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SCE

October 4, 1982

Director, Office of Nuclear Reactor Regulation
Attention: Ms. Janis D. Kerrigan, Acting Branch Chief
Licensing Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-363
San Onofre Nuclear Generating Station
Unit 3

During the week of September 13, 1982, meetings were held at San Onofre Nuclear Generating Station (SONGS) with D. Brinkman and D. Hoffman of the NRC to discuss Unit 3 Technical Specifications. Fourteen items required for Unit 3 technical specifications remained open at the conclusion of the meetings with responsibilities for resolution of these items as indicated in Attachment 1. The purpose of this letter is to transmit the information required to resolve those items listed in Attachment 1 as Southern California Edison Company's (SCE) responsibility. This information is summarized as follows:

Item 1 Clarification of Action Requirements for Items 19, 20, 21 and 22 on Table 3.3-10.

Table 3.3-10 Action 22 in Unit 2 Technical Specifications included a circular reference to specification 3.3.3.6 of which it is a part. Table 3.3-10 and associated Action statements have been revised as shown in Attachment 2 to eliminate this circular reference.

Item 3 Verify Applicability of Table 4.4-5 and provide additional wording for the bases to explain "Lead Factor".

Table 4.4-5 has been revised to be applicable to Unit 3 and is included as Attachment 3. The bases 3/4.4.8 currently contains an adequate explanation of the purpose of the "Lead Factor". No additional changes to the bases are required.

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Item 4 Provide heat up and cool down curves (Figures 3.4-2 and 3.4-3) applicable to Unit 3.

Figures 3.4-2 and 3.4-3 have been revised for Unit 3 and are included in Attachment 4. Additionally, corresponding changes to pages 3/4 4-3, 4-27, 4-32, 4-33, to Bases pages B 3/4 4-1, 4-6, 4-7, 4-9, and to Bases Table B 3/4, 4-1 are included in Attachment 4 reflecting differences from Unit 2.

Item 5 Resolution of correct pressure to be used in 4.5.1.e.1.

The correct pressure to be used in 4.5.1.e.1 is 715 psia which is the pressure above which the safety injection tanks (SIT) are required to be operable. Although the set point for automatic opening of the SIT isolation valves is 515 psia, no credit is taken in the accident analysis for the automatic opening of these valves because they are locked opened when RCS pressure is greater than 715 psia (by locking open the breaker) during normal operation, the initial state in the accident sequences analysed.

Item 6 Pump performance data for Sections 4.5.2.f and g.

Section 4.5.2.f and 4.5.2.g have been revised with Emergency Core Cooling System pump performance data applicable to Unit 3 and are included as Attachment 5.

Item 7 Provide justification in the Refueling Machine bases B3/4.9.6 for the exception in 3.9.6 for four finger CEAs.

Revised bases B3/4.9.6 addressing the exception for not using the refueling machine to move the four finger CEAs is included as Attachment 6.

Item 8 Revise bases B3/4.10.1 clarifying the reason for entry into MODE 3 during performance of CEA worth measurement tests.

A paragraph is added to bases B3/4.10.1 as shown in Attachment 7 which explains the reason for entry into MODE 3 during CEA worth measurement tests.

Item 11 Revise bases to explain that tank volumes required to be maintained by technical specification are usable volumes.

The bases for technical specifications 3/4.1.2 (Boration Systems), 3/4.5.4 (Refueling Water Storage Tank) and 3/4.7.1.3 (Condensate Storage Tanks) have been revised as shown in Attachment 8 to explain the usable volume considerations in the technical specifications. Bases 3/4.6.2.2 (Iodine Removal System) was not revised because the contained volume and usable volume of the tank are the same because the discharge line is at the bottom of the tank and there are no internal structures to reduce the usable volume.

Item 12 Identify location of seismic monitoring instrumentation.

Notes as shown in Attachment 9 have been added to Tables 3.3-7 and 4.3-4 to indicate the location of common seismic instrumentation.

The information provided in this letter should complete SCE's open items enabling the issuance of final draft Unit 3 Technical Specifications. Should you have any questions regarding the information provided in this letter, please call me.

Very truly yours,

M.O. Medford for KPB

Enclosure

cc: Harry Rood, NRC (to be opened by addresse only.)
D. Brinkman, NRC (to be opened by addressee only.)

ATTACHMENT 1

SAN ONOFRE UNIT 3 TECHNICAL SPECIFICATIONS

Action No.	Action	Responsibility	Completion Date
1	Clarify Action Requirements For Items 19, 20, 21, 22 p. 3/4 3-53 Table 3.3-10	SCE	October 4, 1982
2	Determine if Additional wording in 4.4.4.3 c. 4 is accountable p. 3/4 4-11	NRC	October 1, 1982
3.	Verify Lead Factor and Previous Additional Wording for Bases Table 4.4-5 p. 3/44-28	SCE	October 4, 1982
4.	Previous Figures 3.4-2 and 3.4-3 (on graph paper) pps 3/4 4-29 and 3/4 4-30	SCE	October 4, 1982
5.	Resolve Safety Analysis Number for 4.5 1.e.1 (715 or 515) p. 3/4 5-2	SCE	October 4, 1982
6.	Provide Pump Performance Draft for 4.5.2.f and 4.5.2.g p. 3/4 5-5 and 3/4 5-6	SCE	October 4, 1982
7.	Provide Wording for Bases Justifying the Exceptions for the Four Finger CEAs for Specification 3.9.6 p. 3/4 9-6	SCE	October 4, 1982
8.	Provide Wording for Bases Clarifying Reason for Entering MODE 3 When Performing Special Test Exception 3/4 10.1 p. 3/4 10-1	SCE	October 4, 1982
9.	Provide Pages 3/4 11-11 Through 3/4 11-19 as they were Inadvertently Deleted	NRC	October 1, 1982

Action No.	Action	Responsibility	Completion Date
10.	Determine What the ** was Initially Intended For in Technical Specification 3/4 12.2 p. 3/4 12-11	NRC	October 1, 1982
11.	Provide Bases for Usable Tank Volumes and How Determined for a Number of Technical Specifications in the Boration Systems, Plant Systems, ECCS Systems and Containment Systems in the Bases	SCE	October 4, 1982
12.	Identify Location of Seismic Monitoring Information and Provide Proposed Technical Specification for Technical Specification 3.3.3.3 pps 3/4 3-42, 3,-43 and 3-44	SCE	October 4, 1982
13.	Monitor Technical Specifications for Seismic Monitoring, Meteorological Monitoring and the Control Room Emergency Air Cleanup System to Reflect that these are Shared Systems Between Units 2&3 Technical Specifications 3.3.3.3, and 5.7.5	NRC	October 1, 1982
14.	Add Missing Prompt Notification Items J and K to Technical Specification 6.9.1.12 p. 6-20 and Revise Thirty Day Written Reports Technical Specification 6.9.1.13 p. 6-21 for Possible Missing Items	NRC	October 1, 1982

ATTACHMENT 2

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT		REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17.	Containment Water Level - Wide Range	2	1	20, 21
18.	Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19.	Containment Area Radiation - High Range	2	1	~~22, 23~~ 20, 21
20.	Main Steam Line Area Radiation	1/steam line	N.A.	~~22~~ 20
21.	Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	~~22~~ 20
22.	Purge/Vent Stack Radiation Monitor - Wide Range*	2	~~[illegible]~~ N.A.	22, ~~[illegible]~~
23.	Cold Leg HPSI Flow	1/cold leg	N.A.	20
24.	Hot Leg HPSI Flow	1/hot leg	N.A.	20

NOTES:

*The two required channels are the Unit 2 monitor and the Unit 3 monitor.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

~~ACTION 22 - With the number of channels OPERABLE less than the Required Number of Channels, comply with the ACTION requirements of Specification 3.3.3.6.~~

ACTION ~~23~~ 22 - With the number of OPERABLE Accident Monitoring Channels less than the required number of channels, ~~by the Minimum Channels OPERABLE requirement,~~ either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ATTACHMENT 3

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
1	83°	~~1.15~~ 1.5	Standby
2	97°	~~1.15~~ 1.5	5.6 EFPY
3	104°	~~1.15~~ 1.5	15.2 EFPY
4	284°	~~1.15~~ 1.5	24 EFPY
5	263°	~~1.15~~ 1.5	Standby
6	277°	~~1.15~~ 1.5	Standby

ATTACHMENT 4

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the loop(s)/train(s) listed below shall be OPERABLE:

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. At least one of the above Reactor Coolant loops and/or shutdown cooling trains shall be in operation.*

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 285°F of the Reactor Coolant System cold leg temperatures less than or equal to ~~[illegible]~~°F unless 1) the pressurizer water volume is less than 900 cubic feet or 2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

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REACTOR COOLANT SYSTEM

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

A. A MAXIMUM HEATUP OF 10°F IN ANY ONE HOUR PERIOD WITH RC COLD LEG TEMPERATURE LESS THAN 110°F. A MAXIMUM HEATUP OF 30°F IN ANY ONE HOUR PERIOD WITH RC COLD LEG TEMPERATURE GREATER THAN 110°F BUT LESS THAN 330 °F. A MAXIMUM HEATUP OF 60°F IN ANY ONE HOUR PERIOD WITH RC COLD LEG TEMPERATURE GREATER THAN 330 °F.

B. A MAXIMUM COOLDOWN OF 10°F IN ANY ONE HOUR PERIOD WITH RC COLD LEG TEMPERATURE LESS THAN 110°F. A MAXIMUM COOLDOWN OF 30°F IN ANY ONE HOUR PERIOD WITH RC COLD LEG TEMPERATURE GREATER THAN 110°F BUT LESS THAN 200°F. A MAXIMUM COOLDOWN OF 100°F IN ANY ONE HOUR PERIOD WITH RC COLD LEG TEMPERATURE GREATER THAN 200°F.

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

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.

SURVEILLANCE REQUIREMENTS

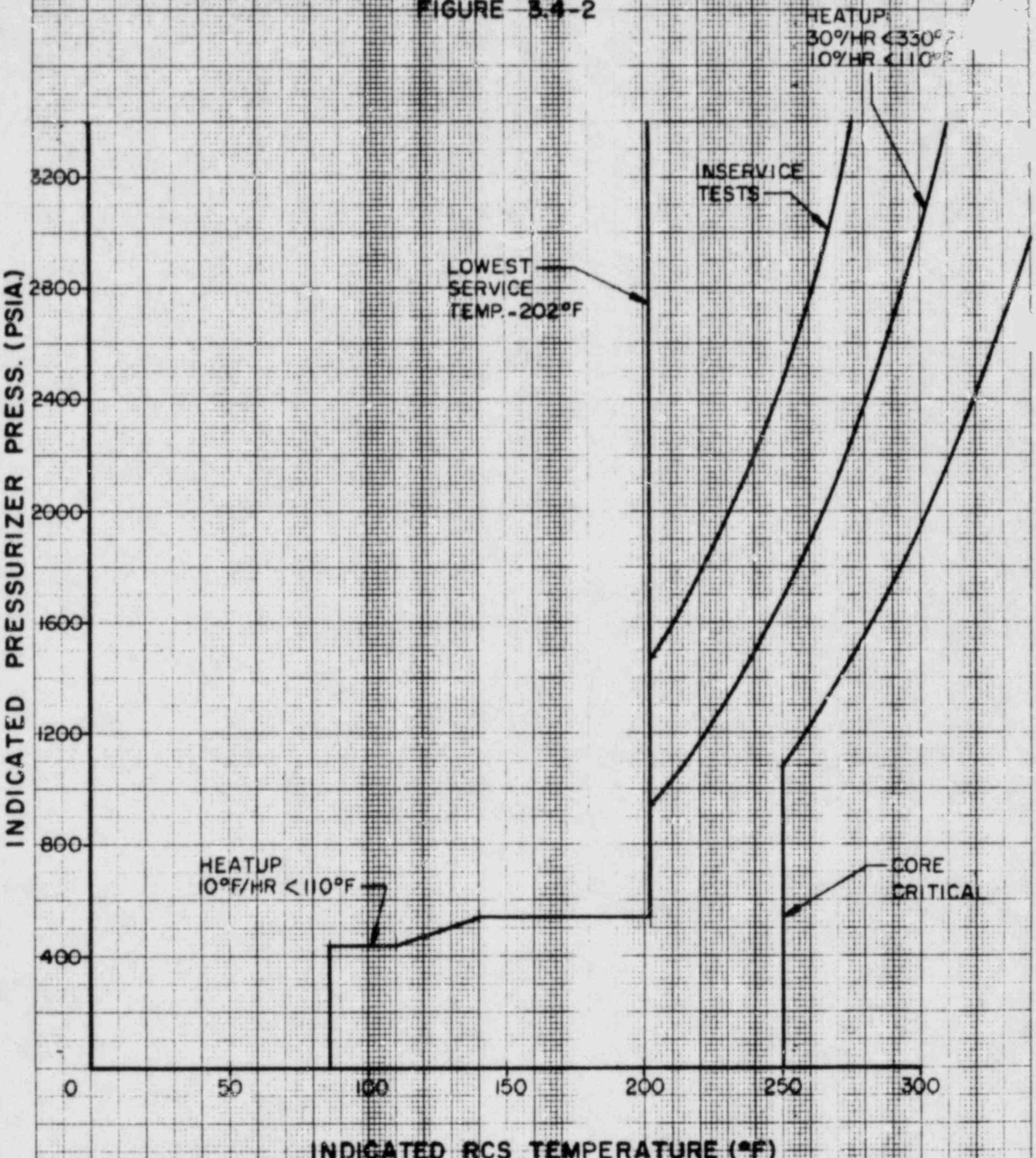
4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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 ~~C-6404-2~~ as determined by impact testing, or

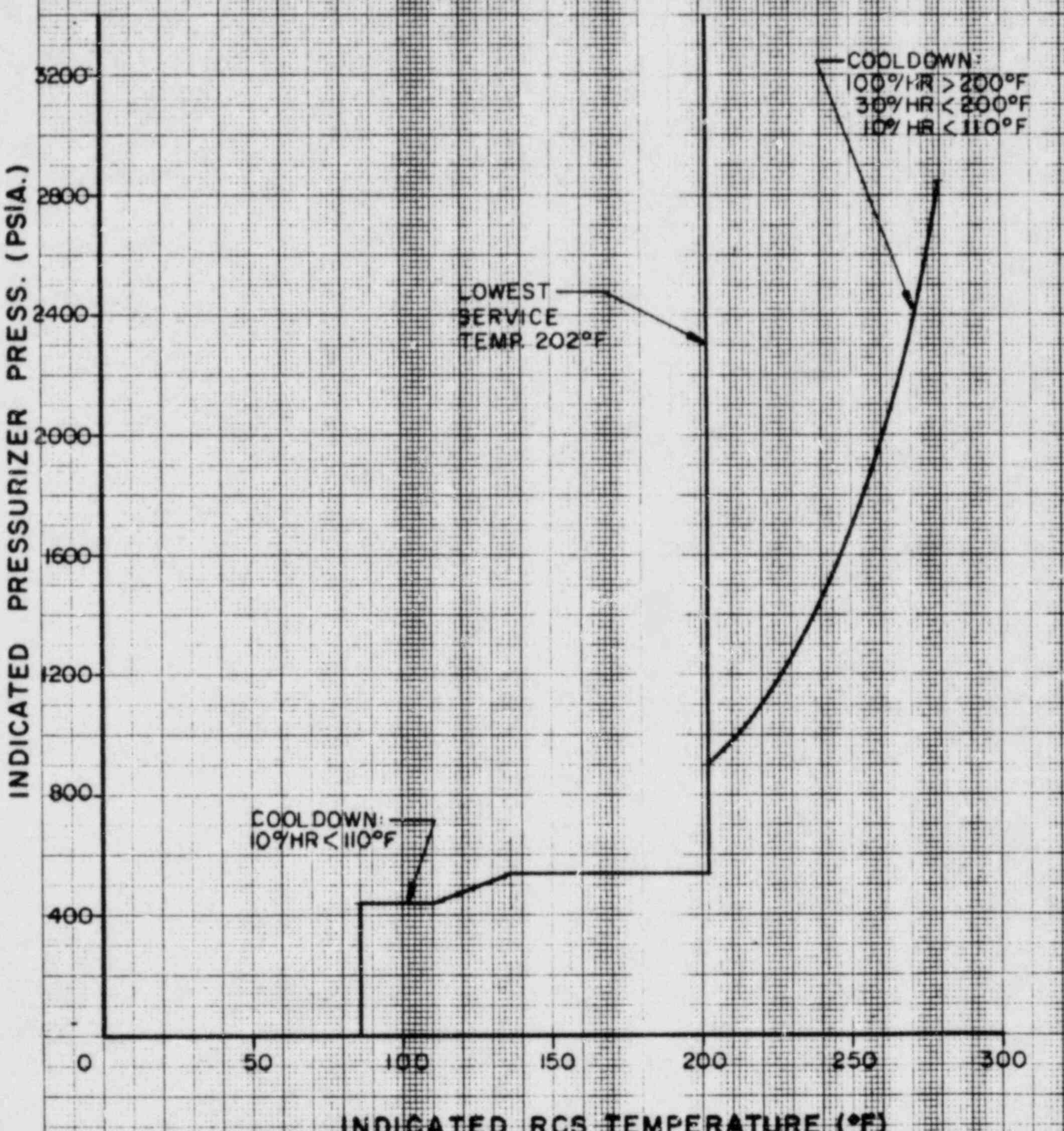
b. The predicted shift in reference temperature for weld seams 2-203A, 2-203B or 2-203C ~~2-203B~~ as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

FIGURE 3.4-2



HEATUP RCS PRESSURE/TEMPERATURE LIMITATIONS FOR SCE III 0-5 YRS

FIGURE 3.4-3



COOLDOWN RCS PRESSURE/TEMPERATURE LIMITATIONS FOR SCE III 0-5YRS

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE ≤ ~~235°F~~ 285°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 ~~2~~3HV9337, ~~2~~3HV9339, ~~2~~3HV9377 and ~~2~~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 ~~235~~ 285°F; 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 ~~2~~3HV9337 and ~~2~~3HV9339 or valve pair ~~2~~3HV9377 and ~~2~~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 ~~2~~3HV9337, ~~2~~3HV9339, ~~2~~3HV9377 and ~~2~~3HV9378 are open.

*For valve temperatures less than or equal to 130°F.

~~JUN 1 8 1982~~

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > ~~[illegible]~~ 285°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 ~~2~~3HV9337, ~~2~~3HV9339, ~~2~~3HV9377 and ~~2~~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 ~~[illegible]~~ 285°F.

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 3~~2~~HV9337, 3~~2~~HV9339, 3~~2~~HV9377 and 3~~2~~HV9378 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.

*For valve temperatures 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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR [redacted] greater than 1.20 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 4 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 285 [redacted]°F 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 10^5 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 285 [redacted]°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to [redacted]°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.
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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 vary from compressive at the inner wall to tensile at the outer wall. These thermally induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermally induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

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 Figure 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test 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 phosphorous content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to 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 at the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

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

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

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

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

SAN ONOFRE UNIT 3

Piece No.	Code No.	Material	Vessel Location	DW_{NDT} ($^{\circ}F$)	Temperature of Charpy V-Notch @ 30 ft - lb	@ 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

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

SAN ONOFRE UNIT 3

Piece No.	Code No.	Material	Vessel Location	RT_{NDT} (°F)	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

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a 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 are less than or equal to ~~[illegible]~~°F. The Shutdown Cooling System relief valve has adequate relieving 285 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) inadvertant safety injection actuation with two HPSI ~~pump and its injection~~ into a water solid RCS with full charging capacity and letdown isolated.
pumps injecting

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.

ATTACHMENT 5

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

   c. Charging pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open and tha recirculation valves ot the refueling water tank close.

f. By verifying that each of the following pumps develops the indicated developed head and/or flow rate when tested pursuant to Specification 4.0.5:

1. High-Pressure Safety Injection pumps developed head, at an indicated flow rate of 650 gpm, greater than or equal to [redacted] 2093 feet for P017, [redacted] 2132 feet for P018 and [redacted] 2099 for P019.

2. Low-Pressure Safety Injection pump developed head greater than or equal to [redacted] 396 feet at mini flow.

3. Charging pump flow rate greater than or equal to 40 gpm.

g. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

1. For High-Pressure Safety Injection pump cold leg injection with a single pump running:

   a. The sum of the injection lines flow rates, excluding the highest flow rate, is greater than or equal to 647 [redacted] gpm for P017 running, 656 [redacted] gpm for P018 running and [redacted] 661 gpm for P019 running, and

   b. The total pump flow rate is greater than or equal to 882 [redacted] gpm for P017 running, [redacted] 894 gpm for P018 running and [redacted] 901 gpm for P019 running.

SURVEILLANCE REQUIREMENTS (Continued)

2. For a single High-Pressure Safety Injection pump hot/cold leg injection.

   a. The sum of the cold leg injection flow rates is greater than or equal to 385 gpm, and

   b. The hot leg injection flow rate is greater than or equal to 385 gpm.

   c. The combined total hot/cold legs injection flow rate is greater than or equal to 896 gpm.

3. For the Low-Pressure Safety Injection pump with a single pump running:

   a. The flow through each injection leg shall be greater than or equal to 3000 gpm when tested individually and corrected to the same pump suction source and leg back pressure conditions. The difference between high and low flow legs shall be less than or equal to 100 gpm.

   b. The total ECCS flow through 2 cold leg injection lines shall be greater than or equal to 4450 gpm when corrected for elevation head.

ATTACHMENT 6

BASES 3/4 9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: 1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, 2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertenly engaged during lifting operation.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure (UGS) with two fingers in each of two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the UGS and lift rig or can be removed with separate tooling prior to UGS removal utilizing the auxiliary hoist of the polar crane.

ATTACHMENT 7

BASES FOR 3/4 10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the shutdown margin requirements of Technical Specification 3.1.1.1.

ATTACHMENT 8

BASES FOR 3/4.1.2 (BORATION SYSTEMS)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

BASES FOR 3/4.5.4 (REFUELING WATER STORAGE TANK)

Change First Paragraph on p. 3/4.5-3 for clarity:

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the minimum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

BASES FOR 3/4.7.1.3 (CONDENSATE STORAGE TANKS)

The OPERABILITY of condensate storage tank T-121 with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for two hours followed by cooldown to shutdown cooling initiation, with steam discharge to atmosphere with concurrent loss of off-site power and most limiting single failure. The OPERABILITY of condensate storage tank T-120 in conjunction with tank T-121 ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 24 hours including cooldown to shutdown cooling initiation, with steam discharge to atmosphere with concurrent loss of off-site power and most limiting single failure. The contained water volume limits are specified relative to the highest auxiliary feedwater pump suction inlet in the tank for T-121, and to the T-121 cross connect siphon inlet for T-120. (Water volume below these datum levels is not considered recoverable for purposes of this specification). Vortexing, internal structure and instrument error are considered in determining the tank levels corresponding to the specified water volume limits.

Prior to achieving 100% RATED THERMAL POWER, Figure 3.7-1 is used to determine the minimum required water volume for T-121 for the maximum power level (hence maximum decay heat) achieved.

ATTACHMENT 9

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

Instruments & Sensor Locations *	Measurement Range	Minimum Instrument Operable
1. Triaxial Time-History Strong Motion Accelerometers		
a. Steam Generator Base Support	-2 to +2g	1
b. Pressurizer Base Support	-2 to +2g	1
c. Reactor Coolant Pump	-2 to +2g	1
d. Containment Base in Tendon Gallery	-2 to +2g	1
e. Containment Operating Level	-2 to +2g	1
f. Unit #1 Free Field	-1 to +1g	1
g. Control Building Basement	-2 to +2g	1
h. Control Building Roof	-2 to +2g	1
i. Safety Equipment Building Base Slab	-2 to +2g	1
j. Safety Equipment Building Piping Support	-2 to +2g	1
k. Radwaste Building Equipment Support	-2 to +2g	1
2. Triaxial Peak Reading Accelerographs		
a. Control Building-Control Room	-2 to +2g	1
b. Control Building Base	-2 to +2g	1
c. Top of Containment Structure	-5 to +5g	1
d. Reactor Coolant Piping	-2 to +2g	1
3. Seismic Triggers		
a. Containment Base in Tendon Gallery	+0.005 to +0.05g	1
b. Containment Operating Level	+0.005 to +0.05g	1
4. Seismic Switches		
a. Steam Generator Base Support	Set pt. 0.45 Horz/0.30 Vert.	1**
b. Containment Base in Tendon Gallery	Set pt. 0.40 Horz/0.50 Vert.	1**
5. Seismic Alarm Annunciator (4a & 4b are sensors)		
a. Control Room Panel L-167		
6. Peak Shock Recorder		
a. Containment Base in Tendon Gallery	2 to 25.4 Hz 1.6 to 90g	1**
7. Peak Shock Annunciator	2 to 25.4 Hz 1.6 to 90g	1
a. Control Room Panel L-167		

* All seismic instrumentation is located in Unit 2 with the exception of item 1.f.

** With control room indication

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS #		CHANNEL CHECK	FUNCTIONAL CALIBRATION	CHANNEL FUNCTIONAL TEST
1.	Triaxial Time-History Strong Motion Accelerometers			
	a. Steam Generator Base Support	M*	R	SA
	b. Pressurizer Base Support	M*	R	SA
	c. Reactor Coolant Pump	M*	R	SA
	d. Containment Base in Tendon Gallery	M*	R	SA
	e. Containment Operating Level	M*	R	SA
	f. Control Building Basement	M*	R	SA
	g. Control Building Roof	M*	R	SA
	h. Safety Equipment Building Base	M*	R	SA
	i. Safety Equipment Building Piping Support	M*	R	SA
	j. Radwaste Building Equipment Support	M*	R	SA
2.	Triaxial Peak Recording Accelerographs			
	a. Control Building-Control Room	N/A	R	N/A
	b. Control Building Base	N/A	R	N/A
	c. Top of Containment Structure	N/A	R	N/A
	d. Reactor Coolant Piping	N/A	R	N/A
3.	Seismic Triggers			
	a. Containment Base in Tendon Gallery	M	R	SA
	b. Containment Operating Level	M	R	S/U***
4.	Seismic Switches			
	a. Steam Generator Base Support	M	R**	SA**
	b. Containment Base in Tendon Gallery	M	R**	SA**
5.	Seismic Alarm Annunciators (4a & 4b are sensors)			
	a. Control Room Panel L-167	M	R	SA
6.	Peak Shock Recorder			
	a. Containment Base in Tendon Gallery	N/A	R**	N/A
7.	Peak Shock Annunciator			
	a. Control Room Panel L-167	N/A	R**	N/A

* Except seismic trigger
** With Control Room indication
*** Need not be performed more frequently than once per 6 months.

All seismic instrumentation is located in Unit 2