July 18, 1995

Mr. Harold B. Ray
Executive Vice President
Southern California Edison Co.
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, California 92674-0128

SUBJECT: ISSUANCE OF AMENDMENT FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2 (TAC NO. M86692) AND UNIT NO. 3 (TAC NO. M84502)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 122 to Facility Operating License No. NPF-10 and Amendment No. 111 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 separate applications for each unit, both dated April 30, 1993, as supplemented for each unit, both dated July 6, 1994, and an additional supplemental letter dated January 27, 1995.

These amendments revise TS 3/4.4.8.1, "Pressure-Temperature Limits," TS 3.4.8.3.1, "Overpressure Protection Systems-RCS Temperature ≤256°F [for Unit 2, ≤246°F for Unit 3]," TS 3.4.8.3.2, "Overpressure Protection Systems-RCS Temperature >256°F [for Unit 2, >246°F for Unit 3]," and the associated TS Bases. The proposed change (1) revises the reactor coolant system (RCS) pressure-temperature (P-T) limits and the low temperature overpressure protection (LTOP) enable temperatures to be effective until 20 effective full power years (EFPY) of operation, and (2) makes minor 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. 122 to NPF-10
2. Amendment No. 111 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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

July 18, 1995

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Executive Vice President
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San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, California 92674-0128

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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. 122 to NPF-10
2. Amendment No. 111 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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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. 122
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 April 30, 1993, as supplemented by letters dated July 6, 1994, and January 27, 1995, 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.

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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. 122, 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 to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mel B. Fields

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 18, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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

REMOVE	INSERT
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--	XXII
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3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
--	3/4 4-29a
3/4 4-30	3/4 4-30
--	3/4 4-30a
--	3/4 4-30b
--	3/4 4-30c
3/4 4-30a	3/4 4-30d
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
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--	B 3/4 4-7a
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3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 With the reactor vessel head bolts tensioned*, the Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-4, 3.4-5, 3.4-6, and 3.4-7 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any 1-hour period with RCS cold leg temperature greater than or equal to 86°F.

b. A maximum cooldown as specified by Figure 3.4-5 in any 1-hour period with RCS cold leg temperature less than or equal to 160°F. A maximum cooldown of 100°F in any 1-hour period with RCS cold leg temperature greater than 160°F.

c. A maximum temperature change of 10°F in any 1-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.

With the reactor vessel head bolts detensioned, the Reactor Coolant System (except the pressurizer) temperature shall be limited to a maximum heatup or cooldown of 60°F in any 1-hour period.

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 within 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.

*With the reactor vessel head bolts detensioned, RCS cold leg temperature may be less than 86°F.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

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.

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 3.4-4 through 3.4-7. Recalculate the Adjusted Reference Temperature in accordance with Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

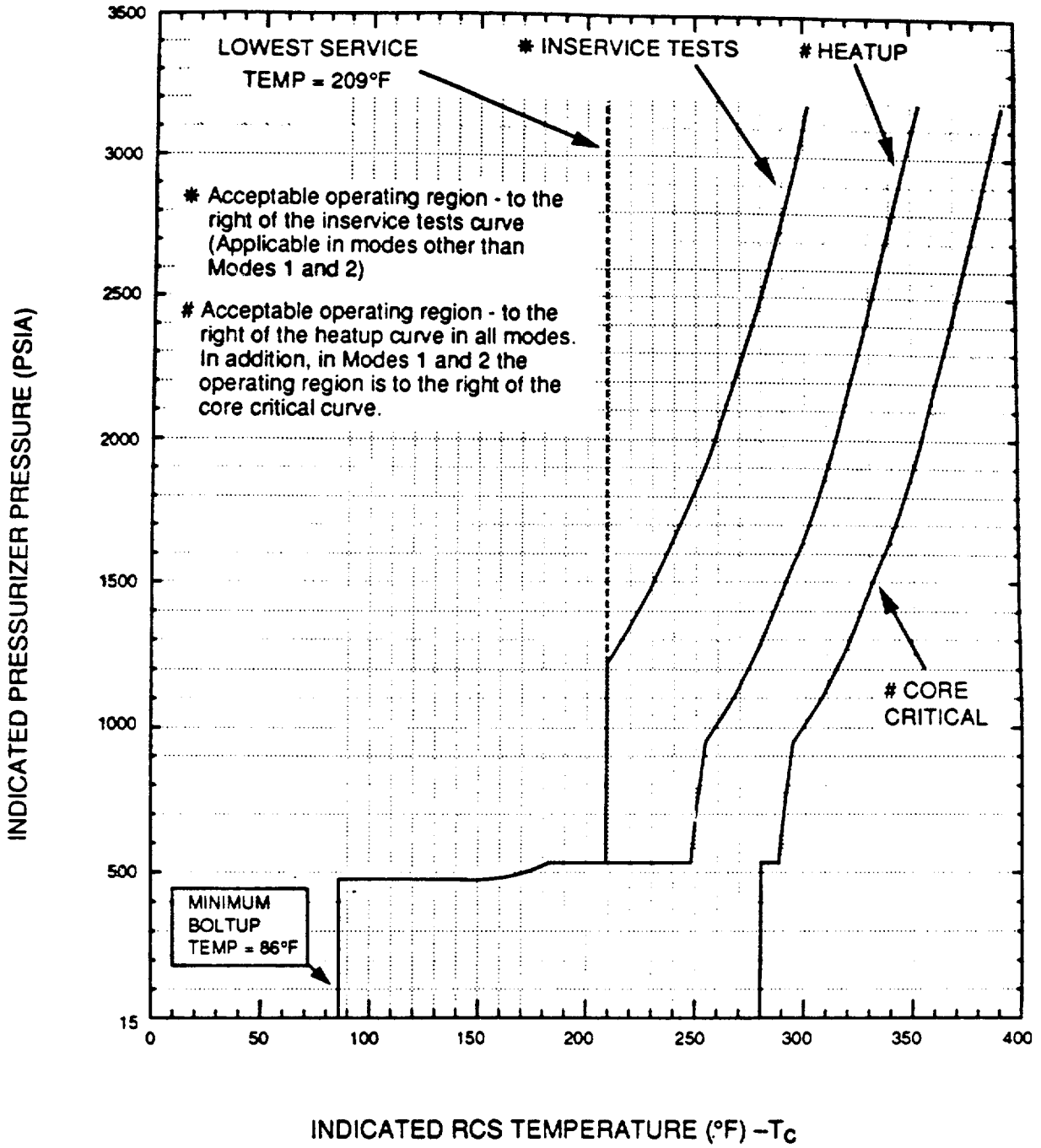


Figure 3.4-2

SONGS 2 HEATUP RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Normal Operation

(Figure 3.4-3 - Not Used)

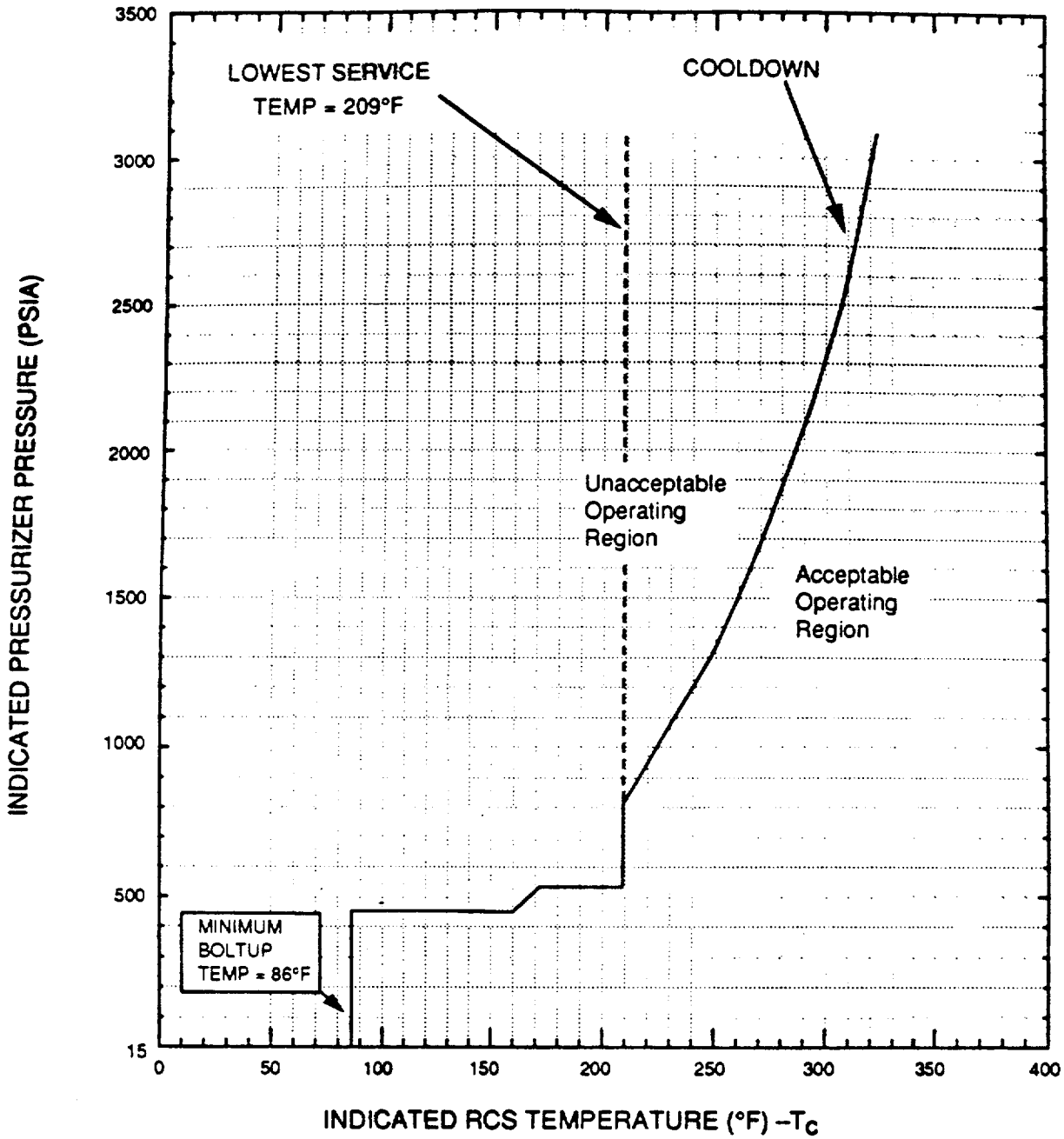
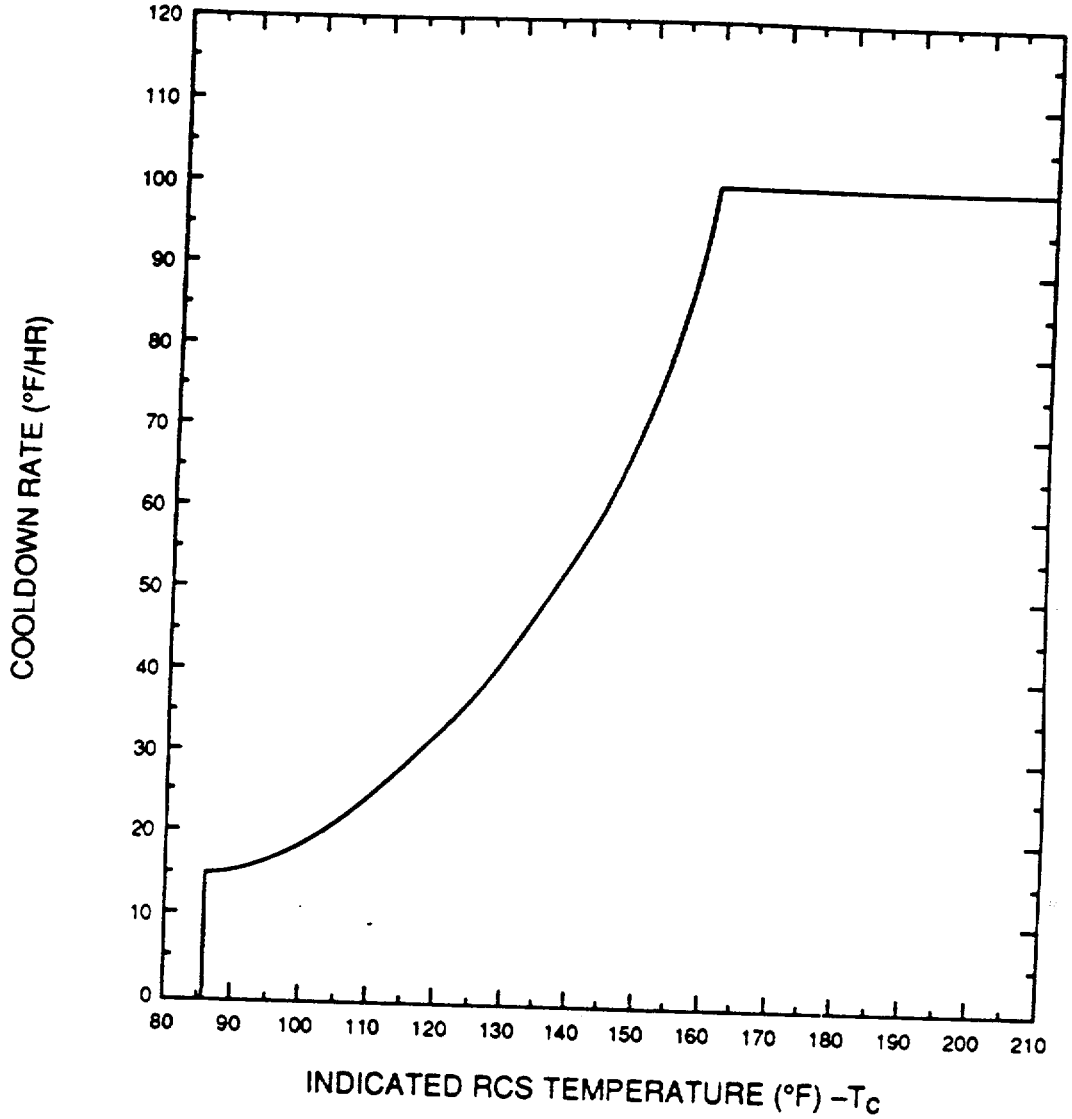


FIGURE 3.4-4

SONGS 2 COOLDOWN RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Normal Operation



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 160°F

FIGURE 3.4-5

SONGS 2 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (UNTIL 20 EFPY)
Normal Operation

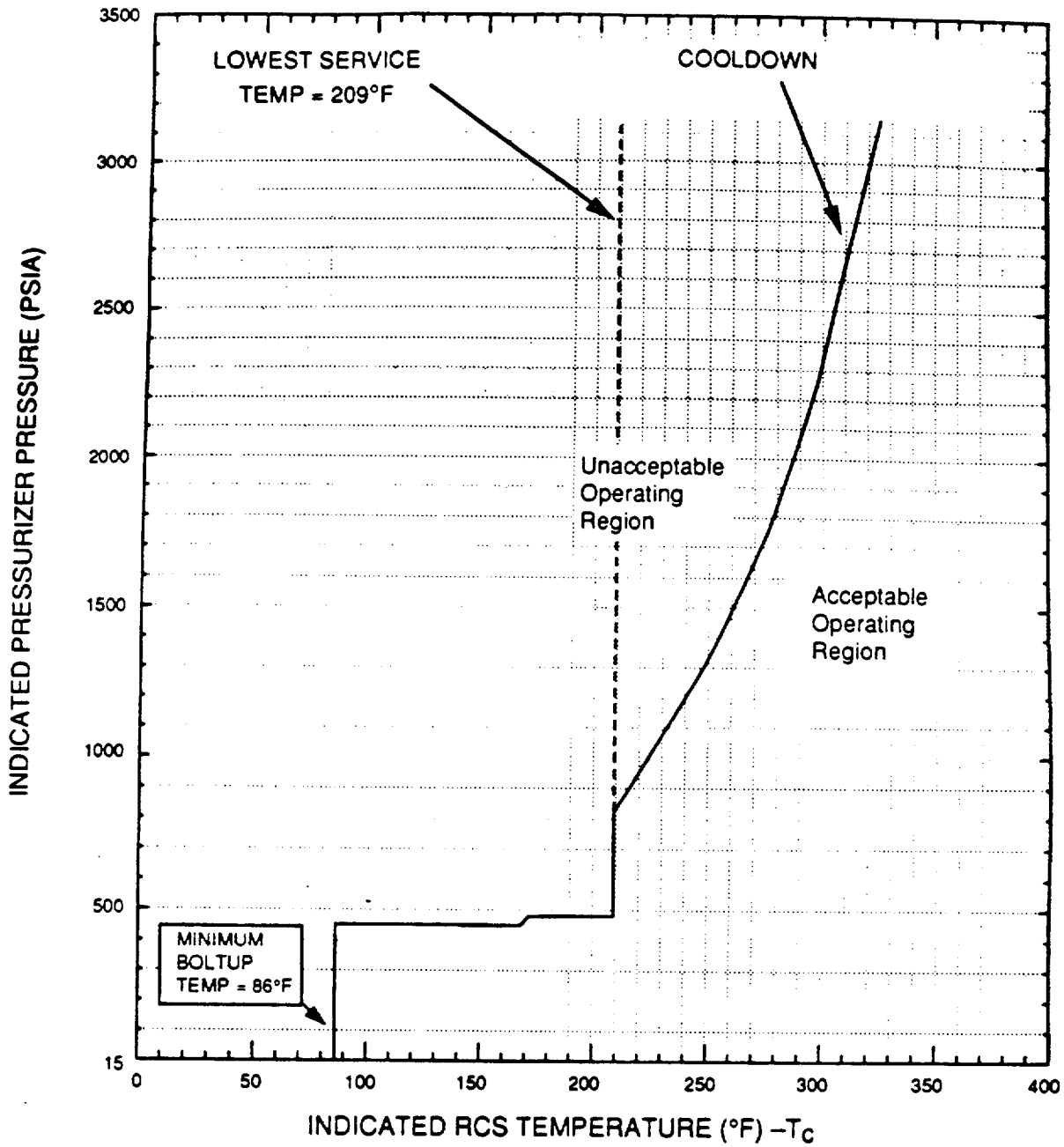
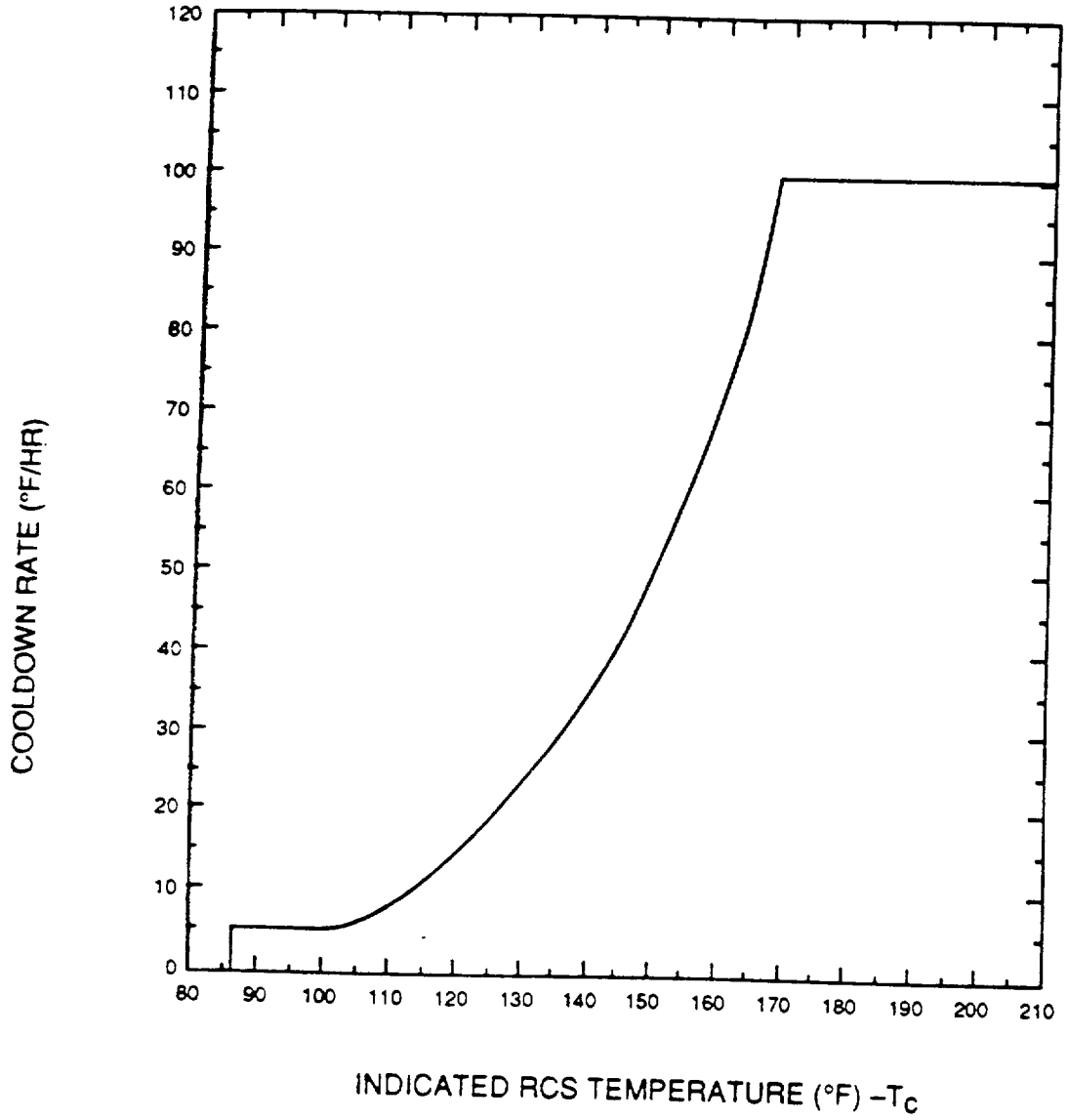


FIGURE 3.4-6

SONGS 2 COOLDOWN RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Remote Shutdown Operation



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 168°F

FIGURE 3.4-7

SONGS 2 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (UNTIL 20 EFPY)
Remote Shutdown Operation

Table 3.4-3

Low Temperature RCS Overpressure Protection Range

Operating Period, EFPY	Cold Leg Temperature, °F	
	During Heatup	During Cooldown
Until 20 (Normal Operation)	≤ 256	≤ 238
Until 20 (Remote Shutdown Operation)	*	≤ 238

*Heatup operations are not normally performed from the Remote Shutdown panels.

PRESSURIZER - HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 for each cycle of main spray when less than 4 reactor coolant pumps are operating and for each cycle of auxiliary spray operation.

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE ≤ 256°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 No more than two high-pressure safety injection pumps shall be OPERABLE and at least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

1) A lift setting of 406 ± 10 psig*, and

2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open

or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to the enable temperatures specified in Table 3.4-3; MODE 5; and MODE 6 when the head is on the reactor vessel and the RCS is not vented.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed, open the closed valve(s) or power-lock open the OPERABLE SDCS Relief Valve isolation valve pair within 24 hours, or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. With more than two high-pressure safety injection pumps OPERABLE, secure the third high-pressure safety injection pump by racking out its motor circuit breaker or locking close its discharge valve within 8 hours.

*The lift setting pressure applicable to valve temperatures of less than or equal to 130°F.

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > 256°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.2 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

1) A lift setting of 406 ± 10 psig*, and

2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open

or,

b. A minimum of one pressurizer code safety valve with a lift setting of 2500 psia ± 1%**.

APPLICABILITY: MODE 4 with RCS temperature above that specified in Table 3.4-3.

ACTION:

a. With no safety or relief valve OPERABLE, be in COLD SHUTDOWN and vent the RCS through a g reater than or equal to 5.6 square inch vent within the next 8 hours.

b. In the event the SDCS Relief Valve is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve code safety valve on the transient and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.

b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.2.2 The pressurizer code safety valve has no additional surveillance requirements other than those required by Specification 4.0.5.

---

* The lift setting pressure applicable to valve temperatures of less than or equal to 130°F.

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

3.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves for normal operation (Figures 3.4-2 and 3.4-4) and the cooldown limit curve for remote shutdown operation (Figure 3.4-6) 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 limit curves for Remote Shutdown operation are determined using the Total Loop Uncertainties (TLUs) for temperature and pressure for the Remote Shutdown Panel instruments in which the pressure TLUs are higher than those for the Control Room shutdown instruments. 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 instrument uncertainties, and static and dynamic heads.

The reactor vessel materials were tested prior to reactor startup to determine their initial RT_{NDT}; the results of these tests and the updates resulting from the evaluation of material properties in response to Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1 are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 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 limit curve (Figure 3.4-2) and the cooldown limit curves, Figures 3.4-4 and 3.4-6, include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for instrument uncertainties, and static and dynamic heads.

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 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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, 3.4-4, and 3.4-6 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.

The Low Temperature Overpressure Protection (LTOP) enable temperatures are based upon the recommendations of NUREG-0800 Branch Technical Position (BTP) RSB 5-2, Revision 1, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." BTP RSB 5-2, Revision 1 defines the enable temperature as "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations."

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	A533GRBCL-1	Upper Shell Plate	0	20	25	133
215-01	C-6403-3	A533GRBCL-1	Upper Shell Plate	-10	20	45	131
215-03	C-6404-1	A533GRBCL-1	Intermed. Shell Plate	-30	40	80	119
215-03	C-6404-2	A533GRBCL-1	Intermed. Shell Plate	-20	70	80	113
215-03	C-6404-3	A533GRBCL-1	Intermed. Shell Plate	-20	70	80	99
215-02	C-6404-4	A533GRBCL-1	Lower Shell Plate	-10	-40	80	104
215-02	C-6404-5	A533GRBCL-1	Lower Shell Plate	-20	50	70	118
215-02	C-6404-6	A533GRBCL-1	Lower Shell Plate	-10	50	50	124
238-02	C-6401	A508C1-2	Vessel Flange Forging	-10	-70	-35	148
209-02	C-6402	A508C1-2	Closure Head Flange Forging	-10	-90	-40	142
205-02	C-6410-1	A508C1-2	Inlet Nozzle Forging	20	-40	-35	130
205-02	C-6410-2	A508C1-2	Inlet Nozzle Forging	0	-20	-5	135
205-02	C-6410-3	A508C1-2	Inlet Nozzle Forging	0	-15	-15	140
205-02	C-6410-4	A508C1-2	Inelt Nozzle Forging	0	-65	-50	140
205-06	C-6411-1	A508C1-2	Outlet Nozzle Forging	-100	-30	-10	140
205-06	C-6411-2	A508C1-2	Outlet Nozzle Forging	0	-35	-10	140
232-01	C-6424	A533GRBCL-1	Bottom Head Torus	-50	-20	10	122
232-02	C-6425	A533GRBCL-1	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	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174
205-03	C-6428-3	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174
205-03	C-6428-4	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174

TABLE B 3/4.4-1 (Continued)

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
205-07	C-6429-1	A508CL-1	Outlet Nozzle Ext. Forging	-30	-40	-25	229
205-07	C-6429-1	A508CL-1	Outlet Nozzle Ext. Forging	-30	-40	-25	229
231-02	C-6430-1	A533GRBCL-1	Closure Head Peels	+10	20	55	118
231-02	C-6431-1	A533GRBCL-1	Closure Head Peels	-20	10	50	100
231-02	C-6432-1	A533GRBCL-1	Closure Head Peels	-10	-15	45	115
231-02	C-6432	A533GRBCL-1	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. 111
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 April 30, 1993, as supplemented by letters dated July 6, 1994, and January 27, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 111, 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 to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mel B. Fields

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 18, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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

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XVIIa	XVIIa
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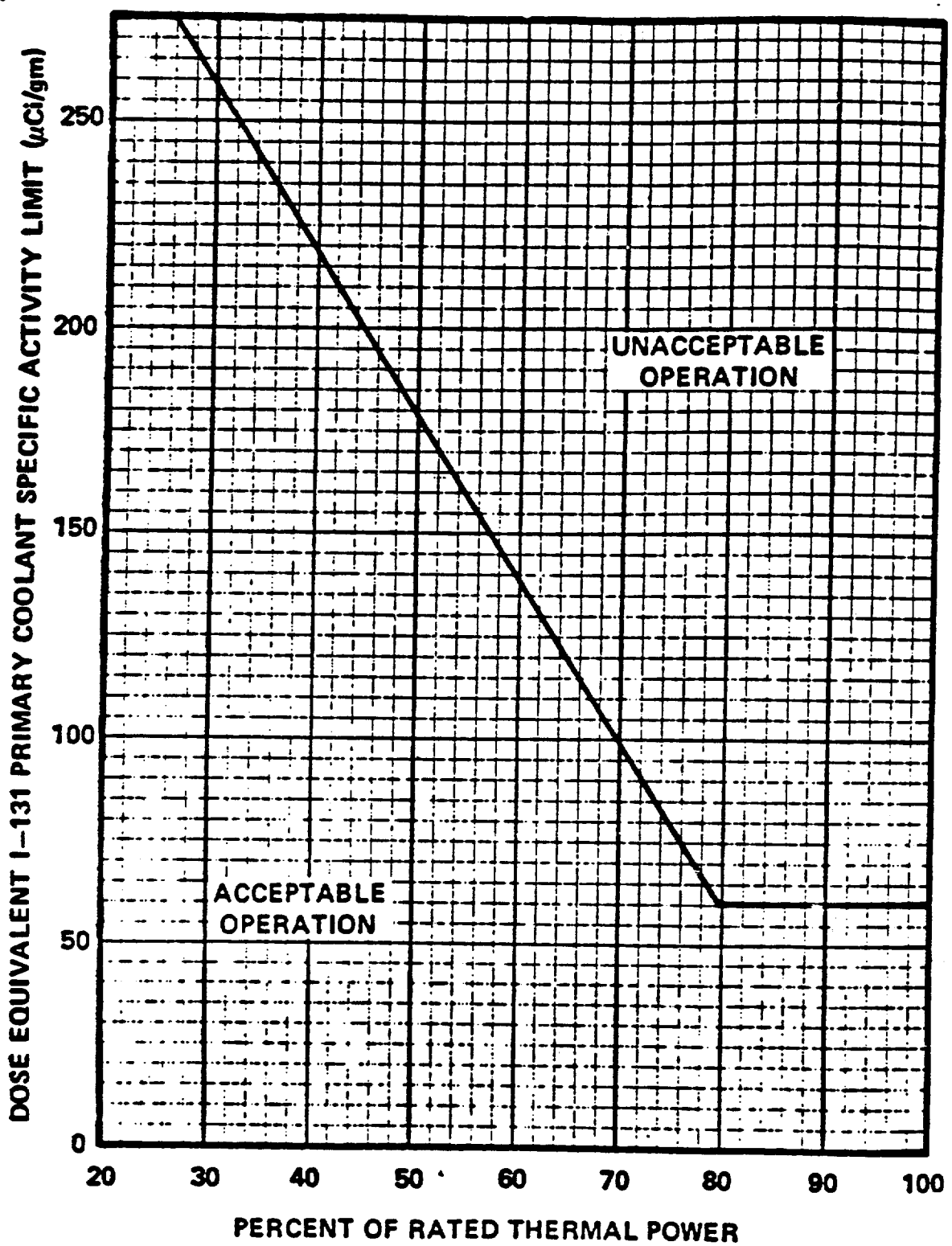


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 With the reactor vessel head bolts tensioned*, the Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-4, 3.4-5, 3.4-6, and 3.4-7 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any 1-hour period with RCS cold leg temperature greater than or equal to 86°F.

b. A maximum cooldown as specified by Figure 3.4-5 in any 1-hour period with RCS cold leg temperature less than or equal to 147°F. A maximum cooldown of 100°F in any 1-hour period with RCS cold leg temperature greater than 147°F.

c. A maximum temperature change of 10°F in any 1-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.

With the reactor vessel head bolts detensioned, the Reactor Coolant System (except the pressurizer) temperature shall be limited to a maximum heatup or cooldown of 60°F in any 1-hour period.

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 within 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.

*With the reactor vessel head bolts detensioned, RCS cold leg temperature may be less than 86°F.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

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.

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 3.4-4 through 3.4-7. Recalculate the Adjusted Reference Temperature in accordance with Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

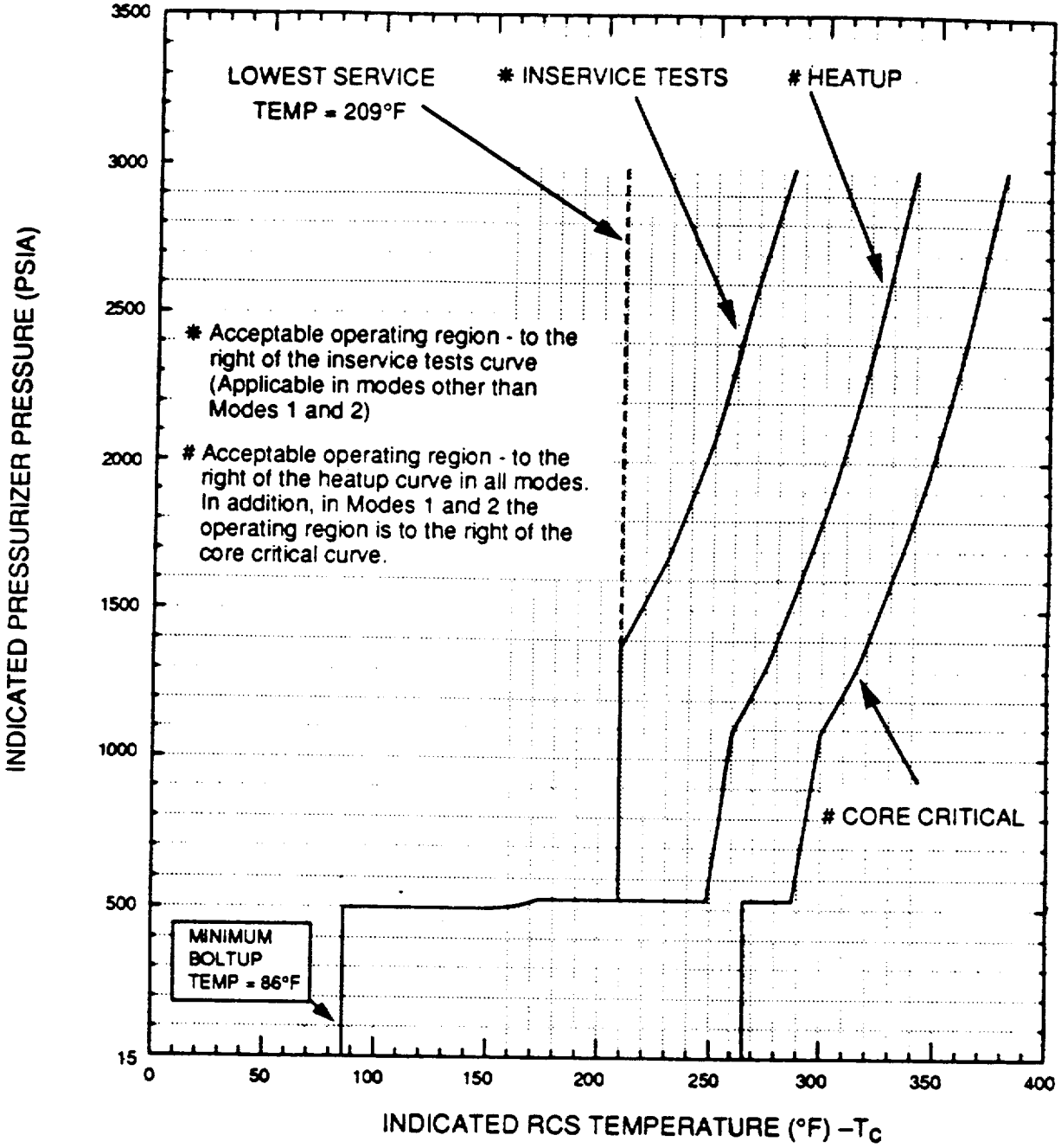


FIGURE 3.4-2

SONGS 3 HEATUP RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Normal Operation

(Figure 3.4-3 - DELETED)

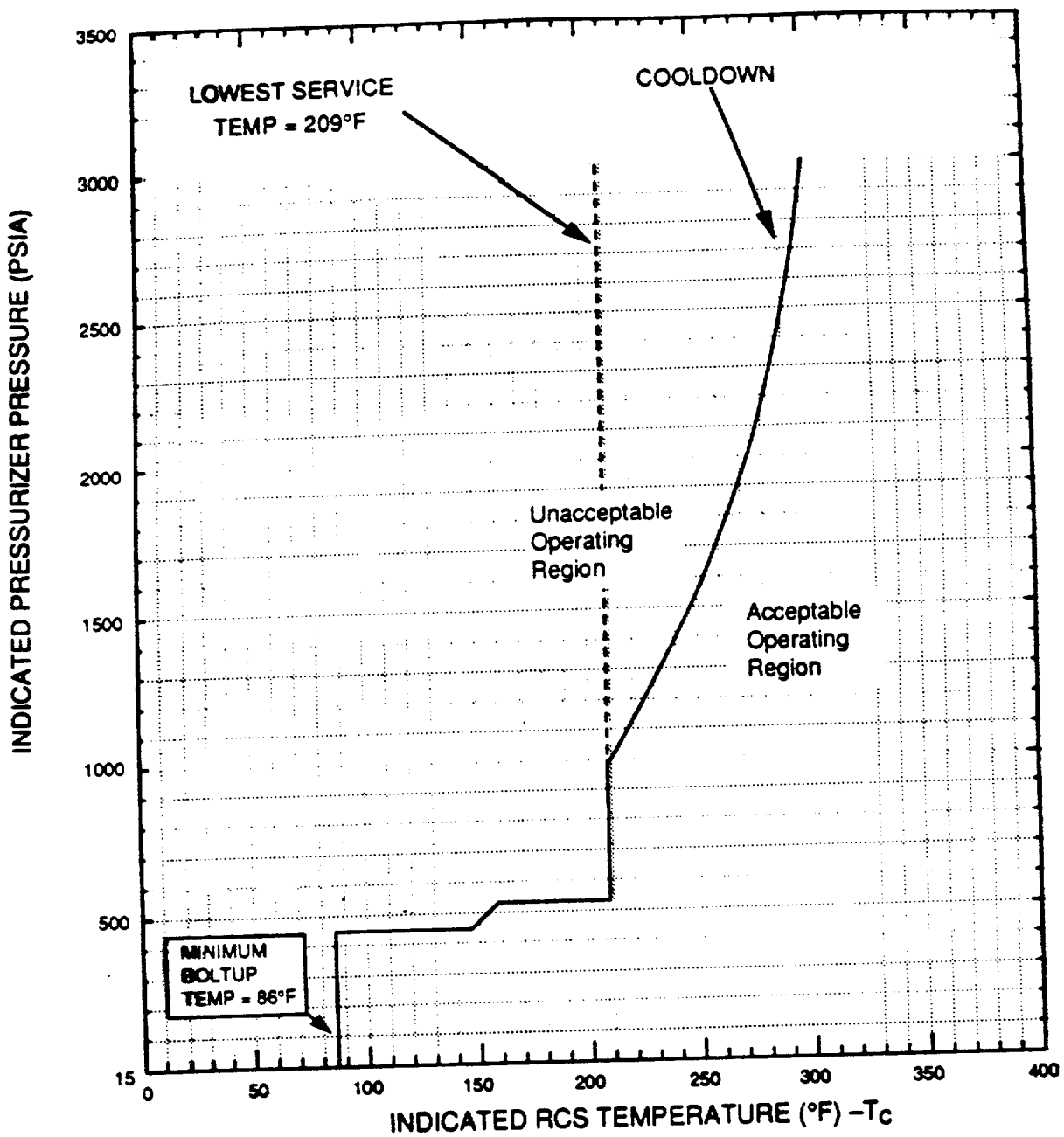
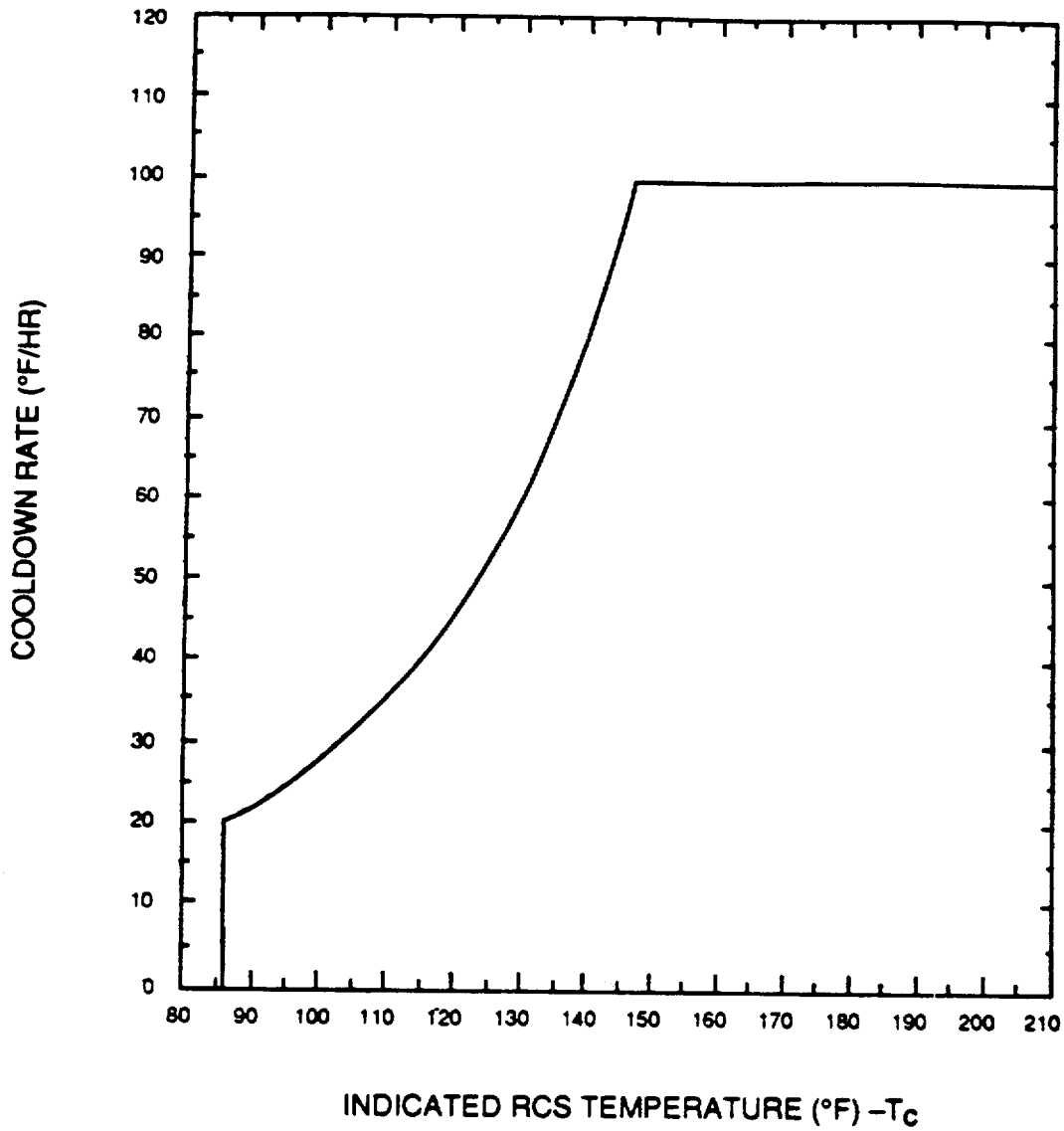


FIGURE 3.4-4

SONGS 3 COOLDOWN RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Normal Operation



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 147°F

FIGURE 3.4-5

SONGS 3 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (UNTIL 20 EFPY)
Normal Operation

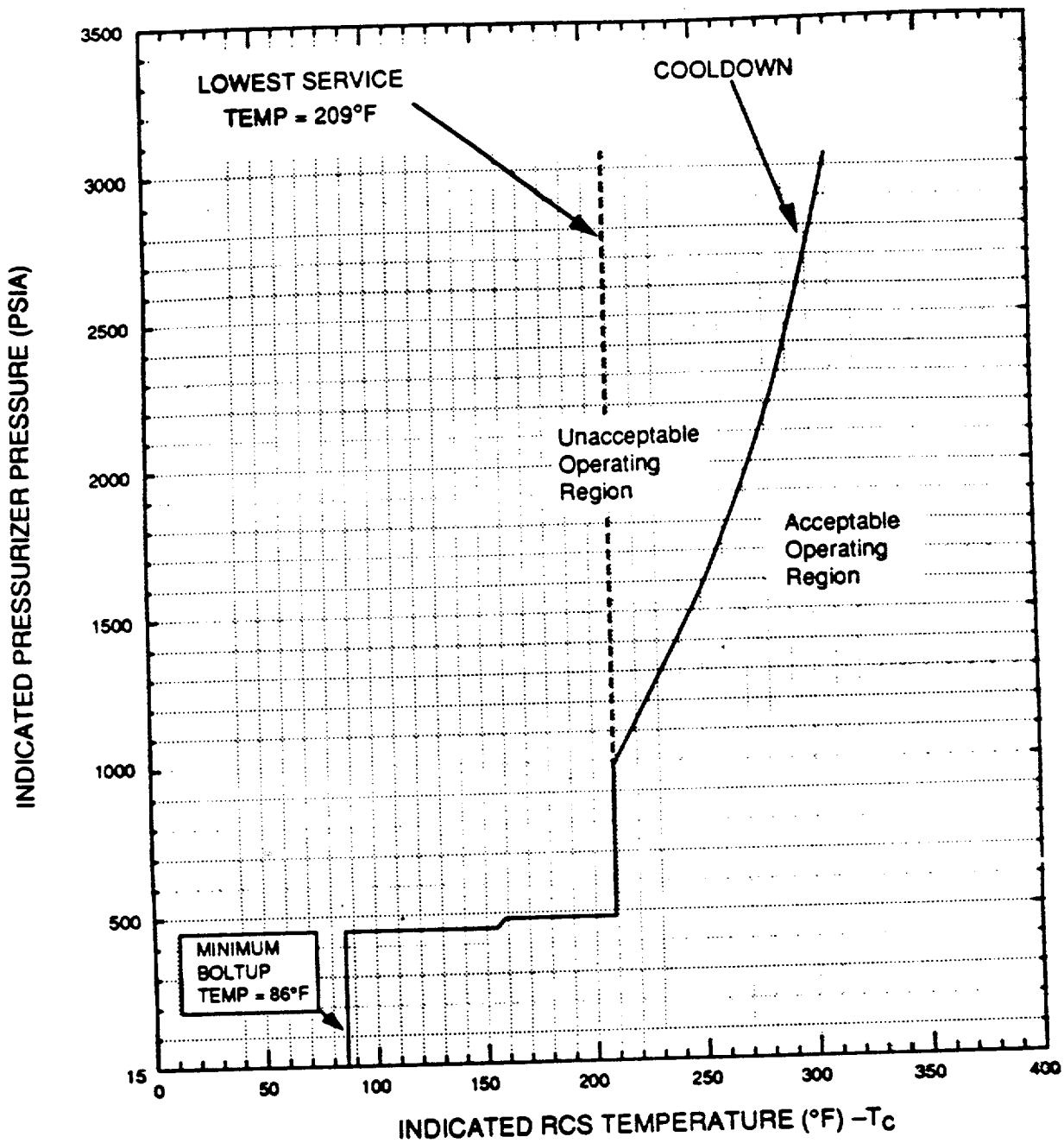
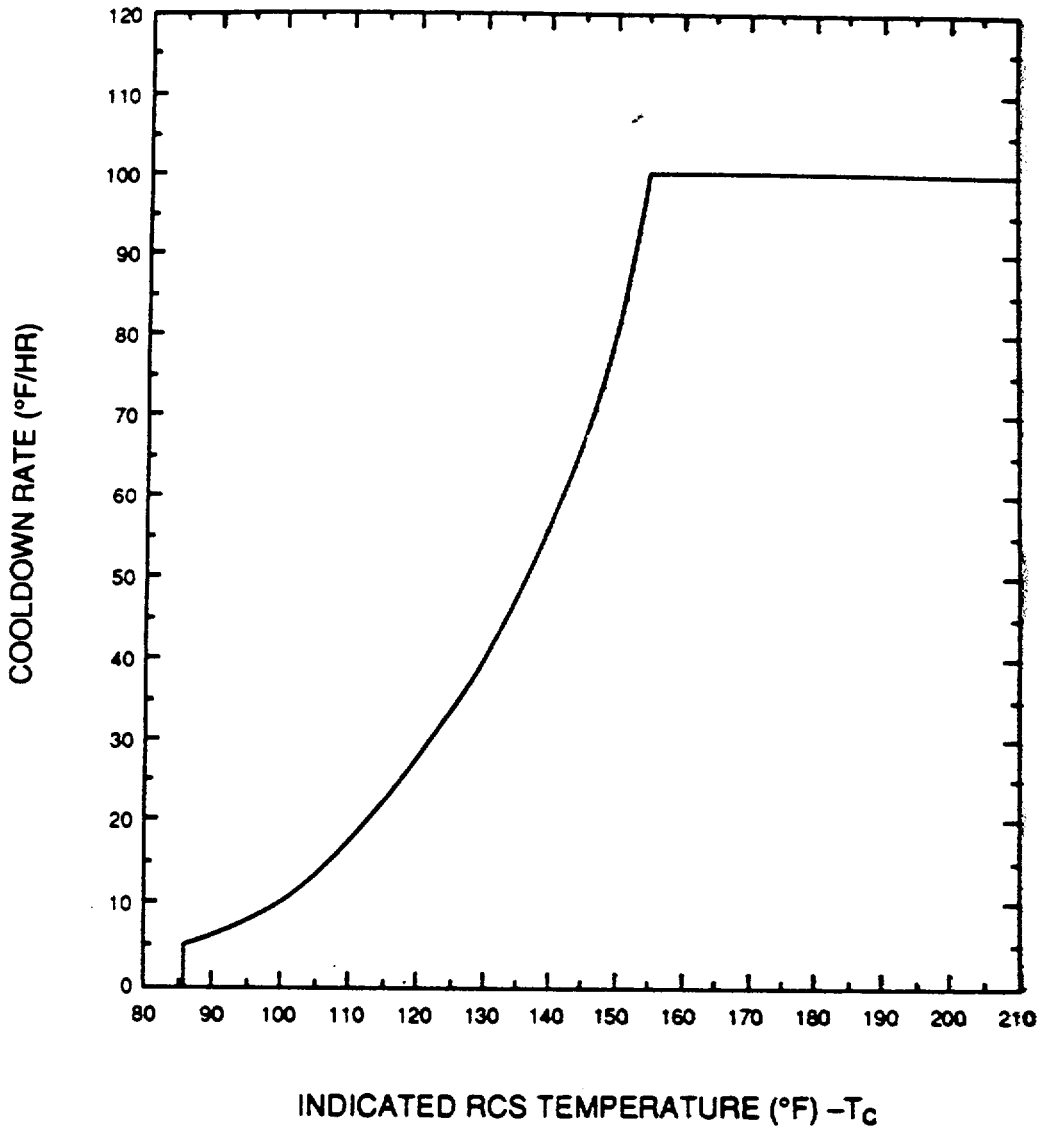


FIGURE 3.4-6

SONGS 3 COOLDOWN RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Remote Shutdown Operation



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 155°F

FIGURE 3.4-7

SONGS 3 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (UNTIL 20 EFPY)
Remote Shutdown Operation

Table 3.4-3

Low Temperature RCS Overpressure Protection Range

Operating Period, EFPY	Cold Leg Temperature, °F	
	During Heatup	During Cooldown
Until 20 (Normal Operation)	≤ 246	≤ 225
Until 20 (Remote Shutdown Operation)	*	≤ 225

*Heatup operations are not normally performed from the Remote Shutdown panels.

PRESSURIZER - HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any 1 hour period,

b. A maximum cooldown of 200°F in any 1 hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 for each cycle of main spray when less than 4 reactor coolant pumps are operating and for each cycle of auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 No more than two high-pressure safety injection pumps shall be OPERABLE and at least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

   1) A lift setting of 406 ± 10 psig*, and

   2) Relief valve isolation valves 3HV9337, 3HV9339, 3HV9377, and 3HV9378 open

or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to the enable temperatures specified in Table 3.4-3; MODE 5; and MODE 6 when the head is on the reactor vessel and the RCS is not vented.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 3HV9337 and 3HV9339 or valve pair 3HV9377 and 3HV9378) closed, open the closed valve(s) or power-lock open the other SDCS Relief Valve isolation valve pair within 24 hours, or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. With more than two high-pressure safety injection pumps OPERABLE, secure the third high-pressure safety injection pump by racking out its motor curcuit breaker or locking close its discharge valve within 8 hours.

*The lift setting pressure applicable to valve temperatures of less than or equal to 130°F.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. In the event either the SDCS Relief Vallve or an RCS vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve or RCS vent on the transient and any correcticve action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that SDCS Relief Valve isolation valves 3HV9337, 3HV9339, 3HV9377, and 3HV9378 are open.

b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.1.2 At least once per 12 hours, the third high-pressure safety injection pump shall be demonstrated to be secured by verifying that its motor circuit breaker is not racked-in or its discharge valve is locked closed. The requirement to rack out the third HPSI pump breaker is satisfied with the pump breaker racked out to its disconnected or test position.

4.4.8.3.1.3 At least once per 12 hours, the OPERABLE SDCS Relief Valve isolation valve pair (valve pair 3HV9337 and 3HV9339, or valve pair 3HV9377 and 3HV9378) that is used for overpressure protection due to the other SDCS Relief Valve isolation valve pair being INOPERABLE shall be verified to be in the power-lock open condition until the INOPERABLE SDCS Relief Valve isolation valve pair is returned to OPERABLE status or the RCS is depressurized and vented. The power-lock open requirement is satisfied either with the AC breakers open for valve pair 3HV9337 and 3HV9339 or the inverter input and output breakers open for valve pair 3HV9377 and 3HV9378, whichever valve pair is OPERABLE.

4.4.8.3.1.4 The RCS vent shall be verified to be open at least once per 12 hours* when the vent is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > 246°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.2 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

1) A lift setting of 406 ± 10 psig*, and

2) Relief valve isolation valves 3HV9337, 3HV9339, 3HV9377, and 3HV9378 open

or,

b. A minimum of one pressurizer code safety valve with a lift setting of 2500 psia ± 1%**.

APPLICABILITY: MODE 4 with RCS temperature above that specified in Table 3.4-3.

ACTION:

a. With no safety or relief valve OPERABLE, be in COLD SHUTDOWN and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. In the event the SDCS Relief Valve is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve code safety valve on the transient and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 3HV9337, 3HV9339, 3HV9377 and 3HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.

b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.2.2 The pressurizer code safety valve has no additional surveillance requirements other than those required by Specification 4.0.5.

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* The lift setting pressure applicable to valve temperatures of less than or equal to 130°F.

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

3.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves for normal operation (Figures 3.4-2 and 3.4-4) and the cooldown limit curve for remote shutdown operation (Figure 3.4-6) 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 limit curves for Remote Shutdown operation are determined using the Total Loop Uncertainties (TLUs) for temperature and pressure for the Remote Shutdown Panel instruments in which the pressure TLUs are higher than those for the Control Room shutdown instruments. 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 instrument uncertainties, and static and dynamic heads.

The reactor vessel materials were tested prior to reactor startup to determine their initial RT_{NDT}; the results of these tests and the updates resulting from the evaluation of material properties in response to Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1 are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 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 5.2-6 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup limit curve (Figure 3.4-2) and the cooldown limit curves, Figures 3.4-4 and 3.4-6, include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for instrument uncertainties, and static and dynamic heads.

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 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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, 3.4-4 and 3.4-6 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.

The Low Temperature Overpressure Protection (LTOP) enable temperatures are based upon the recommendations of NUREG-0800 Branch Technical Position (BTP) RSB 5-2, Revision 1, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." BTP RSB 5-2, Revision 1 defines the enable temperature as "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations."

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	Temperature of Charpy V-Notch @ 50 ft - lb	Minimum Upper Shelf Cv Energy for Longitudinal Direction-ft lb
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	40	70	118
215-02	C-6802-5	A533GRBCL1	Lower Shell Plate	0	40	70	116
215-02	C-6802-6	A533GRBCL1	Lower Shell Plate	-40	40	80	92
215-03	C-6802-1	A533GRBCL1	Intermediate Shell	-20	80	100	94
215-03	C-6802-2	A533GRBCL1	Intermediate Shell	-20	40	70	115
215-03	C-6802-3	A533GRBCL1	Intermediate Shell	-10	60	80	105
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. 122 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 111 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 separate letters dated April 30, 1993, as supplemented by letters dated July 6, 1994, and letter dated January 27, 1995, 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 (SONGS) Unit 2 and Unit 3. The proposed changes would revise TS 3/4.4.8.1, "Pressure-Temperature Limits," TS 3.4.8.3.1, "Overpressure Protection Systems-RCS Temperature ≤256°F [for Unit 2, ≤246°F for Unit 3]," TS 3.4.8.3.2, "Overpressure Protection Systems-RCS Temperature >256°F [for Unit 2, >246°F for Unit 3]," and the associated TS Bases. The proposed change 1) revises the Reactor Coolant System (RCS) Pressure-Temperature (P-T) limits and the Low Temperature Overpressure Protection (LTOP) enable temperatures to be effective until 20 Effective Full Power Years (EFPY) of operation and 2) makes minor editorial changes.

The July 6, 1994, supplemental letters for each unit and the January 27, 1995, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination which was published in the Federal Register on July 7, 1993 (58 FR 36445) for Unit 2 and June 23, 1993 (58 FR 34094) for Unit 3.

2.0 DISCUSSION

The maximum allowable RCS pressure at any temperature is based upon the stress limitations for brittle fracture. TS 3/4.4.8.1, "Reactor Coolant System-Pressure Temperature Limits," provides operational constraints in all modes of

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reactor operation to ensure that the most stress limiting location in the reactor vessel is not susceptible to brittle failure as a consequence of reactor operations. The neutron-induced embrittlement of the reactor vessel wall also affects the temperature below which LTOP is required. LTOP is provided by the shutdown cooling system (SDCS) relief valve. The SDCS relief valve must be aligned below the specified temperature to provide assurance that the reactor vessel wall will be operated in the ductile region in accordance with 10 CFR 50 Appendix G during both normal operation and overpressurization events due to equipment malfunction or operator error. The existing TS require alignment of the SDCS relief valve below the temperature corresponding to the P-T curve pressurizer relief valve setpoint of 2500 psia.

The existing TS RCS P-T limit curves in TS 3/4.4.8, Figures 3.4-2 and 3.4-3, were originally valid for 8 EFPY of plant operation. The proposed TS P-T limit curves will be valid for 20 EFPY of plant operation.

The proposed LTOP enable temperatures represent the most limiting temperatures for normal heatup, normal cooldown, and remote shutdown cooldown operations. Below these LTOP enable temperatures the SDCS relief valve must be aligned during heatup operations and during cooldown operations from either the remote shutdown panels or the control room.

Due to the proposed TS changes, the Updated Final Safety Analysis Report (UFSAR) will be revised to reflect the (1) LTOP enable temperatures for heatup and cooldown operations, and (2) updated material properties evaluated in response to Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)."

3.0 EVALUATION

The staff evaluated the proposed P-T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; GLs 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. The proposed changes to the LTOP enable temperatures were evaluated using the guidance provided in SRP 5.2.2 Branch Technical Position RSB 5-2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the ASME Code. GL 88-11 provides that licensees may use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff will use in the review of the P-T limits submittals. SRP 5.3.2 provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. In a letter dated January 27, 1995, the licensee provided additional technical information to explain how

the proposed P-T limits comply with the NRC regulations and guidance described above.

For the Unit 2 reactor vessel, the licensee determined that base plates C-6404-5 and C-6404-1 are the limiting materials at the 1/4T and 3/4T locations, respectively (T = the thickness of the reactor vessel beltline). The licensee calculated an ART of 129°F at the 1/4T location and 106.5°F at the 3/4T location. The input variables for the ART calculations are shown in Table 1.

For the Unit 3 reactor vessel, the licensee determined that base plate C-6802-1 is the limiting material for both the 1/4T and 3/4T locations. The licensee calculated an ART of 116°F at the 1/4T location and an ART of 99.8°F at the 3/4T location.

The staff verified that the copper and nickel contents and initial RT_{NDT} used by the licensee in their analysis were in agreement with the values in the NRC reactor vessel material database. The staff used the material properties to perform an independent calculation of the ART values for the limiting materials using RG 1.99, Rev. 2. Based on the staff's calculation, the staff verified that the licensee's calculated ARTs are acceptable.

By substituting the ARTs from the attached Table into the equations listed in SRP 5.3.2, the staff verified that the proposed P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test satisfy the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 40°F provided by the licensee, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

The staff has performed an independent analysis to verify the licensee's proposed P-T limits. The staff concludes that the proposed P-T limits for heatup, cooldown, inservice hydrostatic test, and criticality are valid for 20 effective full power years because (1) the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11, and (2) the material properties and chemistry used in calculating the P-T limits are consistent with data submitted under GL 92-01. Hence, the proposed P-T limits are acceptable for incorporation into the San Onofre Units 2 and 3 TS.

The staff has reviewed the analysis performed by the licensee to support the proposed changes to the LTOP enable temperatures, and has determined that the analysis is in accordance with SRP Branch Technical Position (BTP) RSB 5-2,

Revision 1, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." BTP RSB 5-2, Revision 1 defines the LTOP enable temperature as "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4T or 3/4T) that is controlling in the Appendix G limit calculations." This calculational basis changes the LTOP alignment temperature from 312°F to 256°F for Unit 2, and from 302°F to 246°F for Unit 3. The P-T limits will not be exceeded by postulated overpressurization transients with the SDCS relief valve aligned to the RCS system at the proposed LTOP enable temperatures. Therefore, the proposed LTOP enable temperature change is acceptable to the staff.

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 36445 and 58 FR 34094). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Table

Principal Contributor: John Tsao, NRR

Date: July 18, 1995

Table

ADJUSTED REFERENCE TEMPERATURE FOR
REACTOR VESSEL LIMITING MATERIALS
FOR PRESSURE-TEMPERATURE LIMITS
20 EFFECTIVE FULL POWER YEARS
SAN ONOFRE UNITS 2 AND 3

Limiting Materials: Plates	Cu	Ni	Fluence	ΔRT_{NDT}	Initial RT_{NDT}	Margin	ART
Unit 2							
C6404-5 (1/4T)	.11	.64	1.625	85	10	34	129
C6404-1 (3/4T)	.10	.56	0.503	52.5	20	34	106.5
Unit 3							
C6802-1 (1/4T)	.06	.58	1.625	42	40	34	116
C6802-1 (3/4T)	.06	.58	0.503	29.9	40	29.9	99.8

Fluence units are in 10^{19} neutron/cm^2