TABLE OF CONTENTS (continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ 1.31.

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained at < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 60 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine Pressurizer is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1.1	-------------------NOTE-------------------- Only required to be performed during Pressurizer heatup and cooldown operations. --- Verify Pressurizer heatup and cooldown rates within the following limits: a. A maximum heatup of 200°F in any 1 hour period, b. A maximum cooldown of 200°F in any 1 hour period.	30 minutes
SR 3.4.3.1.2	The spray water temperature differential shall be determined for use in the UFSAR.	For each cycle of auxiliary spray operation and for each cycle of main spray operation when the RCS cold leg temperature is < 500°F.

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Tube Surveillance Program (continued)

4. The provisions of Technical Specifications Surveillance Requirement 3.0.2 are applicable to SG Tube Surveillance inspection frequencies except those established by Category C-3 inspection results.

The above required inservice inspections of SG tubes repaired by sleeving shall be performed at the following frequencies:

1. Steam generator tube sleeves shall be inspected prior to initial operation and in service. The initial operating period before the initial inservice sample inspection shall not be shorter than six months nor longer than 24 months. The inspections of sleeves shall be configured to ensure that each individual sleeve is inspected at least once in 60 months.

2. If the results of the inservice inspection of SG tube sleeves conducted in accordance with Table 5.5.2.11-2 fall in category C-3, the inspection frequency shall be increased to ensure that each remaining sleeve is inspected at least once in 30 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria for Category C-1.

f. Acceptance Criteria

1. Terms as used in this specification will be defined as follows:

a) Degradation - A service-induced cracking, wastage, wear, or general corrosion occurring on either the inside or outside of a tube;

b) Degraded tube - A tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

c) % Degradation - The percentage of the tube wall thickness affected or removed by degradation;

d) Defect - An imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.12 Ventilation Filter Testing Program (VFTP) (continued)

The provisions of Technical Specification Surveillance Requirement 3.0.2 and Technical Specification Surveillance Requirement 3.0.3 are applicable to the VFTP test frequencies.

5.5.2.13 Diesel Fuel Oil Testing Program

This program implements required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a water and sediment content within limits.

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to the storage tanks, with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3; and,

c. Total particulate concentration of fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A.

5.5.2.14 Deleted

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 45.9 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (56.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage rate acceptance criteria are:

a. The Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 9.0 psig.

(continued)

5.7 Reporting Requirements (continued)

5.7.1.4 (Deleted)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Specification 3.1.4, "Moderator Temperature Coefficient;"
2. Specification 3.1.5, "Control Element Assembly (CEA) Alignment;"
3. Specification 3.1.7, "Regulating CEA Insertion Limits;"
4. Specification 3.1.8, "Part Length Control Element Assembly Insertion Limits;"
5. Specification 3.2.1, "Linear Heat Rate;"
6. Specification 3.2.4, "Departure From Nucleate Boiling Ratio;"
7. Specification 3.2.5, "Axial Shape Index;"
8. Specification 3.9.1, "Boron Concentration."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(continued)

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

5. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"

6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)

7. Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.7.1.6 Not Used

5.7.1.7 Hazardous Cargo Traffic Report

Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

(continued)

5.7 Reporting Requirements (continued)

5.7.2 Special Reports

Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

Special Reports shall be submitted to the U. S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, in accordance with 10 CFR 50.4 within the time period specified for each report.

The following Special Reports shall be submitted:

a. When a pre-planned alternate method of monitoring post-accident instrumentation functions is required by Condition B or Condition H of LCO 3.3.11, a report shall be submitted within 30 days from the time the action is required. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to OPERABLE status.

b. Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

c. Following each inservice inspection of steam generator (SG) tubes, in accordance with the SG Tube Surveillance Program, the number of tubes plugged and tubes sleeved in each SG shall be reported to the NRC within 15 days. The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

   1. Number and extent of tubes and sleeves inspected, and

   2. Location and percent of wall-thickness penetration for each indication of an imperfection, and

   3. Identification of tubes plugged and tubes sleeved.

(continued)

5.7 Reporting Requirements (continued)

5.7.2 Special Reports (continued)

Results of SG tube inspections which fall into Category C-3 shall be reported to the NRC prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Each high radiation area as defined in 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.

(continued)

5.8. High Radiation Area (continued)

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty or health physics supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas, where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.