



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

SOUTHERN CALIFORNIA EDISON COMPANY AND

SAN DIEGO GAS AND ELECTRIC COMPANY

DOCKET NO. 50-206

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 91
License No. DPR-13

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Southern California Edison Company and San Diego Gas and Electric Company (the licensees) dated April 12, 1985, as modified July 1 and July 31, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

8510180318 851015
PDR ADDOCK 05000206
PDR

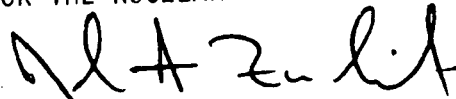
2. Accordingly, the license is amended by deleting paragraph 3.I. of the license and by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Provisional Operating License No. DPR-13 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 91, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to License DPR-13
and to the Technical
Specifications

Date of Issuance: October 15, 1985

ATTACHMENT TO LICENSE AMENDMENT NO.91

PROVISIONAL OPERATING LICENSE NO. DPR-13

DOCKET NO. 50-206

1. Revise License DPR-13 by removing page 3b and inserting the enclosed page 3b.
2. Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE

iii
iv
8
17
33aa
39o
39p
39r
39s
39t
39w
39x
39dd
46e
60
60-G
60-S
60-T
68 through 95

INSERT

iii
iv
8
17
33aa
39o
39p
39r
39s
39t
39w
39x
39dd
46e
60
60g
60s
60t
6-1 through 6-38

[3.I. Deleted]

K. Post-Accident Sampling System (PASS), NUREG-0737, Item II.B.3

- (1) By July 1, 1986 or startup from the Cycle IX refueling outage, whichever is earlier, SCE shall install a PASS and implement a post-accident sampling program at San Onofre Unit 1.
- (2) Prior to the date in (1) above or until the PASS is operable, SCE shall maintain in effect those compensatory measures described in the SCE letter dated August 14, 1984.

Amd. No. 85
11-23-84

	<u>Page</u>
4.5 RADIOACTIVE LIQUID EFFLUENTS	54
4.6 RADIOACTIVE GASEOUS EFFLUENTS	56
4.7 INSERVICE INSPECTION REQUIREMENTS	58
4.8 REACTIVITY ANOMALIES	60
4.9 REACTOR VESSEL SURVEILLANCE PROGRAM	60-A
4.10 AUGMENTED INSERVICE INSPECTION OF HIGH ENERGY LINES OUTSIDE CONTAINMENT	60-B
4.11 CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM	60e
4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES	60g
4.13 TURBINE DECK LOAD BEARING TEST AND VISUAL INSPECTION	60-H
4.14 SHOCK SUPPRESSORS (SNUBBERS) SURVEILLANCE	60-I
4.15 FIRE PROTECTION SYSTEMS SURVEILLANCE	60-L
4.16 INSERVICE INSPECTION OF STEAM GENERATOR TUBING	60-O
4.17 DOSE	60w
4.18 RADIOLOGICAL ENVIRONMENTAL MONITORING	60x
4.19 SOLID RADIOACTIVE WASTE	60cc
 <u>SECTION 5 DESIGN FEATURES</u>	
5.1 SITE DESCRIPTION	61
5.2 CONTAINMENT	62
5.3 REACTOR	65
5.4 AUXILIARY EQUIPMENT	67
 <u>SECTION 6 ADMINISTRATIVE CONTROLS</u>	
6.1 RESPONSIBILITY	6-1
6.2 ORGANIZATION	6-2
FIGURE 6.2-1	6-5
FIGURE 6.2-2	6-6
TABLE 6.2-1	6-7

ADMINISTRATIVE CONTROLS--(continued)

	<u>Page</u>
6.3 UNIT STAFF QUALIFICATIONS	6-8
6.4 TRAINING	6-9
6.5 REVIEW AND AUDIT	6-10
6.6 REPORTABLE EVENT ACTION	6-17
6.7 SAFETY LIMIT VIOLATION	6-18
6.8 PROCEDURES AND PROGRAMS	6-19
6.9 REPORTING REQUIREMENTS	6-22
6.10 RECORD RETENTION	6-27
6.11 RADIATION PROTECTION PROGRAM	6-29
6.12 HIGH RADIATION AREA	6-30
6.13 PROCESS CONTROL PROGRAM	6-32
6.14 OFFSITE DOSE CALCULATION MANUAL	6-33
6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS	6-34
6.16 ENVIRONMENTAL PROTECTION	6-35

3.1 REACTOR COOLANT SYSTEM

3.1.1 MAXIMUM REACTOR COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum activity in the reactor coolant system at any time.

OBJECTIVE: To limit the consequences of an accidental release of reactor coolant to the environment.

SPECIFICATION: The specific activity of the reactor coolant shall be limited to:

1. $\leq 1.0 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$.
2. $\leq 100/\bar{E} \mu\text{Ci/gm}$, where \bar{E} is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant.

ACTION:

- A. With the specific activity of the coolant determined to be $>1.0 \mu\text{Ci/gm}$ but $<60 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$, STARTUP or POWER OPERATION may continue for up to 48 hours provided that operation under these circumstances does not exceed 800 hours in any consecutive 12 month period. Should the total operating time at a reactor coolant specific activity $>1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$ exceed 500 hours in any consecutive six month period, within 30 days the licensee shall report to the NRC, pursuant to Specification 6.9.2, the number of hours of operation above this limit.
- B. With the specific activity of the reactor coolant determined to be $>1 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$ for more than 48 hours during one continuous time interval or $>60 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$ or $>100/\bar{E} \mu\text{Ci/gm}$, be in at least HOT STANDBY with the average temperature of the reactor coolant (T_{avg}) less than 535°F within 6 hours.
- C. With the specific activity of the reactor coolant $>1.0 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$ or $>100/\bar{E} \mu\text{Ci/gm}$, perform the sampling and analysis requirements of item 1a.4.a of Table 4.1.2 until the specific activity of the reactor coolant is restored to within its limits. A Licensee Event Report shall be prepared and submitted to the Commission pursuant to Specification 6.6. This report shall contain the results of the specific activity analysis together with the following information:

3.1.4 LEAKAGE

APPLICABILITY:- Applies to reactor coolant system leakage.

OBJECTIVE: To ensure that leakage from the reactor coolant system does not exceed acceptable limits.

SPECIFICATION: A. The reactor coolant system shall be monitored for evidence of leakage.

B. Detectable leakage from the primary coolant system shall be investigated and evaluated. In any event, if the leakage exceeds 1 gpm and the source of leakage is not identified, the reactor shall be shut down. If the sources of leakage have been identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 6 gpm shall be permitted.

C. The reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of any of the following conditions:

1. An increase in primary to secondary leakage of 140 gpd (0.1 gpm) over a period of twenty-four hours in any steam generator.
2. Any primary to secondary leakage in excess of 215 gpd (0.15 gpm) in any steam generator; or
3. Measured increase in primary to secondary leakage in excess of 15 gpd (0.01 gpm) per day, when measured primary to secondary leakage is above 140 gpd.

Following reactor shutdown, leaking tubes will be repaired or plugged.

D. In addition, in accordance with the Technical Specifications, the reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of primary to secondary leaks in excess of 0.3 gpm in any steam generator. Following reactor shutdown, an eddy current inspection will be performed as required by the Technical Specifications, any leaking steam generator tubes will be repaired or plugged and the NRC notified pursuant to Specification 6.9.2 prior to resumption of plant operation.

TABLE 3.5.10-1 (Continued)ACTION STATEMENTS

- ACTION 25 -** With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, within 1 hour: (1) either initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation, or (2) initiate the preplanned alternate method of monitoring and alarming the area radiation.
- ACTION 26 -** With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.
- ACTION 27 -** With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement of Table 3.5.10-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:
- (1) either restore the inoperable channel(s) to OPERABLE status within 7 days of initiating the preplanned alternate method, or
 - (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2, within 14 days following initiating the preplanned alternate method, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3.15.2 LIQUID EFFLUENT DOSE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the release of radioactive liquid effluents from the site as low as is reasonably achievable.

SPECIFICATION: A. The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

1. During any calendar quarter to ≤ 1.5 mrem to the total body and to ≤ 5 mrem to any organ, and
2. During any calendar year to ≤ 3 mrem to the total body and to ≤ 10 mrem to any organ.

B. Action:

1. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.
2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Basis:

This specification is provided to implement the requirements of Section II.A and IV.A of Appendix I, 10CFR Part 50. Specification A implements the guides set forth in Section II.A of Appendix I. Specification B provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable."

3.15.3 LIQUID WASTE TREATMENT

APPLICABILITY: At all times.

OBJECTIVE: Maintain radioactive releases from the site as low as is reasonably achievable by use of the liquid radwaste treatment system.

- SPECIFICATION:
- A. The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected dose due to the liquid effluent from San Onofre Unit 1, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.
 - B. Action:
 - 1. With radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:
 - a. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reason for inoperability.
 - b. Action(s) taken to restore the inoperable equipment to OPERABLE status.
 - c. Summary description of action(s) taken to prevent a recurrence.
 - 2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Basis:

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10CFR Part 50.36a and the design objective given in Section II.D of Appendix I to 10CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3.16.2 DOSE, NOBLE GASES

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose due to noble gases in gaseous effluents as low as is reasonably achievable.

- SPECIFICATION:
- A. The air dose due to noble gases released in gaseous effluents, from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:
1. During any calendar quarter: ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta radiation.
 2. During any calendar year: ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta radiation.
- B. Action:
1. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.
 2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS:

This specification is provided to implement the requirements of Section II.B and IV.A of Appendix I, 10CFR Part 50. Specification A implements the guides set forth in Section II.B of Appendix I. Specification B provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable."

3.16.3 DOSE, IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose due to radioiodines, radioactive materials in particulate form and radionuclides other than noble gases in gaseous effluents as low as is reasonably achievable.

- SPECIFICATION:
- A. The dose to a MEMBER OF THE PUBLIC from I-131, I-133, from tritium and from all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:
1. During any calendar quarter: ≤ 7.5 mrem to any organ; and
 2. During any calendar year: ≤ 15 mrem to any organ.
- B. Action:
1. With the calculated dose from the release of I-131, I-133, tritium and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.
 2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification is provided to implement the requirements of Sections II.C and IV.A of Appendix I, 10CFR Part 50. Specification A is the guide set forth in Section II.C of Appendix I. Specification B provides the required operating flexibility and at the same time implements the guides set forth in section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable."

3.16.4 GASEOUS RADWASTE TREATMENT

APPLICABILITY: At all times.

OBJECTIVE: Maintain radioactive gaseous releases from the site as low as is reasonably achievable by use of the GASEOUS RADWASTE and VENTILATION EXHAUST TREATMENT SYSTEMS.

SPECIFICATION:

A. The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation over 31 days. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.3 mrem to any organ over 31 days.

B. Action:

1. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:
 - a. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reasons for the inoperability.
 - b. Action(s) taken to restore the inoperable equipment to OPERABLE status.
 - c. Summary description of action(s) taken to prevent a recurrence.
2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS:

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the

3.17 DOSE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose due to the release of radioactive materials within specified limits.

SPECIFICATION: A. The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and to radiation, from uranium fuel cycle sources shall be limited to < 25 mrem to the total body or any organ (except the thyroid, which shall be limited to < 75 mrem).

B. Action:

1. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.15.2.A, 3.16.2.A or 3.16.3.A, calculations should be made to determine whether the above limits of Specification 3.17 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that defines the corrective action to be taken to reduce subsequent releases, to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. The Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Basis: This specification is provided to meet the reporting requirements of 40 CFR 190. In complying with 40 CFR 190, nuclear fuel cycle facilities over five miles away are not considered to contribute to the dose assessment.

3.18 RADIOLOGICAL ENVIRONMENTAL MONITORING

3.18.1 MONITORING PROGRAM

APPLICABILITY: At all times.

OBJECTIVE: Monitor exposure pathways for radiation and radioactive material.

SPECIFICATION: A. The radiological environmental monitoring program shall be conducted as specified in Table 3.18.1.

B. Action:

1. With the radiological environmental monitoring program not being conducted as specified in Table 3.18.1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.
2. With the level of radioactivity as the result of plant effluents in an environmental sampling medium exceeding the reporting levels of Table 3.18.2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report pursuant to Specification 6.9.2. When more than one of the radionuclides in Table 3.18.2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \dots > 1.0$$

When radionuclides other than those in Table 3.18.2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.15.2, 3.16.2 and 3.16.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3. With fresh leafy vegetable samples or fleshy vegetable samples unavailable from one or more of the sample locations required by Table 3.18.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special

3.18.2 LAND USE CENSUS

APPLICABILITY: At all times.

OBJECTIVE: Monitor the UNRESTRICTED AREAS surrounding the site for potential changes to the radiological monitoring program as necessary.

SPECIFICATION: A. A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

B. Action:

1. With the land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.6.3, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new locations. Identify the new locations in the next Semiannual Radioactive Effluent Release Report.
2. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.18.1, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new locations. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31, of the year in which this land use census was conducted.

*Broad leaf vegetation sampling may be performed at the SITE BOUNDARY in the direction sector with the highest D/Q in lieu of the garden census.

$$T = \frac{(22,000 - F_2)}{(F_1 - F)/T_L}$$

- where
- F_1 = measured actuator force from the first Hot SIS test during the current surveillance test (lb_f)
 - F_2 = measured actuator force from the second HOT SIS test during the current surveillance test (lb_f)
 - T_L = time (in days) since the last surveillance testing
 - F = the actuator force from the previous surveillance test (lb_f)*

If the calculated value of T does not exceed 92 days, the next surveillance test must be performed before T days had elapsed.

- c. If the measured actuator force of either HV-851 A or B is greater than 22,000 lb_f, the valve(s) shall be declared inoperable. Test results shall be reported to the NRC pursuant to Specification 6.6 along with proposed corrective actions and NRC approval obtained prior to returning the unit to service.
2. The first test shall be performed not less than 14 days nor more than 21 days following return to power from the current outage which began September 3, 1981.

* For the first surveillance test, the value of F shall be the average actuator force of HV-851 A&B valves from pre-operation testing (3135 lb_f). All subsequent surveillance testing shall assume the F_2 value from the previous surveillance test for each valve. If an F_2 was not required during the previous surveillance test, the F_1 value for each valve shall be assumed.

4.8 REACTIVITY ANOMALIES

APPLICABILITY: Applies to potential reactivity anomalies.

OBJECTIVE: To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION: A. Following a normalization of the computer boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made within 30 days and reported to the NRC pursuant to Specification 6.9.2.

BASIS: To eliminate expected errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safety limit since a reactivity insertion of this amount would not result in pressure or temperature transients which exceed the design conditions of the reactor coolant system.

4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

APPLICABILITY:- Applies to the leakage of radioactive source materials.

OBJECTIVE: To verify the physical integrity of portable and fixed radioactive calibration sources.

SPECIFICATION: A. Byproduct material sealed sources which exceed the quantities listed in 10 CFR 30.71, Schedule B, and all other sealed sources containing greater than 0.1 microcuries shall be leak tested in accordance with Specifications B, C and D below.

Exception: Notwithstanding the periodic leak test required by this specification, any licensed sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material of 10 microcuries or less of alpha emitting material.

B. Each sealed source containing radioactive material, other than Hydrogen 3, with a half life greater than thirty days and in any form other than gas, shall be tested for leakage and/or contamination prior to use out of storage and prior to transfer to another person and thereafter at intervals not to exceed six months. This test does not apply to sealed sources that are stored and not in use.

C. The leakage test shall be capable of detecting the presence of .005 microcuries of radioactive material. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate.

D. If testing reveals the presence of .005 microcuries or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or disposed of in accordance with applicable regulatory requirements and reported in the subsequent annual report filed pursuant to Specification 6.9.1.4.

BASIS:

This Specification assures that leakage from radioactive material sources does not exceed allowable total body or organ limits. In the unlikely event that those quantities of radioactive byproduct materials of interest to this Specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

b. No tube inspected exceeds the plugging limit, plant operation may resume.

3. If, in the inspections performed under Specification B,

a. Less than 10% of the total tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

b. No more than 3 of the tubes inspected exceed the plugging limit,

plant operation may resume after performance of the corrective action in Specification F.

4. If, in the inspections performed under Specification B,

a. More than 10% of the tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, or

b. More than 3 of the tubes inspected exceed the plugging limit,

the situation shall be reported to the Commission in accordance with Technical Specification 6.6 for approval of the proposed remedial action.

5. If in the inspections performed under Specification C.1, wear rates are observed at AVB intersections which are inconsistent with the 50% plugging criterion, the situation shall be reported to the Commission in accordance with Specification 6.6 for approval of the proposed remedial action.
6. If in the inspections performed under Specification C.2, progression of the denting process is observed to be recurring, the situation shall be reported to the Commission in accordance with Specification 6.6 for approval of the proposed remedial action.

F. CORRECTIVE ACTIONS

All leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limit shall be repaired or plugged.

BASIS:

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on 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 degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

- 6.1.1 The Vice President and Site Manager, Nuclear Generation Site shall be responsible for design, construction, operation and maintenance of Unit 1 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility in his absence.
- 6.1.2 The SHIFT SUPERINTENDENT (or during his absence from the Control Room Area,* a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President and Site Manager, Nuclear Generation Site shall be reissued to all site/station personnel on an annual basis.

* "Control Room Area" is defined by the control room and the Shift Superintendent's office.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management technical support shall be as shown on Figure 6.2-1.

UNIT STAFF

6.2.2 The Site organization shall be as shown on Figure 6.2-2 and:

- a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.
- b. At least one licensed Operator shall be at the controls when fuel is in the reactor.* During refueling operations, this operator is permitted to step outside the red line to update the refueling status board. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room Area.**
- c. A health physics technician# shall be on-site when fuel is in the reactor.
- d. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent duties during this operation.
- e. A Fire Brigade of at least five members shall be maintained on site at all times.# The Fire Brigade shall not include the Shift Superintendent and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

* "At the controls" means within the area bounded by the three vertical instrumentation boards and the red line on the floor of the control room.

** "Control Room Area" is defined by the control room and the Shift Superintendent's office.

The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff in the following job classifications:

- 1) Shift Superintendents, Control Room Supervisors, Control Operators, Assistant Control Operators, Nuclear Plant Equipment Operators, Plant Equipment Operators;
- 2) Electricians and their first line supervisors;
- 3) I&C Technicians, Computer Technicians, Test Technicians and their first line supervisors;
- 4) Operational Health Physics Technicians and their first line supervisors;
- 5) Boiler and Condenser Mechanics, Machinists, Welders, Crane Operators and their first line supervisors;
- 6) Contractor or other Department personnel performing functions identical to those performed by personnel identified in items 1 through 5 above and within the organizational framework of the Station.⁽¹⁾

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel identified above work a normal 8-hour day, 40-hour week (excluding shift turnover and meal time) while the plant is operating (MODES 1, 2, 3 and 4). However, in the event that overtime which exceeds 25%⁽¹¹⁾ of normal time is required due to unforeseen problems⁽¹¹¹⁾ or during extended outages^(1v), on a temporary basis, the following guidelines shall be followed:

-
- (1) Shift Technical Advisors are exempt from the overtime guidelines specified, since sleeping accommodations are provided.
 - (11) 25% is established as a level of overtime which will not significantly reduce the effectiveness of personnel, but which requires additional management approval prior to exceeding this level.
 - (111) Unforeseen problems are forced shutdowns or power reductions of any unit, equipment failure or unscheduled repair, surveillance, calibration or maintenance, entry into a Technical Specification ACTION Statement or the absence of personnel required to provide normal shift coverage.
 - (1v) Extended outages are periods in Modes 5 and/or 6.

- 1) An individual should not be permitted to work more than 16 hours straight (shift turnover and meal time are not included when calculating hours worked).
- 2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period (shift turnover and meal time are not included when calculating hours worked).
- 3) A break (the time an individual leaves the work location to the time an individual returns to the work location) of at least 8 hours should be allowed between work periods (shift turnover time is not included when calculating the break; meal time is not included when calculating the break, unless it represents an administrative entry on the timesheet and not extra hours spent at the work location).
- 4) Except during extended shutdown periods, overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Station Manager, the Deputy Station Manager, the Manager, Operations, the Manager, Maintenance, the Manager of Nuclear Generation Services or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

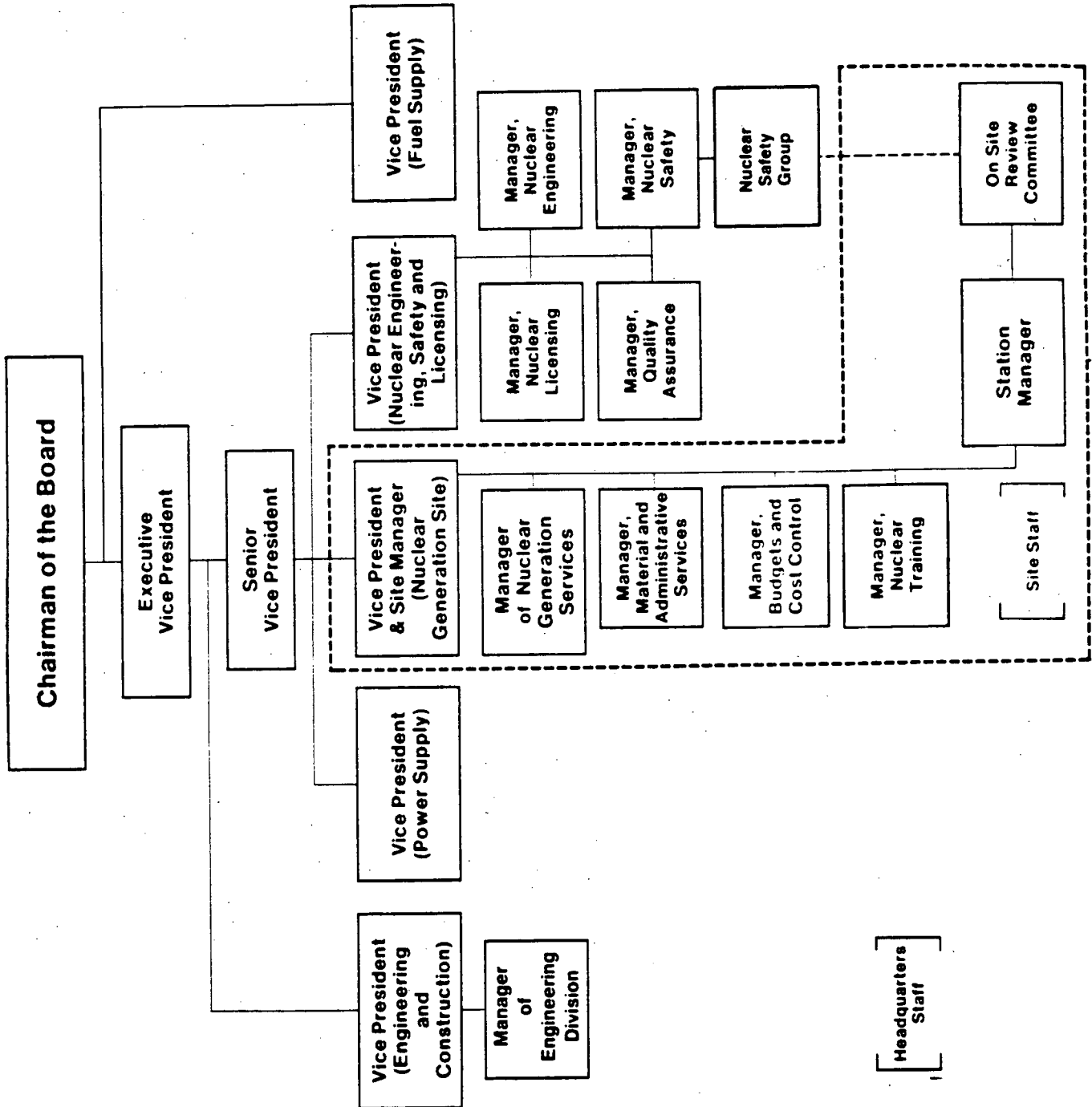


FIGURE 6.2-1
 OFFSITE ORGANIZATION
 SAN ONOFRE NUCLEAR GENERATING STATION — UNIT 1

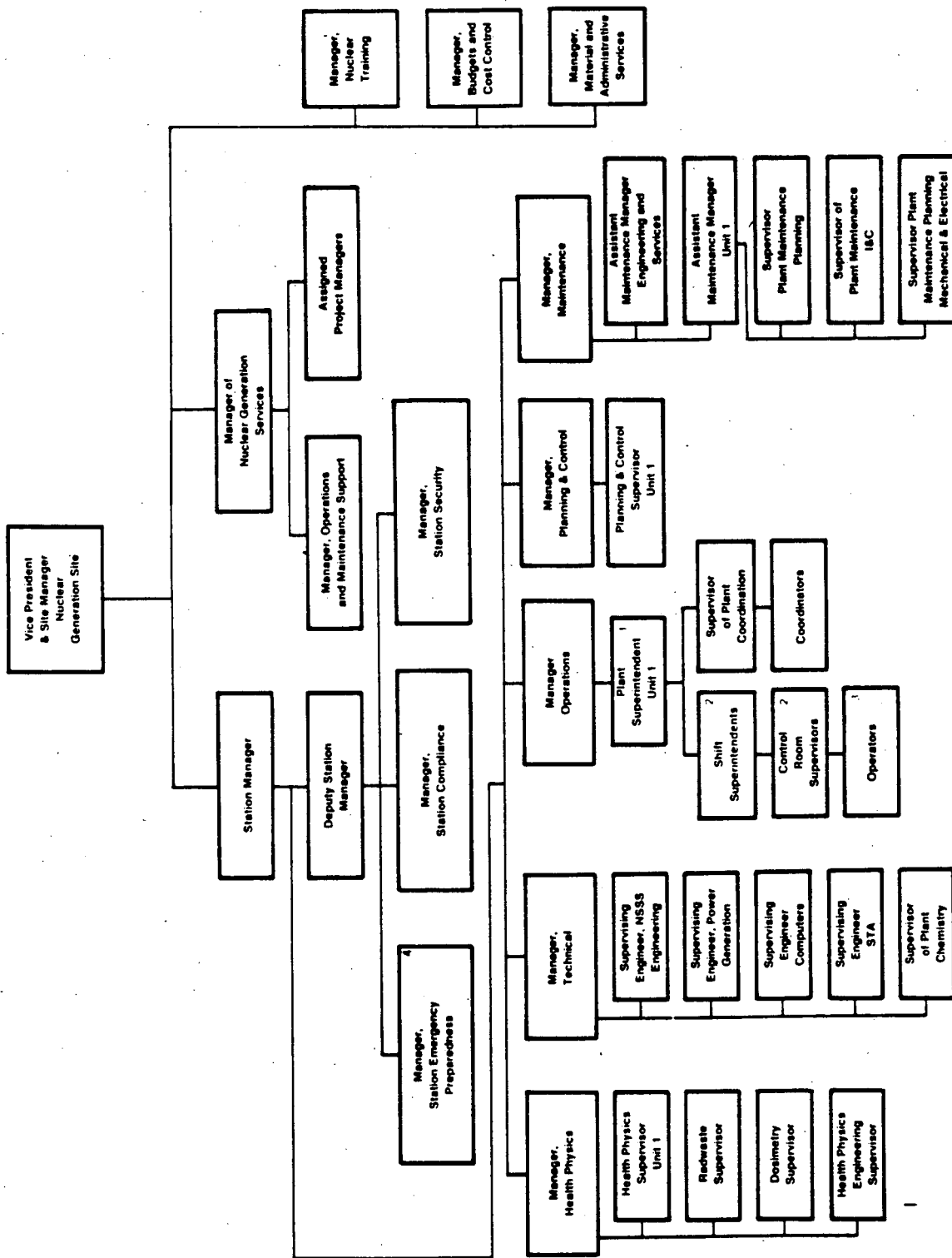


FIGURE 6.2-2
 SITE ORGANIZATION
 SAN ONOFRE NUCLEAR GENERATING STATION

1. At time of appointment to the position Senior Reactor Operator License required
 2. Senior Reactor License required
 3. Control and Assistant Control Operator are holders of Reactor Operator Licenses
 4. Includes fire protection

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1	1
CRS	1	None Required
RO	2	1
NPEO	2	1
STA	1	None Required

- SS - Shift Superintendent with a Senior Reactor Operator's License on San Onofre Unit 1
- CRS - Control Room Supervisor with a Senior Reactor Operator's License on San Onofre Unit 1
- RO - Individual with a Reactor Operator's License on San Onofre Unit 1
- NPEO - Non-Licensed Plant Equipment Operator
- STA - Shift Technical Advisor

Except for the person filling the Control Room Command function, the shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Superintendent from the Control Room Area* while the unit is in MODE 1, 2, 3 and 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room Command function. During any absence of the Shift Superintendent from the Control Room Area* while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

* "Control Room Area" is defined by the control room and the Shift Superintendent's office.

6.3 UNIT STAFF QUALIFICATIONS

- 6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants," for comparable positions, except for the Manager, Health Physics, who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall provide technical support to the Shift Superintendent in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to safe operation of the unit.

6.4 TRAINING

- 6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager, Nuclear Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.
- 6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager, Station Emergency Preparedness and shall meet or exceed the requirements of Section 27 of the National Fire Protection Association Code - 1976.

6.5 REVIEW AND AUDIT

6.5.1 ONSITE REVIEW COMMITTEE (OSRC)

FUNCTION

6.5.1.1 The Onsite Review Committee shall function to advise the Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Onsite Review Committee shall be composed of the:

Chairman:	Station Manager
Member:	Deputy Station Manager
Member:	Manager, Operations
Member:	Manager, Technical
Member:	Plant Superintendent Unit 1
Member:	Supervisor of I&C
Member:	Manager, Health Physics
Member:	Supervisor of Plant Chemistry
Member:	Manager, Maintenance
Member:	Supervising Engineer (NSSS Engineering, Power Generation, Computers, or STA)
Member:	San Diego Gas & Electric Representative, Senior Engineer(1)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the OSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and one-half the remaining membership including alternates.

(1) BS degree in Engineering or Physical Science plus at least four years professional level experience in his field. At least one of the four years experience shall be nuclear power plant experience.

RESPONSIBILITIES

- 6.5.1.6 The Onsite Review Committee shall be responsible for:
- a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).
 - b. Review of all REPORTABLE EVENTS.
 - c. Review of unit operations to detect potential nuclear safety hazards.
 - d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG Supervisor.
 - e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

AUTHORITY

- 6.5.1.7 The Onsite Review Committee (OSRC) shall:
- a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unrevealed safety questions.
 - b. Provide written notification within 24 hours to the Vice President and Site Manager, Nuclear Generation Site and the NSG Supervisor of disagreement between the OSRC and the Station Manager; however, the Vice President and Site Manager, Nuclear Generation Site, shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

- 6.5.1.8 The Onsite Review Committee shall maintain written minutes of each OSRC meeting that, at a minimum, document the results of all OSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Nuclear Safety Group.

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

- 6.5.2.1 The Vice President and Site Manager, Nuclear Generation Site, shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto. Documentation of these activities shall be provided to the NSG.
- 6.5.2.2 Proposed changes to the Appendix "A" Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.
- 6.5.2.3 Proposed modifications to unit nuclear safety related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety related structures, systems and components shall be approved prior to implementation by the Station Manager; or by the Manager, Technical as previously designated by the Vice President and Site Manager, Nuclear Generation Site. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.
- 6.5.2.4 Individuals responsible for review performed in accordance with 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the site/station management staff, previously designated by the Vice President and Site Manager, Nuclear Generation Site, to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary and a verification that the proposed actions do not constitute an unreviewed safety question. If deemed necessary, such review shall be performed by the appropriate designated review personnel.
- 6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager, or members of the site/station management staff previously designated by the Vice President and Site Manager, Nuclear Generation Site. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.

- 6.5.2.6 Recommended changes to the station security plan shall be approved by the Station Manager and transmitted to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG; implementing procedures shall be prepared and approved in accordance with Specification 6.8.
- 6.5.2.7 Recommended changes to the station emergency plan shall be approved by the Station Manager and transmitted to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG; implementing procedures shall be prepared and approved in accordance with Specification 6.8.
- 6.5.2.8 The Station Manager shall assure the performance of a review by a qualified individual/organization of every uncontrolled or unplanned release of radioactivity to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice President and Site Manager, Nuclear Generation Site and to the NSG.
- 6.5.2.9 The Station Manager shall assure the performance of a review by a qualified individual/organization and may designate the approval of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.
- 6.5.2.10 Documentation of each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained.

6.5.3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

- 6.5.3.1 The Nuclear Safety Group shall function to provide independent review and audit of designated activities in the areas of:
- a. nuclear power plant operations
 - b. nuclear engineering
 - c. chemistry and radiochemistry
 - d. metallurgy
 - e. instrumentation and control
 - f. radiological safety
 - g. mechanical and electrical engineering
 - h. quality assurance practices

COMPOSITION

- 6.5.3.2 The NSG shall consist of a Supervisor and at least three staff specialists. The Supervisor shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of 6 years of professional level managerial experience in the power field. Each staff specialist shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of 5 years of professional level experience in the field of his specialty.

The NSG shall use specialists from other technical organizations to augment its expertise in the disciplines of 6.5.3.1. Such specialists shall meet the same qualification requirements as the NSG members.

CONSULTANTS

- 6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

RESPONSIBILITIES

- 6.5.3.4 The NSG shall review:
- a. The safety evaluations for 1) changes to procedures required by Specification 6.8, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

- b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
- c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
- d. Proposed changes in Technical Specifications or this Operating License.
- e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.
- f. Significant operating abnormalities or deviation from normal and expected performance of unit equipment that affect nuclear safety.
- g. All REPORTABLE EVENTS.
- h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components that could affect nuclear safety.
- i. Reports and meeting minutes of the Onsite Review Committee.

AUDIT

- 6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:
- a. The conformance of unit operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.
 - b. The performance, training and qualifications of the entire unit staff at least once per 12 months.
 - c. The results of all actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.
 - d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.
 - e. Any other area of unit operation considered appropriate by the Nuclear Safety Group or the Vice President and Site Manager, Nuclear Generation Site.

- f. The Fire Protection Program and implementing procedures at least once per 24 months.
- g. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.
- h. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUTHORITY

- 6.5.3.6 The NSG shall report to and advise the Manager, Nuclear Safety on those areas of responsibility specified in Sections 6.5.3.4 and 6.5.3.5.

RECORDS

- 6.5.3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be distributed monthly to the Vice President and Site Manager, Nuclear Generation Site, and to the management responsible for the areas audited.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following action shall be taken for REPORTABLE EVENTS:

- a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73, 10 CFR Part 50, and
- b. Each REPORTABLE EVENT shall be reviewed by the OSRC, and the results of this review shall be submitted to the Vice President and Site Manager, Nuclear Generation Site, and the NSG.

6.7 SAFETY LIMIT VIOLATION

- 6.7.1 The following actions shall be taken in the event a Safety Limit is violated:
- a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President and Site Manager, Nuclear Generation Site, and the NSG Supervisor shall be notified within 24 hours.
 - b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the OSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.
 - c. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President and Site Manager, Nuclear Generation Site, and the NSG within 14 days of the violation.
 - d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

- 6.8.1 Written procedures* shall be established, implemented and maintained covering the activities referenced below:
- a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
 - b. Refueling operations.
 - c. Surveillance and test activities of safety related equipment.
 - d. Security Plan implementation.
 - e. Emergency Plan implementation.
 - f. Fire Protection Program implementation.
 - g. PROCESS CONTROL PROGRAM implementation.
 - h. OFFSITE DOSE CALCULATION MANUAL implementation.
 - i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, Revision 1, February 1979.
 - j. "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," June 14, 1977, as specified in Section 6 of the Fire Protection Safety Evaluation Report dated July 19, 1979.
- 6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Vice President and Site Manager, Nuclear Generation Site; or by (1) the Station Manager; or by (2) the Deputy Station Manager; or by (3) the Manager of Nuclear Generation Services; or by (4) Cognizant Managers reporting directly to them as previously designated by the Vice President and Site Manager, Nuclear Generation Site, prior to implementation and shall be reviewed periodically as set forth in administrative procedures.
- 6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:
- a. The intent of the original procedure is not altered.
 - b. The change is approved by two members of the site/station management staff exercising responsibility in the specific area and unit or units addressed by the change, and at least one of whom holds a Senior Reactor Operator's License on the unit affected.

* Procedures and administrative policies shall meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1976, Administrative Controls for Nuclear Power Plants.

- c. The change is documented, reviewed and approved by responsible management, as delineated in 6.8.2 above, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

- (1) Preventative maintenance and periodic visual inspection requirements, and
- (11) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

- (1) Training of personnel
- (11) Procedures for monitoring, and
- (111) Provisions for maintenance of sampling and analysis equipment.

c. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

- (1) Training of personnel, and
- (11) Procedures for monitoring

d. Secondary Water Chemistry Monitoring Program

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

- (1) Identification of a sampling schedule for the critical parameters and control points for these parameters,

- (ii) Identification of the procedures used to measure the values of the critical parameters,
- (iii) Identification of process sampling points,
- (iv) Procedures for the recording and management of data,
- (v) Procedures defining corrective actions for off control point chemistry conditions, and
- (iv) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include:

- (i) Training of personnel,
- (ii) Procedures for sampling and analysis, and
- (iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC Regional Administrator unless otherwise noted.

STARTUP REPORT

- 6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.
- 6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.
- 6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS*

- 6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.
- 6.9.1.5 Reports required on an annual basis shall include a tabulation of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection,

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

** This tabulation supplements the requirements of 10 CFR 20.407.

routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources shall be assigned to specific major work functions.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

- 6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.
- 6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.18.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results, in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map for all sampling locations keyed to a table giving distances and directions from the site reference point; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.18.3.

(Note: Information which may be required by Specifications 3.18.1.B.1, 2, 3.18.3.B.1 and the Basis of 4.18.1 should be included.)

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

- 6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.
- 6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

* A single submittal may be made for multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

** In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release reports shall include the following information for each type of solid waste shipped offsite during the report period:

- a. Container volume,
- b. Total curie quantity (specify whether determined by measurement or estimate),
- c. Principal radionuclides (specify whether determined by measurement or estimate),
- d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottom),
- e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
- f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive release reports shall include unplanned releases from the site to UNRESTRICTED AREAS of radioactive material in gaseous and liquid effluents on a quarterly basis.

The Radioactive Effluent Release Reports shall include any changes made to the PROCESS CONTROL PROGRAM (PCP), to the OFFSITE DOSE CALCULATION MANUAL (ODCM), or major changes to radioactive waste treatment systems during the reporting period.

MONTHLY OPERATING REPORT

- 6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to pressurizer safety and relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Regional Administrator, unless otherwise indicated, within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

- a. Records and logs of unit operation covering time interval at each power level.
- b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
- c. All REPORTABLE EVENTS submitted to the Commission.
- d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
- e. Records of changes made to the procedures required by Specification 6.8.1.
- f. Records of radioactive shipments.
- g. Records of sealed source leak tests and test results, in units of microcuries, for leak tests performed pursuant to Specification 4.12.
- h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

- a. Record and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.
- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Records of radiation exposure for all individuals entering radiation control areas.
- d. Records of gaseous and liquid radioactive material released to the environs.
- e. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

- f. Records of reactor tests and experiments.
- g. Records of training and qualification for current members of the unit staff.
- h. Records of in-service inspections performed pursuant to these Technical Specifications.
- i. Records of Quality Assurance activities not included in 6.10.1 that are required by the QA Manual.
- j. Records of reviews performed for changes made to procedures or equipment or reviews or tests and experiments pursuant to 10 CFR 50.59.
- k. Records of OSRC meetings and NSG reports.
- l. Records of the service lives of all safety related hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.
- m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr 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 which continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device which continuously integrates the radiation dose rate in the areas and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.
- c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Exposure Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Superintendent on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem^{**} that are located within large areas,

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following approved plant radiation protection procedures for entry into high radiation areas.

** Measurement made at 18" from source of radioactivity.

such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and unless a health physics technician is in continuous attendance, a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the changes(s) was made effective. This submittal shall contain:
 - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;
 - b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
 - c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.9.
2. Shall become effective upon review and approval pursuant to 6.5.2.9.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:
 - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);
 - b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and
 - c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.9.
2. Shall become effective upon review and approval pursuant to 6.5.2.9.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS* (Liquid, Gaseous and Solid)

6.15.1 Licensee-initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective pursuant to 6.5.2.9. The discussion of each change shall contain:
 - a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;
 - b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
 - c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;
 - d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
 - e. An evaluation of the change which shows the expected maximum exposures to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;
 - f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;
 - g. An estimate of the exposure to plant operating personnel as a result of the change; and
 - h. Documentation of the fact that the change was reviewed and found acceptable pursuant to 6.5.2.9.
2. Shall become effective upon review and approval pursuant to 6.5.2.9.

* Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.16 ENVIRONMENTAL PROTECTION

FACILITY DESIGN AND OPERATION

6.16.1 This section contains a description of facility design features and operating practices which, if changed, could have a significant effect on environmental impact. Any significant change in facility design features or operating practices described here must be reported to the NRC in accordance with the provisions of Section 6.16.2.a prior to the change.

a. Intake System

The circulating water system, under normal operating conditions, draws water from the ocean at a point approximately 3,200 feet offshore. The ocean bottom at this point is approximately 27 feet below mean lower low-water level. The intake structure rests on a foundation located 33 feet beneath the ocean bottom and rises vertically to a point 10 1/2 feet above the ocean floor. The inside horizontal dimensions of the intake structure are 16 to 21 feet. A velocity cap, 1-foot thick, rests on eight columns above the top of the intake structure. The top surface of the velocity cap is 15 1/2 feet above the ocean bottom and 11 1/2 feet below mean lower low-water.

A 12-foot ID reinforced concrete conduit is connected horizontally to the shoreward side of the intake structure. This conduit is buried beneath the ocean bottom, with a minimum of 4 feet of sand cover over its top and 4 feet of rock cover surrounding the intake structure. All sand cover was placed so as to approximate the local ocean bottom profile.

Water entering the top of the intake structure is accelerated to a design velocity of about 2.5 feet per second and directed into a 12-foot ID reinforced concrete conduit. As the water enters the concrete conduit from the structure, it is accelerated to a design velocity of 6.9 feet per second. The circulating water system is designed to deliver 350,000 gpm at this velocity.

The offshore system joins the onshore portion of the circulating water system at the screenwell. The screenwell is located just inside the seawall on the Station property.

Cooling water entering the onshore system passes, through a coarse bar screen, through finer traveling screens, and proceeds to two circulating water pumps designed to operate at 175,000 gpm. Water entering the screenwell structure is decelerated so that the approach velocity at the screens is approximately 2.0 feet per second.

The circulating water system uses three methods of handling the marine growth and debris associated with the flow of seawater through the plant condensers. These are heat treating, bar and traveling screens, and chlorination.

Heat treatment is used for incrustation control. This method consists of reversing the flow in the intake conduit and adjusting the temperature of the water to approximately 100°F and maintaining this temperature for approximately two hours once every five to six weeks and occasionally once every four weeks, and discharging through the intake conduit. This is accomplished by recirculating a portion of the condenser discharge back through the condenser. Cross-connections between intake and outfall conduits are provided to create the reversal of flow necessary for the treatment of the conduits. Normally only the intake conduit is treated. The water temperature in the outlet conduit can be raised for treatment when necessary. The sudden temperature increase of the cooling water causes incrustations growing in the circulating water system to expire, relax their hold, and be flushed out of the system.

Traveling and bar screens are provided to remove marine growth and debris from the seawater passing through the screenwell. The materials removed from the seawater are marine growth, shells, fish, driftwood, and other debris present in the ocean.

For chlorination, sufficient sodium hypochlorite is injected into the circulating water upstream of the circulating water pumps three times a day for each condenser half to eliminate slime-forming organisms on condenser internal surfaces.

The traveling screens and bar screens are placed in series, perpendicular to the flow. The screens are cleaned automatically, with the frequency of cleaning being dependent on the rate of material buildup on the screens. The bar screens are cleaned by a traveling mechanical rake that deposits accumulated debris, by means of a seawater jet spray washing process, into sluiceways for removal. The traveling screens are motor driven, and are capable of rotating as a unit in continuous sequence when activated by pressure differential due to trash buildup. The debris picked up by the traveling screens is also deposited in a sluiceway by means of a seawater jet spray.

b. Discharge System

Under normal operating conditions, the heated cooling water leaves the condenser and is discharged to the ocean through a 12-foot ID 2,600-foot-long concrete conduit. A single point discharge is effected through a discharge structure located in 24 feet of water. The dimension of the structure is the same as the intake; however, there is no velocity cap. The top of the discharge structure is about 11.5 feet below mean lower low-water.

A 12-foot ID reinforced concrete conduit is connected horizontally to the shoreward side of the discharge structure. This conduit is buried beneath the ocean bottom, with a minimum of 4 feet of sand cover over its top and 4 feet of rock surrounding the discharge structure. All sand and rock cover was placed so as to approximate the local ocean bottom profile.

The water travels through the discharge conduit with a design velocity of 6.9 feet per second and exits with a vertical velocity of about 2.5 feet per second. The vertical orientation creates a single orifice jet diffuser which entrains surrounding cooler water and assists in rapid diminution of the discharge temperature. About seven minutes is required for water to travel from the condensers to the end of the discharge.

c. Land Management

The facility occupies about 16 acres of the 84 acre site.

No use of herbicides is practiced to manage vegetation along the transmission line except in isolated cases to meet property owners' requests or permit stipulations from public agencies. Standard erosion control measures are used to minimize erosion at the facility, at tower sites, and along access roads.

REPORTS

6.16.2 The following reports shall be submitted pursuant to Specification 6.9.2, with a copy to the Director, Office of Nuclear Reactor Regulation.

- a. A report shall be made to the NRC prior to implementation of a change in plant design, in plant operation, or in procedures described in Section 6.16.1 if the change would have a significant adverse effect on the environment or involves an environmental matter or question not previously reviewed and evaluated by the NRC. The report shall include a description and evaluation of the change and a supporting benefit-cost analysis.

b. Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to station operation shall be recorded and promptly reported to the NRC within 24 hours followed by a written report within 30 days. No routine monitoring programs are required to implement this condition.

The written report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

The following are examples of unusual or important events: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; unusual fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substances.

c. Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES Permit or State certification (pursuant to Section 401 of the Clean Water Act) shall be reported to the NRC by submittal of copies of the reports required by the NPDES Permit or certification. The licensee shall also provide the NRC with a copy of the results of the following studies at the same time they are submitted to the permitting agency:

Section 316(b) Demonstration Study

Changes and additions to the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.