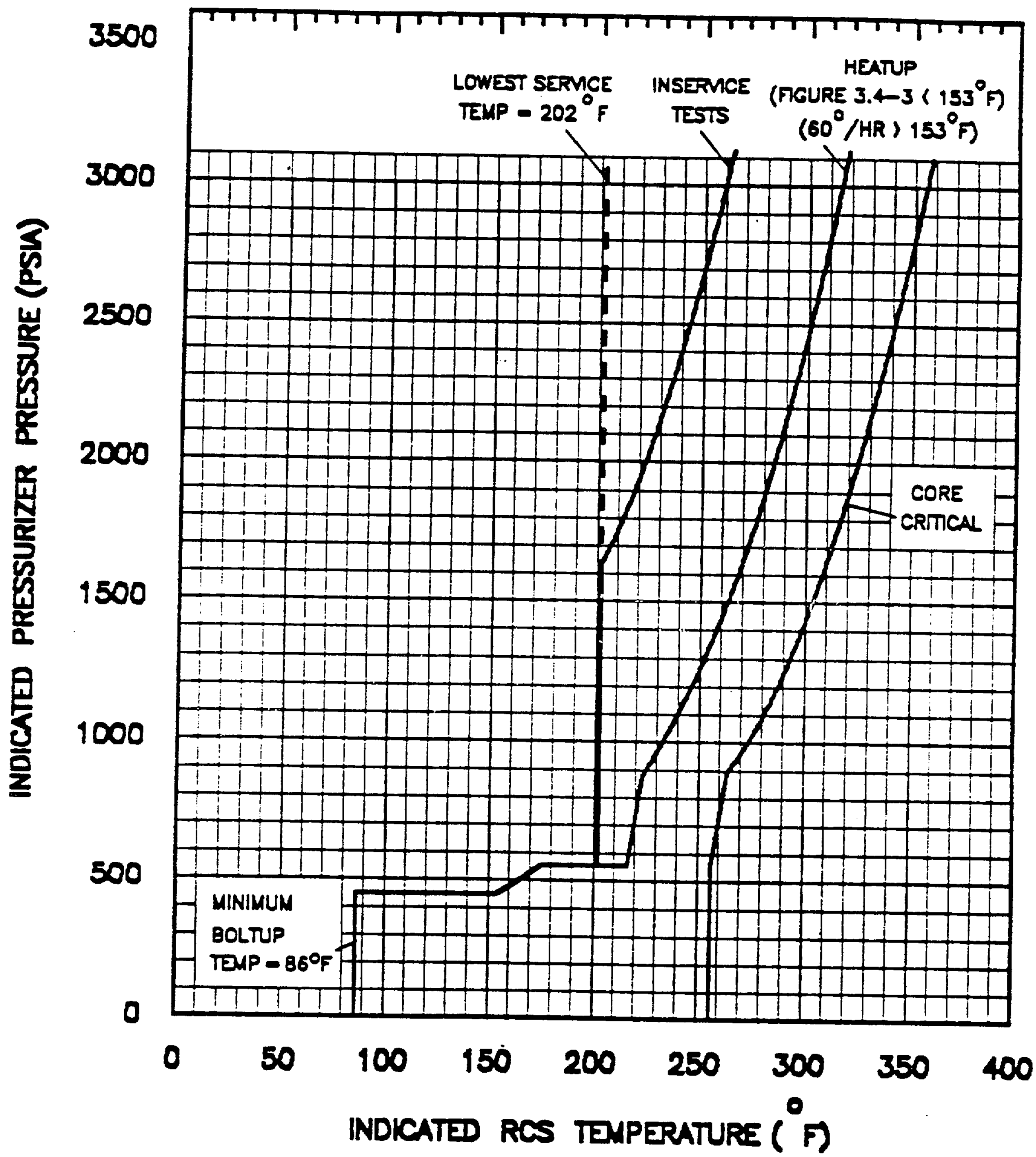

FIGURE 3.4-2

SONGS 3 RCS PRESSURE/TEMPERATURE
LIMITATION FOR 4-8 EFPY

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F/hr or cooldown rate of up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3.

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The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

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TABLE B 3/4.4-1

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215-01	C-6801-1	A533GRBCL1	Upper Shell Plate	-20	28	64	115
215-01	C-6801-2	A533GRBCL1	Upper Shell Plate	-20	-6	34	106
215-01	C-6801-3	A533GRBCL1	Upper Shell Plate	-20	18	36	115
215-02	C-6802-4	A533GRBCL1	Lower Shell Plate	-30	32	62	115
215-02	C-6802-5	A533GRBCL1	Lower Shell Plate	0	36	64	110
215-02	C-6802-6	A533GRBCL1	Lower Shell Plate	-40	32	100	90
215-03	C-6802-1	A533GRBCL1	Intermediate Shell	-20	56	100	95
215-03	C-6802-2	A533GRBCL1	Intermediate Shell	-20	40	66	113
215-03	C-6802-3	A533GRBCL1	Intermediate Shell	-10	44	80	101
203-02	C-6823	A508CL2	Vessel Flange Forging	0	-30	-15	NA
209-02	C-6824-1	A508CL2	Closure Head Flange Forging	-40	-100	-100	NA
205-02	C-6829-1	A508CL2	Inlet Nozzle Forging	10	-35	-5	109
205-02	C-6829-2	A508CL2	Inlet Nozzle Forging	0	-55	-35	156
205-02	C-6829-3	A508CL2	Inlet Nozzle Forging	10	-25	35	112
205-02	C-6829-4	A508CL2	Inlet Nozzle Forging	10	-30	25	108
205-06	C-6830-1	A508CL2	Outlet Nozzle Forging	-10	-30	-15	125
205-06	C-6830-2	A508CL2	Outlet Nozzle Forging	-10	-20	-5	131
232-01	C-6840-1	A533GRBCL1	Bottom Head Torus	-50	-10	0	107
232-02	C-6841-1	A533GRBCL1	Bottom Head Dome	-40	10	20	99

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated September 3, 1992, 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 3/4.4.8, "Pressure/Temperature Limits-Reactor Coolant System," and their associated Bases by removing the withdrawal schedules for the reactor vessel material irradiation surveillance specimens in response to guidance provided in Generic Letter 91-01, "Removal of the Schedule for Withdrawal of Reactor Vessel Material Specimens from Technical Specifications."

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of technical specifications are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 Fed. Reg. 39132 (July 22, 1993), in which the Commission indicated that compliance with the final policy statement satisfies §182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the final policy statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1] As a result, existing TS requirements which fall within or satisfy any of the criteria in the final policy statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

3.0 EVALUATION

TS 3/4.4.8, "Pressure/Temperature Limits - Reactor Coolant System," contains a limiting condition for operation for the reactor coolant system (RCS) that limits the rate of change in temperature and pressure to values consistent with the fracture toughness requirements of the American Society of Mechanical Engineers Code and Appendix G to Part 50 of Title 10 of the Code of Federal Regulations. Changes in the values of these limits are necessary because the fracture toughness properties of ferritic materials in the reactor vessel change as a function of the reactor operating time.

For this reason, the present TS include Surveillance Requirement (SR) 4.4.8.1.2, which requires the removal and examination of the irradiated specimens of reactor vessel material. The licensee examines the specimens to

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1 The Commission recently promulgated a proposed change to 10 CFR 50.36, pursuant to which the rule would be amended to codify and incorporate these criteria (59 FR 48180). The Commission's Final Policy Statement specified that only limiting conditions for reactor Core Isolation Cooling, Isolation Condenser, Residual Heat Removal, Standby Liquid Control, and Recirculation Pump Trip meet the guidance for inclusion in the TS under Criterion 4 (58 FR at 39137). The Commission has solicited public comments on the scope of Criterion 4, in the pending rulemaking.

determine the changes in material properties in accordance with the requirements of Appendix H to 10 CFR Part 50. TS Table 4.4-5 specifies the schedule for removal of each capsule. The removal of the schedule for withdrawing material specimens from the TS will eliminate the necessity of a license amendment to make changes to this schedule.

The staff's review of the proposed change determined that the relocation of the withdrawal schedules does not eliminate the requirements for the licensee to ensure that the reactor vessel is capable of performing its safety function. Section I.B.3 of Appendix H to 10 CFR Part 50 requires the submittal of a proposed withdrawal schedule for material specimens to the NRC and approval by the NRC before implementation. Hence, adequate regulatory controls exist to control changes to this schedule without the necessity of subjecting it to the license amendment process by including it in the TS.

The licensee has committed to include this withdrawal schedule in the next revision of the Updated Final Safety Analysis Report (UFSAR). Although the withdrawal schedules are relocated from the technical specifications to the UFSAR, the licensee must continue to evaluate any changes to the schedule in accordance with 10 CFR 50.59 and Appendix H to 10 CFR Part 50. Should the licensee's determination conclude that an unreviewed safety question is involved, due to either (1) an increase in the probability or consequences of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to UFSAR commitments and to take any remedial action that may be appropriate.

The staff's review concluded that 10 CFR 50.36 does not require the withdrawal schedule to be retained in technical specifications. Requirements related to the operability, applicability, and surveillance requirements, including performance of testing to ensure operability of the reactor vessel is retained due to the reactor vessel's importance in mitigating the consequences of an accident. However, the staff determined that the inclusion of the withdrawal schedules are an operational detail related to the licensee's safety analyses which are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions of the affected withdrawal schedules, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety.

The staff has concluded, therefore, that relocation of the withdrawal schedules for the reactor vessel material irradiation surveillance specimens is acceptable because (1) Appendix H to 10 CFR Part 50 provides adequate regulatory guidance for this issue, (2) their inclusion in technical specifications is not specifically required by 10 CFR 50.36 or other regulations, (3) the withdrawal schedules have been relocated to the UFSAR, are adequately controlled by 10 CFR 50.59, and their inclusion in the TS is not required to avert an immediate threat to the public health and safety, and (4) changes that are deemed to involve an unreviewed safety question, will require prior NRC approval in accordance with 10 CFR 50.59(c).

The following editorial corrections are being made on the affected TS pages:

1. Unit 2 TS page 3/4 4-27a is being renumbered 3/4 4-28 with the deletion of Table 4.4.5 on 3/4 4-28.

2. Unit 2 TS section 4.4.8.1.2.a and footnote - "plates" should be "plate."

3. Unit 3 TS page 3/4 4-28a is being renumbered 3/4 4-29 with the deletion of Table 4.4.5 on page 3/4 4-29.

These changes are strictly administrative in nature and are acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 8781). 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.

5.0 CONCLUSION

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

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