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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. NPF-15

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

A. The application for amendment to the license for San Onofre Nuclear Generating Station, Unit 3 (the facility) filed by Southern California Edison Company (SCE) on behalf of itself and San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California (licensees) dated June 12, 1989 and supplemented July 19, 1989 and November 6, 1989, 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 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. 71 , are hereby incorporated in the license. SCE 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

George W. Knighton

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 14, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 71

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 area of change. Also enclosed are the following overleaf pages to the amended pages.

AMENDMENT PAGE	OVERLEAF PAGE
V	VI
XVII	XVIII
XIX	XX
3/4 4-3	3/4 4-4
3/4 4-5	3/4 4-5a
3/4 4-28	3/4 4-27
3/4 4-28a	--
3/4 4-29	--
3/4 4-30	--
3/4 4-30a	--
3/4 4-31	--
3/4 4-31a	--
3/4 4-31b	3/4 4-32
3/4 4-33	3/4 4-34
3/4 4-35	3/4 4-36
B 3/4 4-1	B 3/4 4-2
B 3/4 4-6	B 3/4 4-5
B 3/4 4-7	B 3/4 4-8
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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump,**

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required Reactor Coolant loops and/or shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling train, be in COLD SHUTDOWN within 24 hours.

b. With no Reactor Coolant loop or shutdown cooling train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop/train to operation.

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*All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 when the secondary water temperature of each steam generator is greater than 100°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% (wide range) at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 a. At least one of the following loop(s)/trains listed below shall be OPERABLE and in operation*:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant Pump**
2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant Pump**
3. Shutdown Cooling Train A
4. Shutdown Cooling Train B

b. One additional Reactor Coolant Loop/shutdown cooling train shall be OPERABLE, or

c. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5, with Reactor Coolant loops filled.

ACTION:

a. With less than the above required shutdown trains/loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.

b. With no loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop/train to operation.

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*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 when the secondary water temperature of each steam generator is greater than 100°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS FILLED

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The required Reactor Cooling pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.4.1.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$10% (wide range) at least once per 12 hours.

4.4.1.4.1.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

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, 3.4-3, 3.4-4, and 3/4-5 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup as specified by Figure 3.4-3 in any 1-hour period with RCS cold leg temperature less than 153°F. A maximum heatup of 60°F. in any 1-hour period with RCS cold leg temperature greater than 153°F.

b. A maximum cooldown as specified by Figure 3.4-5 in any 1-hour period with RCS cold leg temperature less than 126°F. A maximum cooldown of 100°F in any 1-hour period with RCS cold leg temperature greater than 126°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.

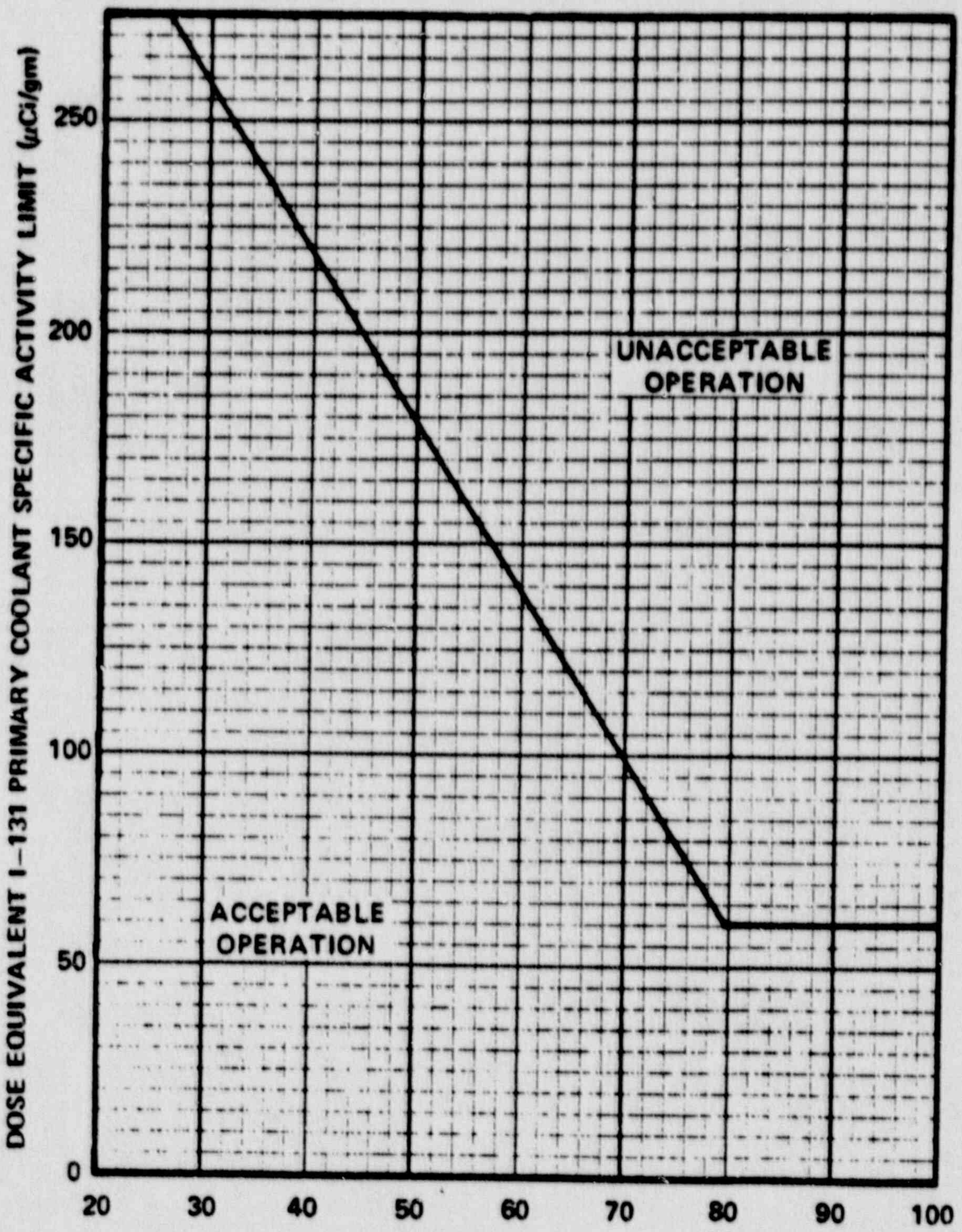


FIGURE 3.4-1

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

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, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. 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 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.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
1	83°	1.5	Standby
2	97°	1.5	4.4 EFPY
3	104°	1.5	15.2 EFPY
4	284°	1.5	24 EFPY
5	263°	1.5	Standby
6	277°	1.5	Standby

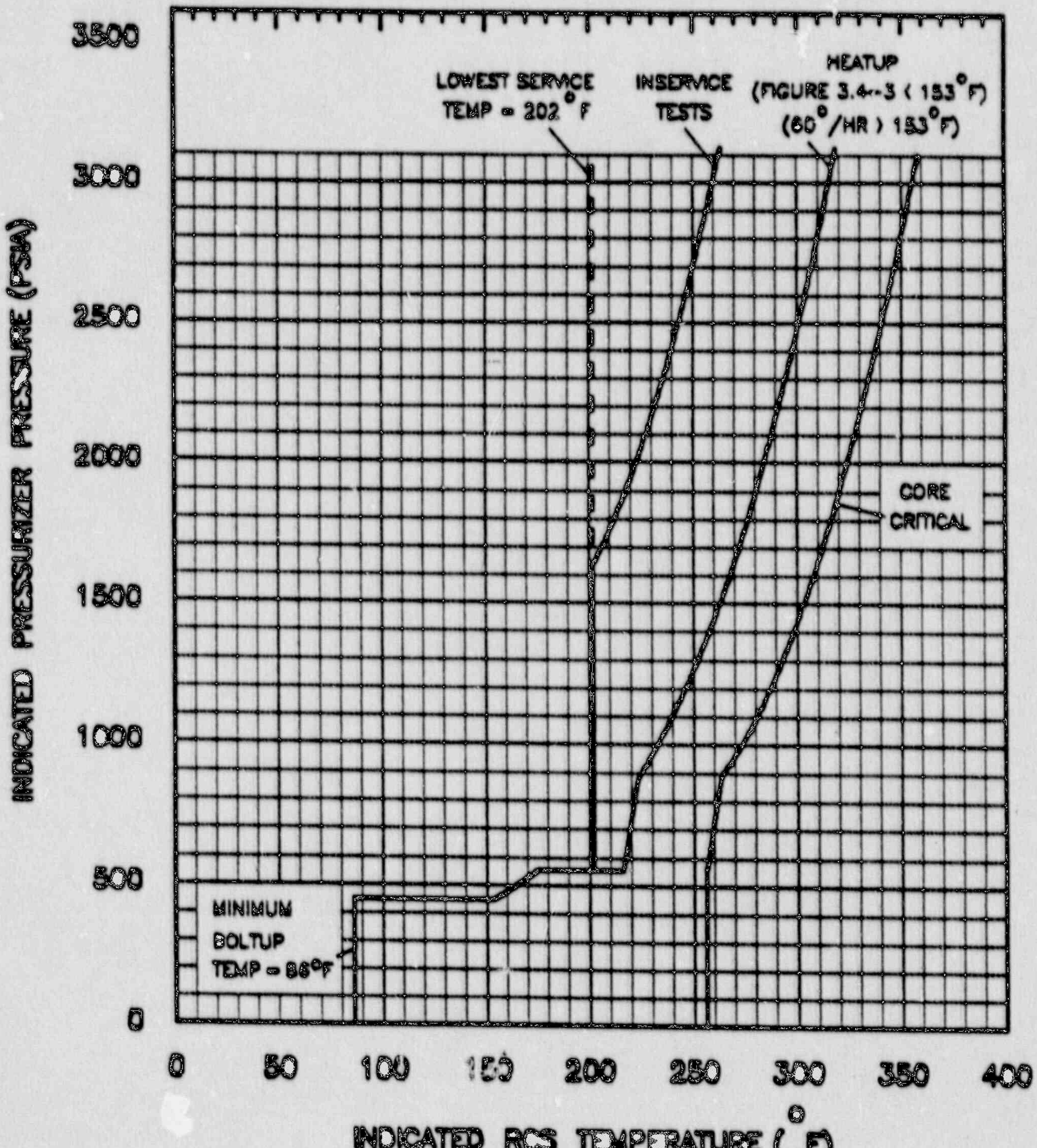
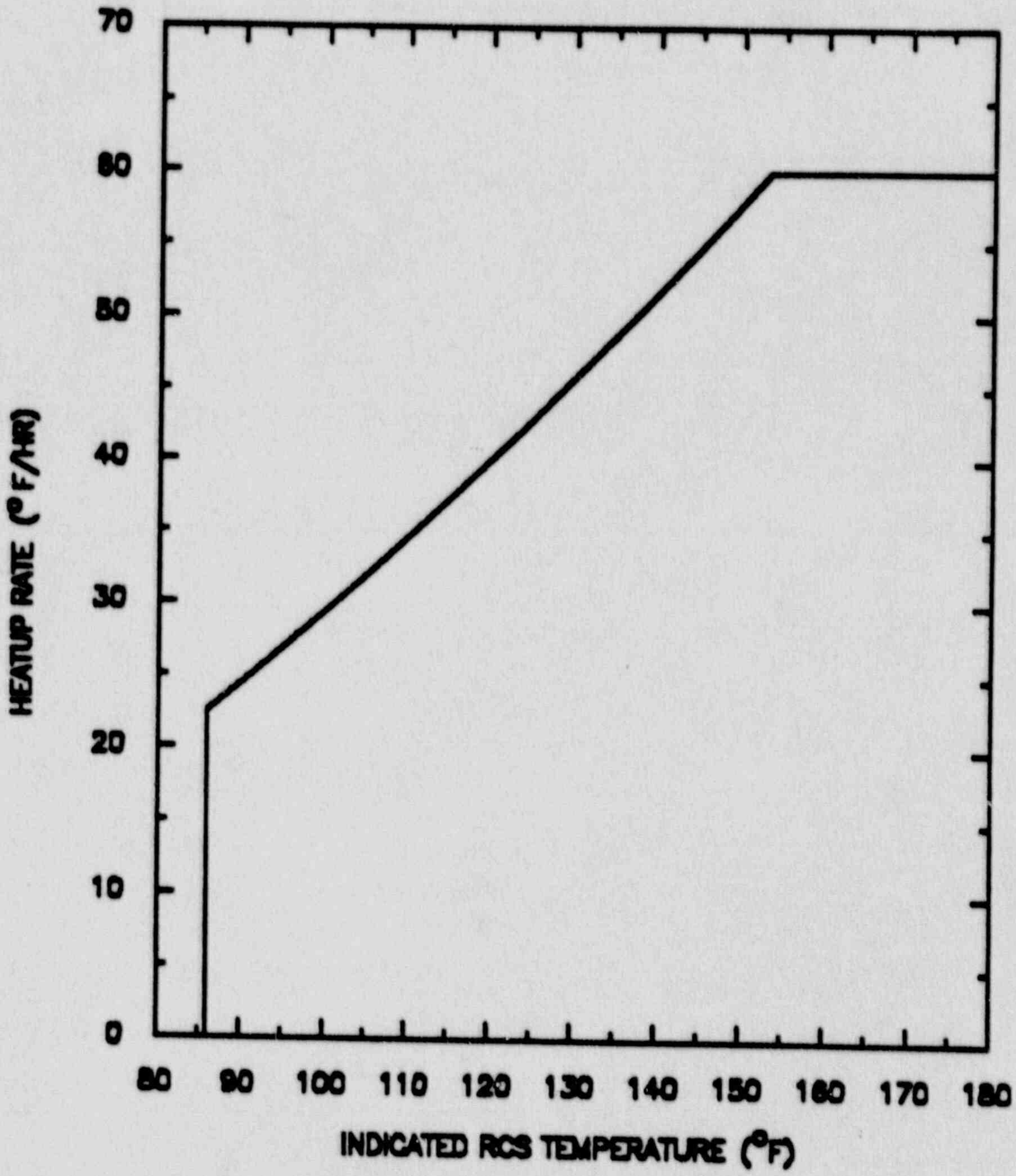


FIGURE 3.4-2

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



NOTE: A MAXIMUM HEATUP RATE OF 60°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 153°F

FIGURE 3.4-3

SONGS 3 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE HEATUP RATES (4-8 EFPY)

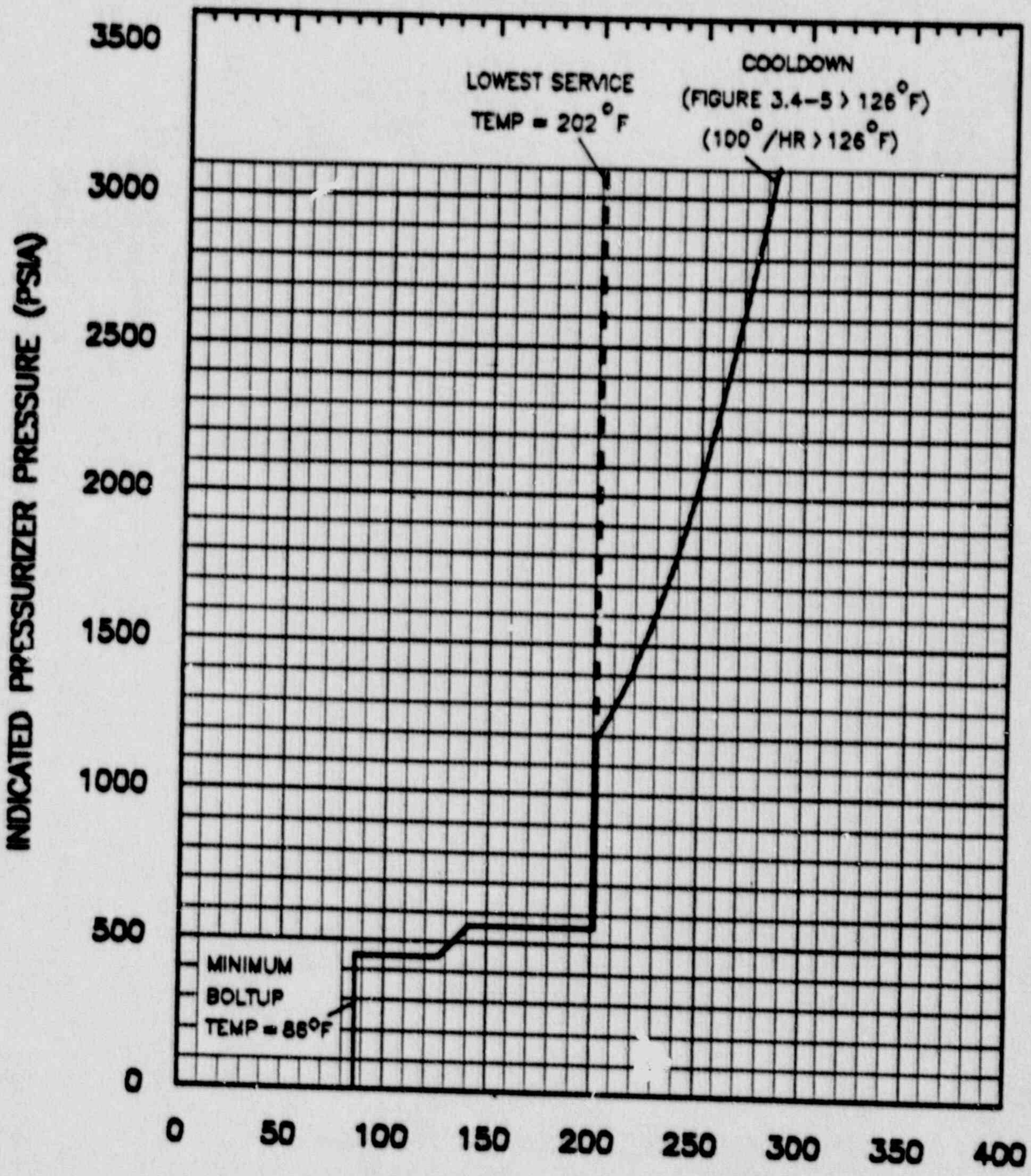
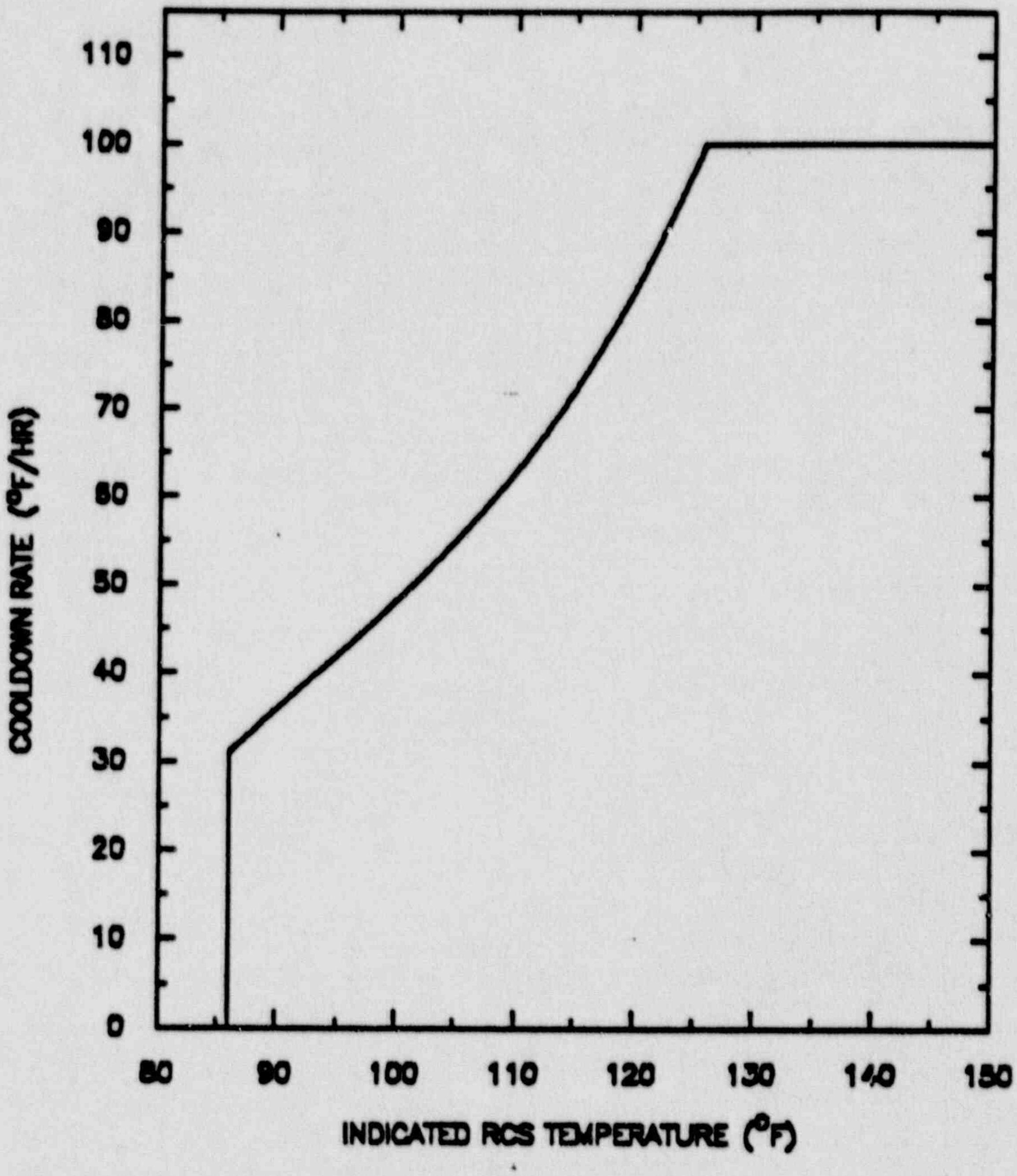


FIGURE 3.4-4

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



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 126°F

FIGURE 3.4-5

SONGS 3 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (4-8 EFPY)

Table 3.4-3

Low Temperature RCS Overpressure Protection Range

Operating Period, EFPY	Cold Leg Temperature, °F	
	During Heatup	During Cooldown
4 to 8	$\leq$ 302	$\leq$ 267

REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE $\leq$ 302°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 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 that specified in Table 3.4-3; MODE 5; MODE 6 with the reactor vessel head on.

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) within 7 days 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. In the event either the SDCS Relief Valve 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 corrective action necessary to prevent recurrence.

d. 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.

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

SURVEILLANCE REQUIREMENTS (Continued)

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 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 > 302°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 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 code safety valve or RCS vent 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.

4.4.8.3.2.3 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.

*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.

REACTOR COOLANT SYSTEM

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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps (RCPs)in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than four reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 with one or more RCS cold legs less than or equal to that specified in Table 3.4-3 are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6 x 105 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than that specified in Table 3.4-3. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 285°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

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.

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 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 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 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 shown in Table 4.4-5. 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 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.

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	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	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

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or an RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs is less than or equal to that specified in Table 3.4-3. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

Reactor coolant system gas vents are provided to exhaust noncondensible gases from the primary system that could inhibit natural circulation core cooling following a non-design bases accident. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The design redundancy of the Reactor Coolant Gas Vent System serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant Gas Vent System are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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
205-03	C-6831-1	A508CL1	Inlet Nozzle Forging S/E	-20	12	40	124
205-03	C-6831-2	A508CL1	Inlet Nozzle Forging S/E	-20	12	40	124
205-03	C-6831-3	A508CL1	Inlet Nozzle Forging S/E	-20	-15	50	114
205-03	C-6831-4	A508CL1	Inlet Nozzle Forging S/E	-20	-15	50	114
205-07	C-6832-1	A508CL1	Outlet Nozzle Forging S/E	-20	-20	0	159
205-07	C-6832-2	A508CL1	Outlet Nozzle Forging S/E	-20	-20	0	152
231-01	C-6833-1	A533GRBCL1	Closure Head Peel	-40	20	NA	NA
231-01	C-6834-1	A533GRBCL1	Closure Head Peel	-30	10	NA	NA
231-02	C-6835-1	A533GRBCL1	Closure Head Dome	-40	10	NA	NA

NA = Not Available