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

August 10, 1995

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NUCLEAR LICENSING

Mr. Harold B. Ray
Senior Vice President
Southern California Edison Company
23 Parker Street
Irvine, CA 92718

Dear Mr. Ray:

Enclosed is the final draft of the NRC safety evaluation report (SER) on your license amendment request PCN-299 dated December 30, 1993, as supplemented by letters dated June 3, 1994, August 25, 1994, and January 3 and 19, 1995, for conversion of the San Onofre Nuclear Generating Station (SONGS) Unit 2 and Unit 3 technical specifications (TS) to the improved standard technical specifications (STS). The final draft SER is provided for your review to verify the accuracy of the technical basis for the conversion to the STS and to prepare the corresponding certified technical specifications to be issued with the license amendment. The change in frequency of SR 3.7.11.1, discussed in section 2.3.7.b of the SER, is contingent upon NRC approval of SONGS PCN-407.

Southern California Edison has advised the staff that you will not be prepared to implement these improved TS until early in 1996. The NRC believes that the conversion amendment should be issued at the earliest possible date, with an effective date that is consistent with your implementation schedule. This action is necessary to minimize the burden on the NRC staff while you are developing the implementation plan. After the issuance of the conversion amendment, subsequent license amendment requests must include justifications for changes to both the existing and improved TS until the effective date of the conversion amendment.

Accordingly, we request that within thirty days following your receipt of this letter, you provide a schedule for the submittal of the certified technical specifications, issuance of the conversion amendment, effective date of the

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415-1187

amendment, and any specific implementation dates that must be included in the amendment. Should you have any questions regarding this request, please contact the Project Manager for your facility, Mr. Mel Fields, at 301-415-3062.

Sincerely,

William H. Bateman

William H. Bateman, Director
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-361
and 50-362

Enclosure: As stated

cc w/encl: see next page

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

DRAFT

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. AND TO

FACILITY OPERATING LICENSE NOS. NPF-10 AND NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 and 50-362

1.0 INTRODUCTION

San Onofre Nuclear Generating Station (SONGS) has been operating using Technical Specifications (TS) issued with the original operating licenses (Nos. NPF-10 and NPF-15) on February 16 and November 15, 1982, as amended over the years. By letter dated December 30, 1993, and as supplemented by letters dated June 3, 1994, August 25, 1994, and January 3 and 19, 1995, Southern California Edison Company (SCE) proposed to amend Appendix A of NPF-10 and NPF-15 to revise, in their entirety, the TS for SONGS Units 2 and 3. The proposed amendments were based on guidance provided by the U.S. Nuclear Regulatory Commission (NRC) in its interim "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," published in the Federal Register (52 FR 3788) on February 6, 1987. The overall objective of the proposed amendments, consistent with the NRC's "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (58 FR 39132), issued on July 22, 1993, was to completely rewrite, reformat, and streamline the existing SONGS TS.

SCE's proposal to revise the SONGS TS follows closely the example provided in NUREG-1432, "Standard Technical Specifications - Combustion Engineering Reactors," September 1992. From 1989 through 1992, the utility Owners Groups and the NRC staff developed improved standard technical specifications (STS) that would establish models of the Commission's policy for each primary reactor type. The Commission's final policy statement described the safety benefits of the improved STS, and encouraged licensees to use the improved STS as the basis for plant-specific TS, particularly complete conversions to the improved STS. Moreover, the policy statement provided guidance to evaluate the scope of the TS. The guidance in the interim and final policy statement was used to develop NUREG-1432, which serves as a model for developing

improved TS for Combustion Engineering plants. The interim and final policy statements reflect the Commission's view that satisfying the guidance in the policy statements also satisfies Section 182a of the Atomic Energy Act and 10 CFR 50.36.

NUREG-1432 was established as the model for CE plants in general and the improved SONGS TS in particular. Portions of the existing TS were also used as a basis for the improved SONGS TS. Plant-specific issues (including unique plant design features, requirements, and operating practices) were discussed with the licensee during a series of meetings concluded on May 25, 1994. In addition, meetings were held with the Owners Groups to discuss matters of a generic nature that were considered for changes to NUREG-1432. Those generic issues not incorporated into NUREG-1432 were considered for specific applications in the improved SONGS TS.

Changes in the licensee's proposed TS that resulted from discussions with the licensee during the review are discussed in the following sections. These plant-specific changes clarify the TS with respect to the guidance in the Commission's policy statement and the guidelines in NUREG-1432, but do not affect the intent of the specifications. Therefore, the changes are within the scope of the NRC's proposed action on the amendment request, which was published in the Federal Register (59 FR 49434) on September 28, 1994. The Commission's policy statement envisioned that the TS conversion would result in transferring some TS requirements to other licensee-controlled documents. The staff emphasized human factors principles to clarify the improved SONGS TS, and to define more clearly the appropriate scope of the TS. In addition, the staff proposed significant changes to the Bases section of the SONGS TS to enhance and clarify each specification.

This safety evaluation (SE) documents the basis for the staff's conclusion that SCE can convert its existing TS to those based on NUREG-1432, modified by plant-specific changes, and that the use of these improved SONGS TS is acceptable for continued operation of SONGS Units 2 and 3. Individual section topics and the corresponding section numbers are identical to those given in NUREG-1432. The staff has identified the changes to the existing SONGS TS and has explained the significant changes in this evaluation. The staff also acknowledges that, in accordance with the Commission's final policy statement, the conversion to the STS is a voluntary process. Therefore, the improved TS for SONGS Units 2 and 3 reflect some differences that correspond to the existing licensing basis for the units. Since SONGS Units 2 and 3 are identical, this SE applies to both units, except as noted in the text.

Since its original submittal of December 30, 1993, SCE has submitted, and the staff has accepted, a number of changes to the existing SONGS TS. The review and approval of these TS amendments were independent of the improved TS review effort. These changes are reflected, as appropriate, in the improved SONGS TS. This SE describes only those TS amendment changes that affected implementation of the improved SONGS TS.

During its review, the staff relied on the NRC's interim policy statement and later on NUREG-1432, which was issued in September 1992. For the reasons stated elsewhere in this SE, the staff finds that: (1) the proposed improved TS satisfy the requirements in Section 182a of the Atomic Energy Act and 10 CFR 50.36, and the guidance in the Commission's final policy statement; (2) that they are consistent with the common defense and security; and (3) that they adequately protect the health and safety of the public.

2.0 BACKGROUND AND EVALUATION

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses shall state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In Title 10, Section 50.36, of the Code of Federal Regulations (10 CFR 50.36), the Commission established its regulatory requirements related to the content of TS. In so doing, the Commission emphasized those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968). Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories:

(1) safety limits, limiting safety system settings, and limiting control settings
(2) limiting conditions for operation
(3) surveillance requirements
(4) design features
(5) administrative controls

Section 50.36 of 10 CFR Part 50 does not specify, however, the particular requirements to be included in a plant's TS.

For several years, the NRC and industry representatives sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors." From 1989 through 1992, the utility Owners Groups and NRC staff developed improved STS that would establish models of the Commission's policy for each primary reactor type. In

addition, the staff, licensees, and the Owners Groups developed generic administrative and editorial guidelines (in the form of a "Writers Guide" for technical specifications), which significantly enhance human factors considerations and were used throughout the development of licensee-specific improved TS.

In September 1992, the Commission issued NUREG-1432, which was established as a model for developing improved TS for CE plants in general, and for the improved SONGS Unit 2 and 3 TS in particular. NUREG-1432 reflects the results of a detailed review of the application of the Commission's interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the Nuclear Steam Supply System (NSSS) Owners Groups in May 1988. NUREG-1432 also reflects the results of extensive discussions on various drafts of STS, so that the application of the TS criteria and the "Writers Guide" would consistently reflect detailed system configurations and operating characteristics for all NSSS designs. As such, the generic Bases presented in NUREG-1432 provide an abundance of information regarding the extent to which the STS present requirements that are necessary to protect the health and safety of the public.

On July 22, 1993, the Commission issued its final policy statement, in which the Commission expressed its view that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act and 10 CFR 50.36. The final policy statement described the safety benefits of the improved STS, and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments and complete conversions to improved STS.

Further, the final policy statement provided guidance to evaluate the required scope of the TS, and finalized the criteria to be used in determining which design conditions and associated surveillances need to be addressed in the TS. The Commission noted in its final policy statement that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

In accordance with this approach, existing TS requirements that fall within or satisfy any of the criteria in the final policy statement should be retained in the TS, while those TS requirements that do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents. The final policy statement criteria are as follows:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(2) A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(3) A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(4) A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1]

In its license amendment application, the licensee proposed changes to existing TS requirements using the final policy statement and NUREG-1432 as guidance. The licensee also proposed changes to NUREG-1432, because of differences between the plant-specific licensing basis and the design basis provided in the Bases section of NUREG-1432.

In this SE, the licensee's proposed changes to its existing TS requirements are grouped into four general categories, including: administrative or non-technical changes, relocated requirements (that is, requirements moved from an existing NRC-controlled TS to specified licensee-controlled documents); more restrictive requirements (i.e., additions to existing TS) and less restrictive requirements (i.e., relaxations to or deletions from existing TS requirements). These four general categories of changes to the licensee's existing TS requirements are described in the following paragraphs.

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[1] The Commission recently adopted amendments to 10 CFR 50.36, pursuant to which the rule was revised to codify and incorporate these criteria. See Final Rule, "Technical Specifications," 60 Fed. Reg. 36953 (July 19, 1995). The Commission indicated that reactor core isolation cooling, isolation condenser, residual heat removal, standby liquid control, and recirculation pump are to be included in the TS under Criterion 4, although it recognized that other structures, systems and components could also meet this criterion. 60 Fed. Reg. at 36956.

Administrative Changes

Non-technical, administrative changes were intended to incorporate human-factors principles into the form and structure of the improved plant TS so that they would be easier to use for plant operations personnel. These changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content or operational requirements. Every section of the proposed TS reflects this type of change. In order to ensure consistency, the NRC staff and SCE have used NUREG-1432 as guidance to reformat and make other administrative changes. The licensee proposed such changes as (a) providing the appropriate numbers for NUREG-1432 bracketed information (information that must be supplied on a plant-specific basis, and that may change from plant to plant), (b) identifying plant-specific wording for system names, and (c) changing NUREG-1432 wording to conform to existing licensee practices.

The staff has reviewed all of the administrative changes proposed by SCE (or the licensees in general) and finds them acceptable, since they are consistent with the Commission's regulations and are compatible with the "Writers Guide" and NUREG-1432. The more significant non-technical administrative changes are discussed individually in this SER.

Relocated Requirements

As summarized above, the Commission's policy statement allows that existing TS requirements that do not satisfy or fall within any of the four specified criteria may be relocated to appropriate licensee-controlled documents. In the licensee's application, such requirements are generally relocated to the Updated Final Safety Analysis Report (UFSAR) and the Licensee-Controlled Specifications (LCS), which the licensee has committed to incorporate in UFSAR Chapter 16 by reference, prior to implementation of this amendment, by letter dated [ ]. Unless otherwise specified in this SE, the portion of the existing TS related to limiting conditions for operation (LCO) which includes the system description, design limits, functional capabilities, and performance levels, will be relocated to the LCS. The relocated provisions of the existing TS action statements and surveillance requirements will be relocated to appropriate plant procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures), depending on the nature of the requirements being relocated. The requirements that are being relocated from the existing plant TS to licensee-controlled documents are summarized in Table 1.

Any time the operability of a system or component has been affected by repair, maintenance, or replacement of a component, plant procedures require that a post-maintenance test be performed to demonstrate operability of the system or component. Various post-maintenance surveillance requirements scattered throughout the existing TS have been relocated from the improved TS to the LCS, UFSAR or plant procedures. In addition, the details and methods of operation of a system during the performance of a surveillance have been relocated from the existing TS to the LCS, UFSAR, Bases or plant procedures. Examples include descriptions of tests to ensure that system controls are

operable, controls during functional testing of components, and setpoint verification that inherently performs a functional test of the instruments and the cycling of valves. These procedures will also be described in the UFSAR.

The facility and procedures described in the UFSAR and LCS can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures an auditable and appropriate control over the relocated requirements and any future changes to these provisions. Other licensee-controlled documents include provisions for making changes consistent with other applicable regulatory requirements; for example, the Offsite Dose Calculation Manual (ODCM) can be changed in accordance with 10 CFR Part 20; the Emergency Plan Implementing Procedures (EPIP) can be changed in accordance with 10 CFR 50.54(q); and the administrative instructions that implement the Topical Quality Assurance Manual (TQAM) can be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. Temporary procedure changes are also controlled by 10 CFR 50.54(a).

Although the UFSAR already includes most of the design information described above, the licensee committed, in their improved STS submittal letter dated December 30, 1992, to confirm that these details are appropriately reflected in the UFSAR or LCS. The licensee will also maintain an auditable record of, and an implementation schedule for, the procedure changes associated with the development of the improved plant-specific TS. The Licensee will maintain documentation of these changes in accordance with the record retention requirements specified in the SONGS TQAM of December 26, 1976, and the LCS.

As described in more detail throughout this evaluation, the staff concludes that the licensee has identified, as summarized in Table 1, appropriate controls for all of the requirements that are being relocated from the SONGS TS to licensee-controlled documents. Until incorporated in the UFSAR and procedures, changes to the provisions being relocated from the TS will be controlled in accordance with the applicable existing procedures. The NRC will conduct an audit of the relocated requirements following implementation to ensure that an appropriate level of control has been achieved. The staff concludes that, in accordance with the Commission's policy statement, sufficient regulatory controls exist for relocated requirements under the regulations, particularly 10 CFR 50.59. Accordingly, the staff concludes that these requirements, as described in detail in this evaluation, may be relocated from the TS to the UFSAR or to other licensee-controlled documents as specified herein.

More Restrictive Requirements

The licensee's proposed improved TS include certain requirements that are more restrictive than those contained in the existing TS. In some cases, these are more conservative than the corresponding requirements in the existing TS. In other cases, they represent the addition of restrictions contained in NUREG-1432 but not contained in the existing TS. The licensee's more restrictive requirements include the following examples:

- placing an LCO on plant equipment that is not required to be operable by

the present TS

- more restrictive requirements to restore inoperable equipment
- more restrictive surveillance requirements

These more restrictive requirements are discussed individually throughout this evaluation.

Less Restrictive Requirements

Less restrictive requirements are justified on a case-by-case basis as discussed in this evaluation. When requirements have been shown to provide little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of: generic NRC actions, new NRC staff positions that evolved from technological advancements and operating experience, or resolution of Owners Group comments on the improved STS. Generic relaxations contained in NUREG-1432 were reviewed by the staff and found to be acceptable because they are consistent with current licensing practices and NRC regulations. The staff also reviewed the licensee's design to determine if the specific design and licensing bases are consistent with the technical basis for the model requirements in NUREG-1432, and thus provide a basis for the revised TS.

One generic change that has been made in the STS, and is reflected in the improved TS proposed for SONGS Units 2 and 3, is the change from an 18-month to a 24-month surveillance interval associated with longer refueling cycles. This change is reflected in NUREG-1432, consistent with the guidance in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." Other generic changes that have been incorporated in the STS include: NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Lists from TS" (such as, the use of the Core Operating Limits Report (COLR)), and NUREG-1366, "Improvements to TS Surveillance Requirements," which justifies changes to surveillance requirement (SR) frequencies.

The following sections explain how the staff concluded that the conversion of the existing SONGS TS to those based on NUREG-1432, as modified by plant-specific changes, is consistent with the current SONGS licensing basis, applicable regulatory requirements, and guidance of the Commission's policy statement, and is therefore acceptable.

2.1 Use and Application (Section 1.0)

The definitions appearing in Section 1.0 of the improved SONGS TS have been reorganized from the existing SONGS TS by deleting the identification numbers associated with each definition and listing them in alphabetical order.

The following definitions have all been retained in the improved SONGS TS:

- ACTIONS
- AXIAL SHAPE INDEX
- AZIMUTHAL POWER TILT
- CHANNEL CALIBRATION
- CHANNEL CHECK
- CHANNEL FUNCTIONAL TEST
- CORE ALTERATION
- DOSE EQUIVALENT I-131
- $\bar{E}$ AVERAGE DISINTEGRATION ENERGY
- ENGINEERED SAFETY FEATURE RESPONSE TIME
- TYPES OF LEAKAGE
- MODE (including Table 1.1)
- OPERABLE-OPERABILITY
- PHYSICS TESTS
- RATED THERMAL POWER
- REACTOR PROTECTION SYSTEM RESPONSE TIME
- SHUTDOWN MARGIN
- STAGGERED TEST BASIS
- THERMAL POWER

Some editorial changes have been made so that these defined terms are consistent with NUREG-1432 and with plant-specific terminology. The modifications have been accepted by the licensee and, based on staff review, do not change the intent of the definitions as found in NUREG-1432 and in the existing SONGS TS. Therefore, the staff finds these definitions acceptable for the improved SONGS TS.

The following new definition has been added:

CORE OPERATING LIMITS REPORT (COLR)

The addition of this new definition is compatible with changes made throughout the improved SONGS TS to clarify the related requirements, and to reduce the likelihood of misinterpretation of the new TS. This new SONGS definition was also defined in NUREG-1432. Plant-specific wording differences have been reviewed, and do not change the meaning of these definitions.

All other definitions in the existing SONGS TS (1.7, 1.13, 1.14, 1.16, 1.20, 1.22, 1.23, 1.26, 1.28, 1.29, 1.30, 1.34, 1.35, and Table 1.2) are no longer used as defined terms in the improved SONGS TS. However, definitions 1.13 and Table 1.2 have been reformatted, and these concepts are contained in Section 1.4 of the improved SONGS TS.

As noted above, the staff and the licensee have agreed to minor word changes throughout the SONGS definition section. These word changes are clarifications that do not alter the meaning of the definitions or change the restrictive level of the TS. The definitions in Section 1.0 of the improved SONGS TS support other sections in the improved SONGS TS. The staff has reviewed the proposed changes in the definition section for their effect on

the Safety Limits (SLs) and SL violations that appear in Section 2.0 and the Limiting Conditions for Operation (LCOs) and Action Statements in Section 3, including the Surveillance Requirements (SRs). The staff finds no adverse effects of the proposed changes and concludes that when the definitions, as modified, are applied in other sections of the TS, the restrictive level of the requirements is maintained and, therefore, the safety margins are not affected. In addition, the staff concludes that the licensee's proposed changes clarify the definitions and would reduce the tendency for misinterpretation. The staff finds that the proposed improved SONGS TS appropriately apply the guidance provided in NUREG-1432, and are acceptable.

2.1.1 Logical Connectors (Section 1.2)

This is a new section in the improved SONGS TS. This section explains the meaning and use of "logical connectors" using examples to clarify the entire improved SONGS TS from a human factors standpoint. The staff has reviewed this section and considers this proposed addition and reformatting an enhancement to the improved SONGS TS. The staff finds the addition to be consistent with NUREG-1432 and therefore acceptable.

2.1.2 Completion Times (Section 1.3)

This is a new section in the improved SONGS TS. This section does not change completion times, but provides guidance through the use of examples. "Completion time" is the amount of time allowed to complete an action or the amount of time allowed for a structure, system, or component to be inoperable. This section is administrative in nature and is provided as an aid to the licensee's staff. The NRC staff has reviewed this section, and finds it consistent with NUREG-1432, and therefore acceptable.

2.1.3 Frequency (Section 1.4)

This is a new section in the improved SONGS TS. This section defines the proper use and application of surveillance frequency practices using examples to provide a clear understanding of the correct application of a specified frequency. Such an understanding is provided to ensure compliance with an SR.

The NRC staff has reviewed this section and finds that the "frequency notation" definition and the "frequency notation table" (Definition 1.13 and Table 1.2, respectively) of the existing TS have been adequately incorporated into the descriptions and examples of this section. We find that this section is consistent with NUREG-1432 and therefore acceptable.

2.2 Safety Limits (Section 2.0)

This section has been renamed from the existing SONGS TS, Section 2.0 "Safety Limits and Limiting Safety System Settings." Although renamed, this section contains essentially the same information as the existing Section 2.0. Information not retained in this section is contained elsewhere within the improved SONGS TS or other licensee documents.

This section has been reformatted and reorganized to separate the safety limits and safety limit violations. The staff has reviewed SCE's proposed Section 2.0, based on NUREG-1432, as modified to include plant-specific limits and terminology, and finds this section is consistent with the Commission's regulations and is acceptable.

In accordance with the guidance in NUREG-1432, the licensee has proposed to relocate all or portions of the existing TS 2.2, "Limiting Safety System Settings," to the instrumentation specifications in Section 3.3 to bring it into conformance with the improved TS. Existing TS 2.2 specifies the reactor protection system (RPS) instrumentation setpoints and allowable values. The improved TS address these items within TS 3.3.1, "Reactor Protection System (RPS) Instrumentation-Operating," TS 3.3.2, "Reactor Protection System (RPS) Instrumentation-Shutdown," and TS 3.3.4, "Reactor Protection System (RPS) Logic and Trip Initiation."

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. The LSSS are defined in the specifications as the "allowable values," which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including SLs, during design basis accidents (DBAs). Nominal trip setpoints are specified in the setpoint calculations and in the LCS. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the allowable value between successive channel calibrations. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its allowable value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required allowable value. Thus, the RPS setpoints are effectively retained within the improved TS. The allowable values for the reactor protective instrumentation setpoints are specified in the improved TS 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating" and TS 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown." The Manual Reactor Trip requirement is relocated to TS 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation." This is considered an administrative change in the location of the requirements within the TS, and is therefore acceptable.

The following LSSS of existing TS 2.2 have been relocated from the TS to the LCS. Neither the trip setpoints nor the associated instrumentation (as described in Section 2.3.3) are relied upon to prevent or mitigate design basis accidents or transients. The trip setpoints are not directly related to safety limits or significant safety functions, and therefore, these setpoints will be relocated to the LCS with the associated instrumentation, and subject to 10 CFR 50.59 controls.

- Seismic instrumentation -- trips the reactor when the ground acceleration exceeds an established value, to minimize the reliance on the safety systems when the plant is in an abnormal operating condition.

- The reactor protection system (RPS) loss-of-load -- trips the reactor when the turbine trips above a specified reactor power level, to minimize the amount of heat that would have to be rejected when the plant is not producing power.

- The RPS steam generator high level -- trips the reactor when the level in the steam generator exceeds a specified level, to avoid excess moisture carryover that would damage the turbine.

These relocated requirements neither are required to be in the TS under 10 CFR 50.36, nor obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in Section 1.0, "Introduction," because these functions are equipment protection features which are not credited in any design basis accident or transients. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff concludes that these requirements may be relocated from the TS to the LCS.

Existing TS 6.7, Safety Limit Violation, has been moved within the improved TS to TS 2.2. This is considered a more logical location from a human factors perspective, and therefore is considered an acceptable administrative change. The requirement (in existing TS 6.7.1) to submit a "Safety Limit Violation Report" has been revised to require a "License Event Report" pursuant to 10 CFR 50.73 when a safety limit is violated, thereby using an existing reporting process.

2.3 Limiting Conditions for Operation Applicability and Surveillance Requirement Applicability (Section 3.0)

This section has been divided and renamed from the existing TS section entitled "Limiting Conditions for Operation and Surveillance Requirements" to the improved SONGS TS section entitled "Limiting Condition for Operation (LCO) Applicability," and "Surveillance Requirement (SR) Applicability." The following discussion covers changes made throughout Section 3.0.

In accordance with the guidance in the Commission's Final Policy Statement the licensee has proposed to relocate existing TS Surveillance Requirement 4.0.5, which specifies requirements for inservice inspection of ASME Code Class 1, 2, and 3 components, from the TS to the LCS documents. The requirements state that inspections shall be performed in accordance with Section XI of the ASME Boiler & Pressure Vessel Code and applicable Addenda. The same requirements are mandated by 10 CFR 50.55a(g), unless specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The staff concludes that control of this commitment under 10 CFR 50.55a is acceptable, and it can be relocated to the LCS.

The licensee has not proposed to relocate any specific LCOs in Section 3.0 from the improved SONGS TS to other licensee-controlled documents. The specifications in Section 3.0 have been revised to be consistent with NUREG-

1432, which results in improved clarity without changing the intent of the specifications in Section 3.0. These changes are administrative in nature. However, the licensee proposed to add the following three new LCOs from NUREG-1432 (3.0.5, 3.0.6, and 3.0.7) to the improved SONGS TS.

(1) LCO 3.0.5 permits equipment removed from service to be returned under administrative control to perform testing to determine operability.

(2) LCO 3.0.6 permits entry into a support system LCO without entry into the supported system's LCO, except as required by the Safety Function Determination Program (SFDP).

(3) LCO 3.0.7 permits certain physics test exceptions.

In addition, the following clarifying statements have been added to SR 3.0.2, SR 3.0.3, SR 3.0.4, and LCO 3.0.4.

- SR 3.0.2 is clarified to provide a completion time extension for each performance of a periodic surveillance requirement.
- SR 3.0.3 quantifies and clarifies the maximum time delay or allowance that is permitted to perform a given surveillance.
- SR 3.0.4 and LCO 3.0.4 clarify limitations on mode applicability changes during shutdown conditions.

The staff has reviewed these proposed additions and concludes that the additions will enhance the clarity and quality of the improved SONGS TS, and will benefit the operators and others in their understanding of the overall improved TS. The staff further concludes that the proposed improved SONGS TS appropriately apply the guidance provided in NUREG-1432. Therefore, the changes are acceptable.

2.3.1 Reactivity Control Systems (Section 3.1)

a. Relocated Requirements

SCE has proposed to relocate some or all of the following existing TS to the COLR and LCS:

Existing TS Number	Title
3.1.1.3	Moderator Temperature Coefficient
3.1.3.6	Regulating CEA Insertion Limits
3.1.3.7	Part Length CEA Insertion Limits
3/4.1.3.2	Position Indicator Channels-Operating
3/4.1.3.3	Position Indicator Channels-Shutdown

Cycle Specific Parameters

The licensee is relocating cycle specific parameters to a Core Operating Limits Report (COLR). The discussion of the relocation of the cycle specific parameters to the COLR is contained in the Administrative Controls discussion in section 2.5. Cycle specific parameters are relocated from existing TS 3.1.1.3 "Moderator Temperature Coefficient," 3.1.3.6 "Regulating CEA Insertion Limits," and 3.1.3.7 "Part Length CEA Insertion Limits."

Position Indicator Channels

The control element assembly (CEA) position indicator channels determine CEA positions and thereby ensure CEA alignment and insertion limits, so that CEA position inputs are provided to the core protection calculators (CPCs) and the core operating limit supervisory system (COLSS). The improved TS retain the position indication requirements for CEA position through the retained CEA alignment and insertion limit LCO requirements and the position indication SRs in the CEA alignment specification. The safety function related to CEA position indication is retained in TS 3.1.5 "CEA Alignment," 3.1.6 "Shutdown CEA Insertion Limits," 3.1.7 "Regulating CEA Insertion Limits," and 3.1.8 "Part Length CEA Insertion Limits." The other requirements associated with position indicator channel in existing LCOs 3/4.1.3.2 and 3/4.1.3.3 can be relocated to the LCS because they are not relied on for adequate reactivity control, they do not provide an indication of a degradation of the reactor coolant pressure boundary or an indication of a challenge to the integrity of a fission product boundary, nor do they prevent or mitigate the consequences of a DBA.

The above relocated requirements relating to reactivity control systems are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these requirements may be relocated from the TS

to the LCS.

b. Less Restrictive Requirements

The shutdown margin (SDM) LCOs (improved TS 3.1.1 and 3.1.2) no longer specifically state applicability in Modes 1 and 2. During power operation, SDM requirements are met when CEA alignment and insertion limit LCOs are met; when those LCOs are not met, SDM requirements are stipulated in the required actions. Existing SR 4.1.1.1.1.d requires the determination of the SDM before operation above 5 percent RTP after each refueling, with the CEA groups at the transient insertion limits of existing TS 3.1.3.6. This SR is met in the improved TS by the application of improved TS 3.1.7, "Regulating CEA Insertion Limits," and SR 3.0.4 and SR 3.1.1. SR 3.0.4 requires that the SDM be established, per improved TS 3.1.1, before entering Mode 3 or 4. CEA transient insertion limits and SDM (as previously discussed) are controlled by improved TS 3.1.7 in Modes 1 and 2. Therefore, the existing TS requirements are maintained by the improved TS. The factors specified for consideration in calculating the SDM in existing SR 4.1.1.1.1.e are relocated to the Bases. The SDM must be calculated correctly; however, listing factors to be considered is not necessary for the TS but is appropriate detail for the Bases.

The moderator temperature coefficient (MTC) SRs 4.1.1.3.1 and 4.1.1.3.2 have become SRs 3.1.4.1 and 3.1.4.2. These new SRs contain the same requirements with the same frequencies for determining beginning-of-life (BOL) and end-of-life (EOL) MTC, but with slightly expanded windows for performance of the SRs. The BOL MTC can now be conducted in ±14 effective full power days (EFPD) (versus ±7 EFPD of peak predicted boron concentration), and EOL MTC can now be conducted in ±30 EFPD (versus ±7 EFPD of 2/3 of expected core burnup). The first measurement taken for extrapolation to EOL, is taken within ±14 EFPD of reaching peak predicted boron concentration at rated thermal power. The ±14 EFPD window allows time to complete startup and establish steady-state operation in a more methodical way that will produce better data. The second measurement for extrapolation to EOL, is taken within ±30 EFPD of 2/3 of expected core burnup. The expanded ±30 EFPD window is acceptable because MTC is directly proportional to boron concentration which changes at a constant rate to EOL. The expanded windows allow increased flexibility for coordination with other plant evolutions without affecting the results of the determinations and with no adverse effects to plant safety.

The frequency of the surveillance to verify CEA freedom of movement (trippability), SR 3.1.5.3, has been extended from monthly to quarterly in accordance with the recommendation in NUREG-1366, "Improvements to TS Surveillance Requirements." This SR 3.1.5.3 frequency has been extended because of the unlikelihood of a stuck CEA, the potential for a dropped CEA during the performance of the SR, and the fact that an untrippable CEA is most often discovered when starting up or when performing CEA drop tests.

The following modifications have been made in the CEA insertion limit LCOs (improved TS 3.1.6, TS 3.1.7, and TS 3.1.8):

- A note has been added to the applicability statement, indicating that the LCO requirements are not applicable while performing the SR 3.1.5.3, the freedom of movement surveillance. This clarifies the requirements to avoid possible future misunderstanding, allowing temporary misalignment in order to demonstrate operability.

- The completion time to restore CEAs to within insertion limits has been extended from 1 to 2 hours, to be consistent with the completion time in improved TS 3.1.5 to restore CEAs to within alignment limits.

- Improved TS 3.1.6, "Applicability," has been expanded to include "Mode 2 with any regulating CEA not fully inserted." The safety analysis assumes that shutdown CEAs are withdrawn anytime the reactor is critical, and that they are withdrawn before the regulating CEAs. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and to maintain the required SDM following a reactor trip. This also limits the potential effects of a CEA ejection accident to within acceptable limits. The existing applicability statement, "With K_{eff} greater than or equal to 1.0," is less conservative than the new applicability statement, since shutdown CEAs can be fully withdrawn, and withdrawal of regulating CEAs can begin before reaching K_{eff} greater than or equal to 1.0.

Existing SR 4.1.1.1.2, requiring a core reactivity balance comparison, has been moved in the improved TS to TS 3.1.3, "Reactivity Balance." There are several differences between the existing SR 4.1.1.1.2 and the improved TS 3.1.3, as follows:

- The existing SR 4.1.1.1.2 is applicable in Modes 1, 2, 3, and 4, while the improved TS 3.1.3 is applicable in Modes 1 and 2. The reason for this change is that in Modes 1 and 2, while the reactor is critical and producing thermal power, the fuel is being depleted and core conditions change. By contrast, in Modes 3 and 4, the reactor is shut down and the reactivity balance is not changing significantly. A reactivity balance is required only in Modes 1 and 2 to ensure that the core is operating as designed.

- In the existing TS, if the requirements of SR 4.1.1.1.2 are not met, the required action is to immediately borate to restore the SDM. The improved TS allows 72 hours to reevaluate the core design and safety analysis to determine whether continued operation is acceptable, during which time appropriate operating restrictions are to be established. The improved TS 3.1.3 provides the flexibility and direction to address the reactivity balance so that problems can be corrected if and when they occur, rather than forcing corrective action that may not be required. If, after 72 hours, the reactivity balance has not been restored, the plant is required to be placed in Mode 3 within 6 hours.

- Existing SR 4.1.1.1.2 requires that the reactivity balance be verified every 31 EFPD, while the improved TS 3.1.3 requires that the reactivity balance be verified every 31 EFPD, only after 60 EFPD. This change

recognizes that sufficient time must be given for core conditions to reach steady state before an accurate comparison can be made.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More Restrictive Requirements

There are no more-restrictive requirements added to section 3.1 of the improved TS.

d. Administrative Changes

The following LCOs have been moved and reorganized within the TS:

Existing TS Number	Title
3/4.1.1.4	Minimum Temperature for Criticality
3/4.1.2.1	Boration Systems Flow Path - Shutdown
3/4.1.2.2	Boration Systems Flow Paths - Operating
3/4.1.2.3	Charging Pumps-Shutdown
3/4.1.2.4	Charging Pumps-Operating
3/4.1.2.5	Boric Acid Makeup Pumps-Shutdown
3/4.1.2.6	Boric Acid Makeup Pumps-Operating
3/4.1.2.7	Borated Water Sources-Shutdown
3/4.1.2.8	Borated Water Sources-Operating
3/4.1.3.4	CEA Drop Time
3/4.1.3.6	Regulating CEA Insertion Limits
3/4.10.1	Special Test Exception (STE)- Shutdown Margin
3/4.10.2	STE - Group Height, Insertion and Power Distribution Limits
3/4.10.3	STE - Reactor Coolant Loops
3/4.10.4	STE - Center CEA Misalignment and Regulating CEA Insertion Limits

Minimum Temperature for Criticality

The "Minimum Temperature for Criticality" LCO (existing TS 3/4.1.1.4) has been moved to the "Reactor Coolant System" section (3.4) of the improved TS.

Charging Pumps, Boric Acid Makeup Pumps, and Borated Water Sources

Normal reactor coolant system (RCS) boron control is needed to help maintain shutdown margin during both power operation and shutdown. At SONGS the boration systems are assumed operable and are required for mitigation of a DBA, and therefore the boration LCOs are retained in the improved TS. This is

different from a plant based on NUREG-1432 where boration systems are not a part of the safety analysis and do not require an LCO. The boron injection system ensures that negative reactivity control is available during each mode of facility operation. Existing TS 3/4.1.2.1, 3/4.1.2.2, 3/4.1.2.3, 3/4.1.2.4, 3/4.1.2.5, 3/4.1.2.6, 3/4.1.2.7, and 3/4.1.2.8 are reorganized into improved TS 3.1.9 and 3.1.10. The specific system operability requirements are incorporated into the retained boration systems specifications and the components required to be operable to perform this function are described in the bases to TS 3.1.9 and 3.1.10. These TS are reorganized in the following manner. The flow paths and pumps required to support emergency core cooling system (ECCS) analysis assumptions for DBA mitigation are controlled by the separate improved TS 3.5.2. The refueling water storage tank (RWST) is required to supply borated injection water for the ECCS, and can also supply the required boric acid for shutdown. The RWST requirements are in improved TS 3.5.4. The systems and components (i.e., pumps, tanks, water volumes and boron concentrations) supporting the ECCS boration requirements are retained in improved TS 3.5.1 and 3.5.4. The boric acid pumps are typically used to transfer concentrated boric acid from an acid makeup tank or the RWST as part of planned normal RCS boron control activities, and do not require dedicated TS.

CEA Drop Time

The safety function of the CEAs is to provide negative reactivity to shutdown the reactor when required. The CEAs are considered inoperable only when they are degraded such that they cannot provide their safety function (that is, when they cannot scram or are untrippable). Previously, operability included movability, but that is only required for the CEAs to meet their alignment and insertion LCOs. As long as CEAs are trippable, within alignment limits, and within insertion limits, they are operable. The CEA drop time requirements in the existing TS (3/4.1.3.4) have been converted to surveillance requirements in the retained CEA alignment specification, where they are most appropriate (SR 3.1.5.5 and SR 3.1.5.6).

Regulating CEA Insertion Limits

In improved TS 3.1.7, "Regulating CEA Insertion Limits," the power-dependent insertion limit (PDIL) alarm circuit has been explicitly included in the LCO statement, rather than merely mentioned in the SRs. This is a human factors editorial change for clarity that does not materially change the specification. SR 3.1.7.1 also now includes a note clarifying that regulating CEAs need not be verified within insertion limits before entry into Mode 2, when they are not expected to be met.

Special Test Exceptions

The licensee has elected to reformat and move all special test exceptions (STEs) into improved SONGS TS 3.1.12, TS 3.1.13 and TS 3.1.14, which were contained in existing TS 3/4.10. NUREG-1432 contains all of the exceptions in these three sections in a format that is easier to understand. The STEs have been rearranged such that only one STE per evolution need be invoked and all

portions of the STE would be applicable. This reorganization simplifies the STEs, reduces the potential for human error, and allows for easier implementation. The reorganization results in three STEs, including (1) Low-Power Physics Testing, (2) At-Power Testing and, (3) Center CEA and Regulating CEA Insertion Limits. Combining four existing STE LCOs (3/4.10.1 "Shutdown Margin," 3/4.10.2 "Group Height, Insertion, and Power Distribution Limits," 3/4.10.3 "Reactor Coolant Loops," and 3/4.10.4 Center CEA Misalignment," into three improved TS, constitutes a reformatting of the TS and is an administrative change.

The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, these improved TS changes are purely administrative and are acceptable.

2.3.2 Power Distribution Limits (Section 3.2)

a. Relocated Requirements

The licensee is relocating cycle specific parameters to a Core Operating Limits Report (COLR). The discussion of the relocation of the cycle specific parameters to the COLR is contained in the Administrative Controls discussion in section 2.5. Cycle specific parameters are relocated from the following existing TS:

Existing TS Number	Title
3/4.2.1	Linear Heat Rate
3/4.2.4	Departure From Nucleate Boiling Ratio
3.2.7	Axial Shape Index

b. Less Restrictive Requirements

Improved SONGS TS Section 3.2 contains two requirements that are less restrictive than those in the existing TS, as follows.

(1) The frequency for performing the Planar Radial Peaking Factor (F^{M}_{xy}) verification surveillance (SR 3.2.2.1) has been changed from "prior to 70% RTP" to "prior to 85% RTP." The level of 85 percent is chosen so that when performing physics tests (which are restricted "to the test power plateau, which shall not exceed 85% RTP"), this verification can be performed at the same level. An advantage to this change is that it provides additional time to obtain incore power distribution data and improved "signal-to-noise ratios," and the change is therefore acceptable because it enhances safety.

(2) The completion time specified in TS 3.2.3 Required Action C.3, "Reduce Linear Power Level-High Trip Setpoints to ≤ 55% RTP," has been extended from 4 hours to 16 hours. After power has been reduced to ≤ 50 percent, the rate and magnitude of changes in the core flux are greatly reduced, and the time increase is therefore acceptable. This change is consistent with NUREG-1432.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More Restrictive Requirements

The licensee added an Action Condition (B) to improved TS 3.2.3, "Azimuthal Tilt (Tq)," to ensure that, for Tq > 0.03 and ≤ 0.10, the CPCs use a conservative value for tilt and provide adequate safety margin for certain limited CEA misalignments. This is in agreement with SONGS plant operating

instruction SO23-3-2.13, which was imposed in response to an ABB-CE suggestion to ensure that the safety analysis remains valid under all circumstances. This is a more restrictive requirement that SCE has volunteered to adopt. The staff has reviewed this more restrictive requirement and believes that it enhances the improved SONGS TS, and is therefore acceptable.

The staff has reviewed the more restrictive requirements and concludes that they result in an enhancement to the improved TS. Therefore, the more restrictive requirements are acceptable.

d. Administrative Changes

Other than the administrative changes required to reformat the existing TS into the improved TS format, there are no additional administrative changes to section 3.2 of the improved TS.

2.3.3 Instrumentation (Section 3.3)

a. Relocated Requirements

In accordance with the guidance in the Commission's final policy statement, the licensee has proposed to relocate and reorganize the following existing TS requirements:

Existing TS Section	Title
3/4.3.2(10)	Toxic Gas Isolation Signal
3/4.3.3.1	Radiation Monitoring Instrumentation
3/4.3.3.2	Incore Detectors
3/4.3.3.3	Seismic Instrumentation
3/4.3.3.4	Meteorological Instrumentation
3/4.3.3.7	Fire Detection Instrumentation
3/4.3.3.9	Explosive Gas Monitoring Instrumentation
3/4.3.3.10	Vibration and Loose-Parts Monitoring System
3/4.3.4	Turbine Overspeed Protection

Trip Function

Loss of Load Trip
Steam Generator Level High Trip
Response Time Tests

Toxic Gas Isolation Signal

The toxic gas isolation signal (TGIS, in existing TS 3/4.3.2) is provided for monitoring and controlling the concentration of potentially toxic gas mixtures in containment and isolating the control room. The TGIS is designed to automatically terminate the supply of outside air to the control room and to initiate operation of the emergency HVAC system to minimize operator exposure, if a toxic hazard is detected. Station operating procedures require isolating the Control Room if a noxious environment is detected; operators are always present to detect this condition and manually actuate the system. The operator may manually initiate the TGIS if plant conditions require or for testing. The toxic gas isolation is not relied on to prevent or mitigate a design basis accident or transient because the Plant design includes other means to safely shutdown the plant if the control room becomes uninhabitable. Therefore, these requirements are relocated to the LCS.

Radiation Monitoring Instrumentation

Existing TS 3/4.3.3.1 on radiation monitors that indicate possible gross failure of fuel cladding or that a release may have originated from the primary containment due to a break in the reactor coolant pressure boundary are relocated and retained in LCOs in the improved TS, as described in the Administrative Changes section below. Radiation monitoring instrumentation remaining in the improved TS are: the containment area radiation monitor, the fuel storage pool airborne monitor, the containment purge isolation monitor,

the containment airborne monitor, and the control room airborne monitor.

Other radiation instrumentation has been relocated from the TS to the LCS, including: the Containment High Range Area Monitor, the Main Steam Line Isolation Area Monitor, the Plant Vent Stack Wide Range Noble Gas Monitor, the Plant Vent Stack Normal Range Noble Gas Monitor, the Condenser Evacuation System Wide Range Noble Gas Monitor, and the Condenser Evacuation System Normal Range Noble Gas Monitor. These radiation monitoring instruments are primarily supporting indications and are not used to detect a degradation of the RCS boundary and do not have any automatic isolation function to prevent or mitigate DBA radioactive releases. Therefore, these provisions for radiation monitors do not meet the Final Policy Statement screening criteria for matters to be set forth in TS (except for those listed in the Administrative Changes section), and will be relocated to the LCS.

Incore Detectors

Existing TS 3/4.3.3.2 provided the operability requirements for the incore detectors. The incore detector system is used to provide detailed information on the reactor core neutron flux distribution. The incore detectors map the spatial neutron flux distribution of the core and provide this information to the plant computers. The information is used by the core operating limit supervisory system (COLSS) and the core protection calculators (CPCs) to verify that the axial power distribution and quadrant tilt are within their limits. The power level information required by COLSS and the CPCs from the incore detectors is retained in the improved TS by the SRs in the power distribution limit TS, as described in the Administrative Changes section. The provisions of the existing incore detector LCO, establishing incore detector operability requirements, will be relocated to the LCS, except as described in the Administrative Changes section. The power range (excore) neutron flux instrumentation is also available to measure axial power distribution (axial imbalance) and quadrant power tilt. The reactor protection system does not use the incore detector instrumentation but instead uses the power range (excore) neutron flux instrumentation to initiate a reactor trip as a result of unacceptable axial core power distribution. No automatic actions result from the incore detector system. The incore detector instrumentation does not provide an indication of a degradation of the reactor coolant pressure boundary or an indication of a challenge to the integrity of a fission product boundary, nor does it mitigate the consequences of a DBA, and therefore it does not meet the Final Policy Statement criteria for retention in the improved TS.

Seismic Instrumentation

The seismic monitoring instrumentation provides monitoring capability by recording information regarding the severity of an earthquake to permit comparison of the measured response to that used in the design basis of the facility to determine if the plant can continue to be operated safely and to permit such timely action as may be appropriate pursuant to 10 CFR Part 100, Appendix A. Since this is determined after the event has occurred, it has no bearing on the prevention or mitigation of any DBA or transient. The safety

analysis requirements do not address the need for seismic monitoring instrumentation that would automatically shut down the plant when an earthquake occurs which exceeds a predetermined intensity. The seismic monitoring instrumentation is not relied upon by operators to take immediate action in the event of an earthquake. Therefore, the existing TS (3/4.3.3.3) requirements for seismic monitoring instrumentation are relocated to the LCS.

Meteorological Instrumentation

Meteorological instrumentation is used to measure environmental parameters (wind direction, speed, and air temperature differences) that may affect distribution of fission products and gases following a DBA. Meteorological instrumentation is to be used in connection with the plans for coping with radiological emergencies, pursuant to 10 CFR 50.34(b), and it is used to provide a basis for estimating maximum potential annual radiation doses resulting from radioactive materials released in gaseous effluents, pursuant to 10 CFR 50.36a(a)(2). The meteorological instrumentation is not relied upon to actuate safety systems based on predetermined environmental effects, and is not required to prevent or mitigate a DBA in the facility. These provisions (in existing TS 3/4.3.3.4) will be relocated to the LCS.

Fire Detection Instrumentation

Fire detection instrumentation ensures that adequate warning capability is available in order to detect and locate fires in their early stages. The minimum fire detection and mitigation requirements are established in the regulations, with which the licensee must comply regardless of whether the requirements are restated in the TS. As recommended in Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications," fire protection requirements can be relocated from TS in the areas of fire detection systems, fire suppression systems, fire barriers and fire brigade staffing requirements. The regulations require that the licensee have a fire protection plan (10 CFR 50.48), including design features that satisfy Criterion 3 of Appendix A to 10 CFR Part 50, and a quality assurance (QA) program which ensures appropriate maintenance of such plans (Appendix B to 10 CFR Part 50). In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 and 10 CFR 50.54(a). Since the fire detection instrumentation (in existing TS 3/4.3.3.7) is not relied on to prevent or mitigate a design basis accident or transient, as described in more detail in section 2.3.7, these requirements will be relocated to the LCS.

Explosive Gas Monitoring Instrumentation

Explosive gas monitoring instrumentation is provided for monitoring and controlling the concentration of potentially explosive gas mixtures in the waste gas holdup system. The explosive gas monitoring instrumentation ensures that the concentration of hydrogen in the waste gas holdup system is maintained below the flammability limit. This instrumentation is not relied upon to prevent or mitigate a DBA or to prevent exceeding applicable dose limits at the Exclusion Area Boundary upon the rupture or uncontrolled release of any single tank's contents in the waste gas holdup system (consistent with

Standard Review Plan 15.7.1, "Waste Gas System Failure"). Accordingly, these requirements (in existing TS 3/4.3.3.9) will be relocated to the LCS.

Loose-Parts Detection Instrumentation

Loose-parts detection instrumentation (existing TS 3/4.3.3.10) monitors core noises to identify any loose parts located inside the reactor vessel. The loose-parts detection system provides information only and is not considered in any design-basis accident or transient. In addition, the potential for fuel failure due to fuel bundle flow blockage from a lost part will be detected by the radiation monitors retained in the improved TS. Therefore, these requirements will be relocated to the LCS.

Turbine Overspeed Protection Instrumentation

The turbine is equipped with control valves and stop valves which control turbine speed during normal plant operation and protect it from overspeed during abnormal conditions. The turbine overspeed protection system consists of separate mechanical and electrical sensing mechanisms which are capable of initiating fast closure of the steam valves. The existing TS 3/4.3.4 requires particular operability and surveillance requirements for these steam control and stop valves to minimize the potential for fragment missiles that might be generated as the result of a turbine overspeed event.

Although the design basis accidents and transients include a variety of system failures and conditions which might result from turbine missiles striking various plant systems and equipment, the system failures and plant conditions could be caused by other events as well as turbine failures. In view of the low likelihood of turbine missiles, this scenario does not constitute a part of the primary success path to prevent or mitigate such design basis accidents and transients. Similarly, the turbine overspeed control is not part of an initial condition of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Probabilistic safety assessments (PRA) and operating experience have demonstrated that proper maintenance of the turbine overspeed control valves is important to minimize the potential for overspeed events and turbine damage; however that experience has also demonstrated that there is low likelihood of significant risk to public health and safety because of turbine overspeed events. Further, the potential for and consequences of turbine overspeed events are diminished by the favorable orientation of the turbine, relative to the likely path of any turbine missiles, and the licensee's inservice inspection program, which must comply with 10 CFR 50.55(a), and a surveillance program for the turbine control and stop valves derived from the manufacturer's recommendations.

Accordingly, the staff has concluded that the requirements for turbine overspeed controls do not meet the TS criteria in the Final Policy Statement. The limiting conditions for operation and surveillance requirements for turbine overspeed controls were removed from the standard technical specifications. These requirements will be relocated to the LCS.

Loss of Load Trip

The purpose of the loss-of-load trip (in existing 3/4.3.1) in the RPS is to trip the reactor when the turbine has tripped above a certain power level. The loss-of-load trip is intended to prevent material damage and to prevent the lifting of the pressurizer code safety valves and the main steam safety valves (MSSVs) in the event of a turbine trip. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Without a Loss of Load reactor trip, the reactor would trip on high pressurizer pressure -- which is a credited signal and has been maintained in the improved TS. The loss-of-load trip will be relocated to the LCS.

Steam Generator Level High Trip

The steam generator level-high trip (in existing TS 3/4.3.1) is provided to protect the turbine from excessive moisture carry over. As such, this feature primarily protects against turbine damage and is not relied upon to prevent or mitigate any design basis accidents or transients. The steam generator level-high trip operating requirements will be relocated to the LCS.

Response Time Tests

RPS and ESFAS Instrumentation response time tests (RTT) (in existing TS Tables 3.3-2 and 3.3-5, and in new SRs 3.3.1.13, 3.3.2.5, and 3.3.5.5) specify time limits for the entire channel, from the time the monitored parameter exceeds its setpoint until the equipment is capable of performing its intended function. The SRs retained in the improved TS include testing of the channel portion of the instrument on a staggered basis. The SRs no longer include the specific response time values because that detail can be adequately controlled under 10 CFR 50.59. The relocation of response time limits is consistent with GL 93-08, "Relocation of Tables of Instrument Response Time Limits." RPS and ESFAS Instrumentation have RTT SRs in the improved TS to determine operability. Specific RTT requirement details are relocated to the LCS and referenced by the SRs.

The above relocated requirements relating to installed plant instrumentation are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 for any future changes to response time limits. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS.

b. Less Restrictive Requirements

In improved TS 3.3.1, the requirement to verify RCS flow by comparison with the RCP differential pressure instrument or with calorimetric calculations at "above 70%" RTP has been changed to "above 85%" RTP to be consistent with

performance of physics tests (which are restricted "to the test power plateau, which shall not exceed 85% RTP"). Also, the performance of an RCS heat balance to verify that the indicated linear power level, CPC delta T power, and CPC nuclear power agree with calorimetric calculations at "above 15%" RTP, has been changed to above 20% RTP to reflect that accurate data is difficult to obtain at the lower power levels. These changes provide for more efficient operation and the recording of more accurate data which has the overall effect of increasing safety.

The shutdown portions of the "Log Power Level--High" function in Modes 3, 4, and 5, were relocated to LCO 3.3.2, "Reactor Protective System Instrumentation--Shutdown," (with the reactor trip circuit breakers (RTCBs) closed), and LCO 3.3.13, "Logarithmic Power Monitoring Channels" (with the RTCBs open). To detect a loss of SDM (caused by boron dilution), LCO 3.3.13 was developed to provide neutron flux monitoring when the RTCBs are open. The "Log Power Level--High" function is bypassed above 1E-4% RTP, before entering Mode 1, and Mode 1 applicability was therefore deleted. In addition, the frequent quarterly channel functional test of the "Log Power Level--High" function is sufficient to ensure operability; therefore, the channel functional test before startup is unnecessary and has been deleted.

The monthly channel functional test for the control room isolation signal (CRIS), containment purge isolation signal (CPIS), and fuel handling isolation signal (FHIS) gaseous channels has been extended to quarterly, based upon recommendations in NUREG-1366 and associated GL 93-05, which are appropriate and applicable to SONGS.

Remote shutdown (RSD) instrumentation now includes only those instruments required to meet 10 CFR 50, Appendix A, GDC 19. The system is not required to respond to any design basis accident evaluated in the safety analysis, but is provided to comply with GDC-19. The completion time to restore a required RSD channel to operable status has been revised from 7 days to 30 days. This is consistent with NUREG-1432, operating experience, and the low probability of an event requiring control room evacuation.

Improved TS 3.3.11, "Post Accident Monitoring Instrumentation (PAMI)," has been revised to include all Regulatory Guide 1.97, Type A instruments, and all Regulatory Guide 1.97, Category 1, non-Type A instruments. Type A variables provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, and are required for safety systems to accomplish their safety functions for design basis accidents. Category 1 variables are the key variables deemed risk-significant because they are needed to achieve the following functions: (1) determine whether other systems important to safety are performing their intended functions; (2) provide information to the operators that will enable them to determine the potential for causing a gross breach of barriers to radioactivity release; and, (3) provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat. Consistent with the screening criteria and NUREG-1432, the revision to improved TS 3.3.11

includes the following specific changes:

- Containment spray pressure and low pressure system injection (LPSI) header temperature are Type D, Category 2 variables, do not meet the above (category 1) criteria for inclusion in the PAMI specification, and are therefore deleted from this specification.

- The completion time for the required action to restore one inoperable channel (when one is inoperable) has been extended to 30 days (from 7 days), based on the reliability of the remaining channels, the passive nature of the instruments (no critical automatic action is assumed to occur from these instruments), and the low probability of an event occurring requiring PAMI during the 30-day interval.

- The completion time for the required action to restore one inoperable channel (when two or more are inoperable) has been extended to 7 days (from 48 hours) based on the passive nature of the instruments (no critical automatic action is assumed to occur from these instruments), and the low probability of an event occurring requiring PAMI during the 7-day interval.

- The containment hydrogen monitors have been moved to this specification from existing specification 3/4.6.4. The additional changes made to this specification are as follows:

    - The time to restore a monitor to operable status if two monitors are inoperable has been extended to 72 hours (from 48 hours). The 72 hour completion time is based on the relatively low probability of an event requiring hydrogen monitoring and the availability of alternative means to obtain the required information.

    - A special report to the NRC is required, in lieu of a plant shutdown, when a required function cannot be restored within 7 days. The report would discuss alternate equivalent instrumentation/functions, justify areas that do not have equivalent alternate instrumentation/functions, and provide a schedule for restoration of the instrumentation.

    - A channel check, which is the appropriate check (since automatic functions are not initiated by the hydrogen monitors), is now required every 31 days in lieu of a channel functional test.

    - The calibration surveillance frequency has been extended to 24 months (from 92 days) in accordance with NUREG-1366, which is appropriate and applicable.

- The containment area radiation monitor is moved to this specification from the radiation monitoring instrumentation specification (which is otherwise relocated to the LCS as described the Administrative Changes part of this section).

The particulate/iodine radiation monitors in the control room ventilation trains have been deleted from the Control Room Isolation System (CRIS) system TS. The gaseous monitors are more reliable, would provide an earlier indication of an accident condition (i.e., alarm earlier), and only the gaseous monitors are taken credit for in the safety analysis. This modification results in changes to the CRIS system Bases description and a resulting change in operability definition that has no adverse effect on safety.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. <u>More Restrictive Requirements</u>

Consistent with the screening criteria and NUREG-1432, the revision to improved TS 3.3.11 includes the following specific changes:

- The excore neutron flux monitor has been added, to meet Regulatory Guide 1.97, Type A, Category I variable requirements. This will ensure the capability to monitor reactor power/subcriticality levels.

- The condensate storage tank level has been added, to meet Regulatory Guide 1.97, Type A, Category 1 variable requirements. This will ensure the capability to monitor secondary cooling water levels.

- The containment isolation valve position has been added, to meet Regulatory Guide 1.97, Type A, Category 1 variable requirements. This will ensure the capability to monitor containment integrity.

The staff has reviewed the more restrictive requirements and concludes that they result in an enhancement to the improved TS. Therefore, the more restrictive requirements are acceptable.

d. Administrative Changes

The following LCOs or portions of LCOs have been moved and reorganized within the TS:

Existing TS Number	Title
3/4.3.1	Reactor Protective Instrumentation
3/4.3.2	Engineered Safety Feature Actuation System Instrumentation
3/4.3.3.1	Radiation Monitoring Instrumentation (portion relocated, see Relocated Requirements)
3/4.3.3.2	Incore Detectors

Reactor Protective Instrumentation

The reactor protective instrumentation TS (3/4.3.1) has been split into the following five distinct specifications:

- TS 3.3.1 - RPS Instrumentation--Operating
- TS 3.3.2 - RPS Instrumentation--Shutdown
- TS 3.3.3 - Control Element Assembly Calculators
- TS 3.3.4 - RPS Logic and Trip Initiation
- TS 3.3.13 - Source Range Monitoring Channels

Each of the above five improved TS addresses the applicable portion of existing TS 3/4.3.1 from which it was derived. The following table identifies the reorganization of the instrumentation requirements in TS 3/4.3.1.

Functional Unit in TS 3/4.3.1	Improved TS
-Manual Reactor Trip	3.3.4 (all Modes)
-Linear Power Level--High	3.3.1
-Log Power Level--High	3.3.1 (Modes 1 and 2) 3.3.2 (Modes 3, 4 and 5 with RTCBs closed) 3.3.13 (Modes 3, 4 and 5 with RTCBs open)
-Pressurizer Pressure--High	3.3.1
-Pressurizer Pressure--Low	3.3.1
-Containment Pressure--High	3.3.1
-Steam Generator Pressure--Low	3.3.1
-Steam Generator Level--Low	3.3.1
-Local Power Density--High	3.3.1
-DNBR--Low	3.3.1
-RPS Logic	3.3.4
-RTBs	3.3.4
-CPCs	3.3.3
-CEACs	3.3.3
-Reactor Coolant Flow--Low	3.3.1

Engineered Safety Feature Actuation System Instrumentation

The engineered safety feature actuation system (ESFAS) instrumentation TS has been split into the following six distinct specifications:

- TS 3.3.5 - ESFAS Instrumentation
- TS 3.3.6 - ESFAS Logic and Manual Trip
- TS 3.3.7 - Diesel Generator (DG) - Loss of Voltage Start (LOVS)
- TS 3.3.8 - Containment and Purge Isolation Signal (CPIS)
- TS 3.3.9 - Control Room Isolation Signal (CRIS)
- TS 3.3.10 - Fuel Handling Isolation Signal (FHIS)

Each of the above six improved TS addresses the applicable portion of existing TS 3/4.3.2, from which it was derived. The following table identifies the reorganization of the instrumentation requirements within the existing TS 3/4.3.2.

Functional Unit in TS 3/4.3.2	Improved TS
1. Safety Injection	
a. Manual	3.3.6 Function 1.d
b. Containment Pressure--High	3.3.5 Function 1.a
c. Pressurizer Pressure--Low	3.3.5 Function 1.b
d. Logic	3.3.6 Function 1.c
2. Containment Spray	
a. Manual	3.3.6 Function 5.d
b. Containment Pressure--High	3.3.5 Function 2.a
c. Logic	3.3.6 Function 5.c
3. Containment Isolation	
a. Manual CIAS (Containment Isolation Actuation Signal)	3.3.6 Function 2.d
b. Containment Pressure-- High	3.3.5 Function 2.a
c. Logic	3.3.6 Function 2.c
4. Main Steam Line Isolation	
a. Manual	3.3.6 Function 6.d
b. SG Pressure--Low	3.3.5 Function 4.a
c. Logic	3.3.6 Function 4.c
5. Recirculation	
a. RWST--Low	3.3.5 Function 5.a
b. Logic	3.3.6 Function 4.c
6. Containment Cooling	
a. Manual CIAS	3.3.6 Function 3.c
b. Manual SIAS	3.3.6 Function 3.c
c. Logic	3.3.6 Function 3.d

Functional Unit in TS 3/4.3.2	Improved TS
7. LOV	
a. LOV	3.3.7
8. ESFAS	
a. Manual	3.3.6 Function 7.d, 8.d
b. Logic	3.3.6 Function 7.c, 8.c
c. SG Level A/B--Low and delta P A/B--High	3.3.5 Function 6.a, 6.b, 7.a, 7.b
d. SG Level A/B--Low and no SG Pressure Low Trip	3.3.5 Function 6.c, 7.c
9. CRIS	
a. Manual CIAS	3.3.9
b. Manual SIAS	3.3.9
c. Airborne Radiation	
i. Gaseous	3.3.9
d. Logic	3.3.9
10. FHIS	
a. Manual	3.3.10
b. Airborne Radiation	
i. Gaseous	3.3.10
c. Logic	3.3.10
11. CPIS	
a. Manual	3.3.8
b. Airborne Radiation	
i. Gaseous	3.3.8
ii. Particulate	3.3.8
iii. Iodine	3.3.8
c. Containment Area	3.3.8
d. Logic	3.3.8

Radiation Monitoring Instrumentation

Radiation monitors that indicate possible gross failure of fuel cladding or that a release may have originated from the primary containment due to a break in the reactor coolant pressure boundary are moved from existing TS 3/4.3.3.1 and included in the following retained LCOs:

- The post-accident-monitoring instrumentation (PAMI) included in improved TS 3.3.11, "Post Accident Monitoring Instrumentation," includes a subset of the radiation monitoring instruments (specifically, the containment area radiation monitor). However, since PAMI is applicable to Modes 1 through 3, and the containment area radiation monitor is applicable to Modes 1 through 4, only the monitor's Mode 1 through 3 applicability will be retained in the improved TS, while its Mode 4 applicability will be relocated to the LCS (as discussed in the relocated section above, on radiation monitors). Appropriate notes have been added to the LCS to

direct the user to improved TS 3.3.11.

- The fuel storage pool airborne monitor is incorporated into improved TS 3.3.10, "Fuel Handling Isolation Signal (FHIS)."
- The containment purge isolation monitor and containment airborne (gaseous) monitor are incorporated into improved TS 3.3.8, "Containment Purge Isolation Signal."
- The control room airborne (gaseous) monitor is incorporated into improved TS 3.3.9, "Control Room Isolation Signal (CRIS)."

When high radiation levels are detected in the exhaust streams monitored by these parameters, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products.

Incore Detectors

The incore detector system is used to provide detailed information on the reactor core neutron flux distribution. The incore detectors map the spatial neutron flux distribution of the core and provide this information to the plant computers. The information provided by the incore detectors to COLSS and the CPCs is used to verify that the axial power distribution and quadrant tilt are within their limits. While the existing incore detector LCO (3/4.3.3.2) is relocated, as described in the Relocated Requirements section, the information related to core power levels from the incore detectors is retained in the improved TS by the SRs in the power distribution limit LCOs.

The above changes result in the same limits as the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the improved TS changes are purely administrative and are acceptable.

2.3.4 Reactor Coolant System (Section 3.4)

a. Relocated Requirements

In accordance with the guidance in the Commission's final policy statement, the licensee has proposed to relocate the following existing TS to other licensee-controlled documents:

Existing TS Section	Title
3/4.4.6	Chemistry
3.4.4.9	Structural Integrity
3/4.4.10	Reactor Coolant Gas Vent System

Chemistry

The reactor coolant water chemistry program provides limits on particular chemical properties of the primary coolant, and surveillance practices to monitor those properties, to ensure that degradation of the reactor coolant pressure boundary is not exacerbated by poor chemistry conditions. However, degradation of the reactor coolant pressure boundary is a long-term process, and there are other more direct means to monitor and correct the degradation of the reactor pressure boundary which are controlled by regulations and TS; for example, in-service inspection conducted in accordance with 10 CFR 50.55a, and primary coolant leakage limits. On this basis, the staff has concluded that the reactor coolant chemistry program is not required to be in the TS to protect the public health and safety, and may be relocated to the LCS.

Structural Integrity

Existing TS 3.4.9 addresses the structural integrity of ASME Code Class 1, 2, and 3 components, which are monitored so that the possibility of component structural failure does not degrade the safety function of the system. The monitoring activity is of a preventive nature, rather than a mitigative action. In addition, except for the reactor coolant pump (RCP) flywheel inspection, surveillances are already required by 10 CFR 50.55a to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda. The RCP flywheel inspection requirement is not covered by other regulatory requirements and is needed for the safe operation of the plant; therefore, this requirement has been included in the improved SONGS TS Section 5.5.2.5. The improved SONGS TS 5.5.2.10, in conjunction with Section XI of the ASME Boiler and Pressure Vessel Code, provides a programmatic approach to the requirements relating to the structural integrity of ASME Code Class 1, 2, and 3 components. Consequently, the requirements of TS 3.4.9 have been relocated to the UFSAR.

Reactor Coolant Gas Vent System

Existing TS 3/4.4.10, "Reactor Coolant Gas Vent System," ensures the availability of an exhaust pathway from the RCS to remove noncondensable gases that could inhibit natural circulation core cooling to improve the plant's capability to cope with severe accidents beyond the DBA. The system is normally isolated and requires manual operator action to initiate flow. The flow rate is limited to control the hydrogen vented to containment atmosphere so that combustible limits are not exceeded. The Reactor Coolant Gas Vent System does not function to prevent or mitigate a DBA or transient. Consequently, the requirements of TS 3/4.4.10 will be relocated to the LCS.

The above relocated requirements relating to the reactor coolant system are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the UFSAR or to the LCS as applicable.

b. Less Restrictive Requirements

In improved TS 3.4.1, "RCS Pressure, Temperature, and Flow Limits," the completion times for Condition B, to be in Mode 2, and Condition D, to be at ≤30% RTP, have been changed to 6 hours (from 4 hours) to allow for an orderly reduction in power and to be consistent with NUREG-1432. For Conditions A and C, the 2-hour completion times for the restoration of parameters provides sufficient time to adjust plant parameters, determine the cause of the off-normal condition, and restore the readings within limits, on the basis of operating experience.

Existing TS 3.1.1.4, "RCS Minimum Temperature for Criticality," has become TS 3.4.2 in the improved SONGS TS. In this TS, references to T-average have been replaced with the cold leg temperature, T_c, because the safety analysis is done with T_c and it is more conservative. Also, the SR requirement for RCS temperature verification within 15 minutes before criticality has been eliminated, without any adverse effect on safety, for the following reasons:

- It is a requirement of the LCO, by virtue of its applicability, that the MTC requirement be satisfied, with or without the SR.

- Before commencing rod withdrawal for a startup, the RCS temperature must be within the normal operating band.

- Using the SR to check the MTC within 15 minutes before criticality is a distraction to the operator because it places undo importance on the MTC in relation to other plant conditions during startup.

- The SR requirement to check the MTC every 30 minutes after achieving

criticality has been retained.

In improved TS 3.4.3.1, "Pressurizer Heatup/Cooldown," 72 hours is now allowed to evaluate pressurizer structural integrity, rather than the 6 hours given in the existing TS, if the pressurizer is restored to within normal parameters within 30 minutes. This recognizes that once the pressurizer is within normal operating limits, the danger of pressurizer vessel failure has passed and it is very unlikely that structural damage would have occurred. In addition, the 72 hours allows sufficient time to adequately evaluate the structural integrity of the pressurizer vessel, surge line, spray line and spray nozzle, and other associated components, before requiring any MODE changes.

The completion time for the reduction in power to Mode 3 in existing TS 3.4.4, "RCS Loops --Modes 1 and 2," has been extended from 1 hour to 6 hours, to allow for an orderly reduction in power that enhances safety and to be consistent with NUREG-1432.

SR 3.4.8.2 is an addition to the SONGS TS to verify that the required number of trains are operable, thereby ensuring that redundant paths for heat removal are available, if needed, to maintain decay heat removal. Verifying breaker alignment and the availability of indicated power to required pumps every 7 days is sufficient to maintain safety and satisfy the requirements of this SR, in light of other available administrative controls and operating experience.

In improved TS 3.4.14, required action A.2 requires isolation of the high-pressure portion of the affected system with use of a second valve within 72 hours (rather than the 4 hours of existing TS 3.4.5.2). In the existing TS, isolation of both valves is required within 4 hours, while in the improved TS only the first isolation valve is required within 4 hours. This change is acceptable because, based on operating experience, it allows time for repair of the leaking PIV, and because there is a very low probability of a second valve leaking during the 72-hour completion time. The frequency of SR 3.4.14.1 for testing for PIV leakage, before entering Mode 2, has been extended from 72 hours before entry to 7 days, to allow for maintenance planning flexibility. This change also reflects the related recommendation of NUREG-1366, which is appropriate and applicable to SONGS.

Steam generator tube failures during power operation are immediately detectable by steam line or air ejector radiation monitors. Similarly, minor tube leaks can be determined by various analyses or evaluations (steam generator chemistry, inventory balances, etc.), as required by improved SONGS TS LCO 3.4.13, "RCS Operational Leakage," and LCO 3.4.15, "RCS Leakage Detection Instrumentation." Also, improved LCO 3.4.15 now permits two diverse means of monitoring RCS leakage rather than the three means of indication required by the existing TS. This effectively means that atmospheric grab samples need be taken only if there are no radiation monitoring devices operable, and that an inventory balance must be performed within 24 hours, rather than 12 hours. The performance of an inventory balance in 24 hours is a sufficient frequency to discover an RCS leak. These relaxations are acceptable considering the reliability and accuracy of the remaining radiation monitors, and the likelihood that at least one of the two diverse indications

would discover an RCS leak.

The frequencies for performing SR 3.4.15.3 and SR 3.4.15.4, the channel tests for the leakage detection instrumentation, have been changed from 31 to 92 days, per NUREG-1366, which is appropriate and applicable to SONGS.

In improved TS 3.4.16, "RCS Specific Activity," the frequency of SR 3.4.16.1, to measure RCS gross specific activity, has been extended from 72 hours to 7 days because of the time required to perform the analysis and the more immediate availability of data from Dose Equivalent I-131 analysis which is performed (SR 3.4.16.2) within 2 to 6 hours of power changes of 15 percent RTP or more.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More restrictive Requirements

There are no more-restrictive requirements added to section 3.4 of the improved TS.

d. Administrative Changes

Steam Generators

Existing TS 3/4.4.4 for steam generators provides the basis for detailed SRs for steam generator (SG) tube inspection. These inspections, which are prescribed to provide reasonable assurance of SG tube integrity during plant operating conditions, can only be performed during plant shutdown conditions. SG tube inspection requirements in the improved SONGS TS provide the same level of control as those contained in the existing TS. Where the existing TS contain an LCO and numerous SRs, the improved SONGS TS contain a single requirement (SR 3.4.13.2), to verify that steam generator tube integrity satisfies the requirements of Specification 5.5.2.11, "Steam Generator Tube Surveillance Program." Improved TS 5.5.2.11 (except for minor editorial differences) is identical to existing TS 3/4.4.4. Improved TS 5.5.2.11, in conjunction with improved SONGS TS SR 3.4.13.2, contains the information necessary to verify that the SGs are operable, and that SG tube integrity is restored before increasing average RCS temperature above 200° F.

Existing TS 3.4.5.2, "Operational Leakage," has been divided into two TS (improved TS 3.4.13, "RCS Operational Leakage," and improved TS 3.4.14, "RCS PIV Leakage"). This is an editorial change and merely separates the requirements.

The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the improved TS changes are purely administrative and are acceptable.

2.3.5 Emergency Core Cooling Systems (Section 3.5)

a. Relocated Requirements

Existing TS Requirement

TS 3/4.5.1, Safety Injection Tank SRs 4.5.1.c and 4.5.1.e
TS 3/4.5.2 & 3, ECCS Subsystems Reporting Requirement

Safety Injection Tank (SIT) Surveillances

In existing TS 3/4.5.1, "Safety Injection Tanks," SR 4.5.1.c verifies the status of SIT vent valve fuses, and SR 4.5.1.e verifies SIT isolation valves open automatically. These SRs are not necessary to verify that the SITs can meet their design basis safety function, and are relocated to the LCS. Improved TS 3.5.1 retains the requirements that ensure SIT operability: that the SIT isolation valve is open, that the SIT nitrogen cover pressure is acceptable, that the SIT water volume is at an acceptable level, and that the SIT water has an acceptable boron concentration.

ECCS Subsystems Reporting Requirement

In existing TS 3/4.5.2, "ECCS - Subsystems - $T_{avg} \geq 350°$," and TS 3/4.5.3, "ECCS - Subsystems - $T_{avg} < 350°$," the special report of ECCS actuation including excessive nozzle usage factors is relocated from the TS to the LCS because it is redundant to the reporting requirements in 10 CFR 50.73(a)(2)(iv).

The above relocated requirements relating to the emergency core cooling systems are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS.

b. Less Restrictive Requirements

In improved TS 3.5.1, "Safety Injection Tanks (SITs)," a condition has been added to the improved SONGS TS that is not in the existing TS, allowing one SIT to be inoperable due to low boron concentration for up to 72 hours. In this condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if one of the four SITs were not available for injection. The staff concludes that the improved SONGS TS have made

appropriate application of the guidance provided in NUREG-1432 and are acceptable.

The surveillance in existing TS SR 4.5.2.d, to visually inspect containment every 31 days to verify there is no loose debris that could be transported to the containment sump, has been deleted in the improved TS. The requirements of SR 4.5.2.d are normal post-maintenance requirements. The periodic inspection for loose debris inside containment and abnormal degradation of the sump is provided by SR 3.5.2.9 every refueling outage. This deletion is consistent with, and an appropriate application of, GL 93-05 and NUREG-1366 on TS SR improvements.

SR 4.5.2.e.1, to verify, at every refueling outage, that the shutdown cooling system (SDCS) automatic interlocks prevent opening of SDCS valves at prescribed conditions is not necessary to ensure SDCS operability. The improved SONGS TS provide sufficient SRs to demonstrate system operability, and are consistent with NUREG-1432.

In improved TS 3.5.3, "ECCS-Shutdown," the completion time for required action B, to reach MODE 5, has been extended from 20 to 24 hours to allow for a more orderly transition to cold shutdown, and to be consistent with comparable completion times for MODE transitions, for applicability in MODES 3 and 4.

In improved TS 3.5.4, "Refueling Water Storage Tank," the completion time to restore RWST boron concentration and water temperature to within limits has been extended from 1 to 8 hours. The 8 hour completion time is based on the continued availability during the transition time of the contents of the RWST for injection into the RCS and the time required to change boron concentration or temperature. The previous 1 hour completion time assumed the unavailability for RWST contents, which was overly conservative. This 8-hour completion time is consistent with NUREG-1432.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More Restrictive Requirements

SR 3.5.2.8 is added to verify the low pressure safety injection (LPSI) pump stops on an actual or simulated actuation signal to ensure operability. Previously only the isolation valves were required to be verified shut upon receipt of a recirculation actuation signal (RAS), and this new SR now verifies that the pumps must stop to prevent pump damage and ensure sustained operability. This SR has the same 24-month frequency as the other pump operability SRs (e.g., SR 3.5.2.7).

The staff has reviewed the above more restrictive requirement and concludes that it results in an enhancement to the improved TS. Therefore, the more restrictive requirement is acceptable.

d. Administrative Changes

Other than the administrative changes required to reformat the existing TS into the improved TS format, there are no additional administrative changes to section 3.5 of the improved TS.

2.3.6 Containment Systems (Section 3.6)

a. Relocated Requirements

The licensee has proposed to relocate the containment isolation valve list in existing TS 3/4.6, "Containment Systems."

Existing Table 3.6-1, "Containment Isolation Valves," contains a list of all of the valves that are relied upon to perform isolation functions to ensure that containment leakage does not exceed limits assumed in DBAs and transients. The improved TS 3.6.3, "Containment Isolation Valves," provides a limiting condition for operation which specifies sufficient requirements for the overall containment function, including required actions and surveillance requirements and is applicable to all containment isolation valves. Accordingly, there is no need to list each of the containment isolation valves by name in a table in the TS. In accordance with GL 91-08, the detailed listing of containment isolation valves will be relocated to the LCS.

The above relocated requirements relating to the containment systems are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS.

b. Less Restrictive Requirements

A note has been added to SR 3.6.2.2 (existing SR 4.6.1.3.c) stating that the door airlock interlock need only be tested upon entry into containment at a 6-month interval since the interlock is only challenged upon entry. This results in a reduction in wear to the airlock doors and interlock without a reduction in safety.

Existing SR 4.6.1.3.b.2, to verify air lock leakage after maintenance, has been deleted since post-maintenance actions are not needed to be prescribed in TS because, after maintenance, equipment and systems must be restored to operable status in accordance with the general rules that apply to the LCO and related SR leakage requirements (improved TS 3.6.2 and SR 3.6.2.1).

SR 3.6.5.1 (existing SR 4.6.1.5) has been changed to require that containment temperature measurements be taken at locations representative of containment temperature, rather than list specific locations. It is the type of detail that is not necessary to be stated in a surveillance and, if necessary, can appear in the associated procedure. This change also provides flexibility so that the licensee can respond to changing conditions and more adequately meet the SR requirements under differing circumstances.

The frequency for existing SR 4.6.1.7.3 (SR 3.6.3.6 in the improved TS), to

test the leak rate for containment purge with resilient seals, has been changed from 3 months to 6 months, and 92 days after opening the valve. The 6 months is consistent with the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration." A 92-day frequency after opening recognizes that seal degradation can occur due to excessive valve cycling.

In improved TS 3.6.6.1, "Containment Spray and Cooling Systems," the completion time in Condition B to reach Mode 4, after reaching Mode 3, has been increased from 48 to 84 hours based upon NUREG-1432. The extended interval to reach Mode 4 allows a reasonable period of time to restore a containment spray train, and is acceptable considering that the threat of release of radioactive material from the reactor vessel is reduced upon reaching Mode 3. In Condition E, with two spray trains inoperable or with any combination of three or more trains of spray and cooling inoperable, the required action is to enter Specification 3.0.3; this is a more restrictive requirement than is provided in the existing TS. Also, in SR 3.6.6.1.8, the frequency of verifying that the spray valve is unobstructed has been extended from 5 to 10 years, as recommended by NUREG-1366, since the only problems of this nature have been construction-related and are not expected to be detected now that construction has been completed.

In improved TS 3.6.7, "Hydrogen Recombiners," Condition B has been added, allowing two trains to be inoperable for up to 7 days with alternate means of limiting hydrogen buildup available. This is acceptable because the use of the mixing action of the dome air circulators, combined with the containment spray system and the emergency fan coolers, would prevent any localized accumulations above the flammability level, and the hydrogen recombiners would not be needed for about 14 days following a LOCA. In addition, the backup hydrogen purge system could be used to limit hydrogen buildup to below its flammability level. The frequency for SR 3.6.7.1 (existing SR 4.6.4.2), to conduct a functional test of the recombiners, has been extended from 6 months to 24 months, as recommended by NUREG-1366 (which found the hydrogen recombiners to be highly reliable). This extension is acceptable. Existing SR 4.6.4.2.b.1, to conduct a channel calibration every refueling outage, has been deleted because the objective is satisfied by a functional test (SR 3.6.7.1), to ensure that the hydrogen recombiners can attain and sustain temperatures necessary for recombination, and that is now conducted at every refueling outage.

In improved TS 3.6.8, "Containment Dome Air Circulators," Condition B has been added, allowing two dome air circulators to be inoperable for up to 7 days with an alternate means of limiting hydrogen buildup available (provided by the hydrogen recombiners, containment spray system, and the containment emergency air coolers, as discussed in the previous paragraph). In the event of an accident requiring hydrogen recombiner operation, operator action is not required for up to 14 days following the event. This new completion time is acceptable.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant

safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More restrictive Requirements

There are no more-restrictive requirements added to section 3.6 of the improved TS.

d. Administrative Changes

The requirements of the following existing TS have been reorganized and moved.

Existing TS Sections	Title
3/4.6.1.2	Containment Leakage
3/4.6.1.6	Containment Structural Integrity
3/4.6.4.1	Hydrogen Monitors

Containment Leakage

Existing TS 3/4.6.1.2 has been retained as an SR in improved TS 3.6.1. These requirements pertaining to containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the plant safety analysis at the peak accident pressure. Containment leakage rate limits assumed in the accident analysis are controlled by 10 CFR Part 50 Appendix J. The existing TS 3/4.6.1.2 specifications contain details which are also found in Appendix J to 10 CFR Part 50. The regulations require licensee compliance and cannot be revised by the licensee. The improved TS provide for requirements that are "... in accordance with Appendix J and approved exemptions." Therefore, direct reference to Appendix J eliminates the need for repetitious and unnecessary details within the improved TS. This change is consistent with the guidance in NUREG-1432, and is purely administrative in nature.

Containment Structural Integrity

Existing TS 3/4.6.1.6 provides acceptance criteria for containment structural integrity. The requirement to demonstrate the capability of the containment structure to meet its design function on a periodic basis, has been retained as an SR in improved TS 3.6.1. Improved SONGS TS SR 3.6.1.2 establishes the testing program. The details of the "Containment Tendon Surveillance Program" that list the specific tendons and their associated lift-off force, will be located in the LCS. It is noted that the definition of "containment integrity" is no longer utilized, and that its requirements (such as leakage rates and structural integrity) are included under "Containment Operability." Thus, the improved SONGS TS incorporate the same requirements in a different form.

Hydrogen Monitors

The function of the monitors is to measure the hydrogen concentration inside containment and alert the control room operator of the need to activate the hydrogen recombiners or the hydrogen purge system. The hydrogen monitors thereby assist in ensuring the hydrogen concentration is maintained below its lower combustible limit, which helps assure containment integrity is maintained. Existing TS 3/4.6.4 requirements for hydrogen monitors have been incorporated into improved TS 3.3.11, "Post Accident Monitoring (PAM) Instrumentation." This is an administrative change in the location of the safety objective within the TS, and is therefore acceptable.

The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the improved TS changes are purely administrative and are acceptable.

2.3.7 Plant Systems (Section 3.7)

a. Relocated Requirements

The licensee has elected to relocate the following existing TS to the licensee-controlled LCS:

Existing TS Sections	Title
3/4.7.2	Steam Generator Pressure/Temperature Limitations
3/4.7.6	Snubbers
3/4.7.7	Sealed Source Contamination
3/4.7.8	Fire Suppression System
3/4.7.8.1	Fire Suppression Water System
3/4.7.8.3	Fire Hose Stations
3/4.7.9	Fire Rated Assemblies

Steam Generator Pressure/Temperature Limitations

Pressure and temperature (P/T) limits are placed on the steam generators to prevent a nonductile failure of the steam generator (SG) boundary. 10 CFR Part 50, Appendix G provides P/T limits for the reactor coolant pressure boundary (RCPB), and TS requirements for SG tube surveillances ensure the integrity of the boundary between the reactor coolant system and the SG boundary. In addition, 10 CFR 50.55a provides requirements for inservice inspection, including the SG. The requirements of the existing TS (3/4.7.2) for the SG P/T limits are based on the structural analysis of the SG and are calculated using the ASME Code for Class 1 components; and the primary boundary limits are adequately addressed in improved TS 3.4.3, "RCS P/T Limits." Therefore, the requirements specified in the existing TS have been relocated to the LCS and will be controlled in accordance with 10 CFR 50.59 and 10 CFR 50.55a.

Snubbers

The existing "Snubbers" TS (3/4.7.6) states that all snubbers shall be operable. Snubbers are passive devices used for supporting piping systems, and the associated TS action statement only requires that an inoperable snubber be replaced or repaired within 72 hours. The surveillance requirement for snubbers is that they be periodically examined under the inservice inspection program. The existing requirements that all snubbers be operable are requirements that do not impact reactor operation, do not identify a parameter that is an initial condition assumption for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not form part of the primary success path which functions or actuates to mitigate a design basis accident or transient. Snubber surveillance requirements, therefore, do not meet the guidance for inclusion in the TS. Further, the relocation of the lists of snubbers is consistent with GL 84-13. The provisions will be relocated to the LCS and controlled by 10 CFR 50.55a and 10 CFR 50.59.

Sealed Source Contamination

The existing requirement for sealed source contamination (TS 3/4.7.7) specifies limitations on fixed contamination for sources requiring leak testing, and states that sealed sources containing radioactive material shall be free of specified levels of removable contamination. The associated action statement requires that if the removable contamination exceeds limitations, the sealed source shall be either decontaminated or disposed of. The limitations expressed in this TS do not impact reactor operation, do not identify a parameter which is an initial condition assumption for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary and do not provide any mitigation of a design basis event. The provisions will be relocated to the LCS and appropriate health physics procedures will be adequately controlled under 10 CFR 50.59 and Part 20.

Fire Protection Requirements

The design features required for fire protection, the fire protection program, and the Fire Protection Plan provide reasonable assurance that the occurrence of any fires will not present an undue risk to public health and safety. Other details of the fire prevention and mitigation program are not required to avert an immediate threat to the public health and safety, nor do they fall within any of the four criteria for technical specifications. Examples of these details are fire detection instrumentation design, fire suppression system capabilities, fire barrier construction, and the frequency of QA audits of the fire protection plan. Although there are aspects of the fire detection and mitigation functions that have been determined to be risk significant, the minimum requirements for those functions are established in the regulations, with which the licensee must comply regardless of whether the requirements are restated in the TS. As recommended in Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications," fire protection requirements can be relocated from TS in the areas of fire detection systems, fire suppression systems, fire barriers and fire brigade staffing requirements. The relocation of fire protection requirements from TS is consistent with the guidance in NUREG-1432 and in GL 86-10, "Implementation of Fire Protection Requirements." In the generic letters, the staff concluded that the provisions of 10 CFR 50.59 should apply directly to changes a licensee desired to make in the fire protection program so long as those changes did not interfere with the ability to achieve and maintain safe shutdown. The standard license condition, included within GL 86-10, stated that changes which interfere with the ability to achieve and maintain safe shutdown in the event of a fire required prior approval of the staff. Thus, the license condition established as part of the GL 86-10 implementation also makes this administrative control unnecessary. This license condition is License Condition 2.C.(14) of Operating License No. NPF 10 and License Condition 2.C.(14) of Operating License No. NPF 15, for SONGS Units 2 and 3, respectively. The regulations require that the licensee have a fire protection plan (10 CFR 50.48), including design features that satisfy Criterion 3 of Appendix A to 10 CFR Part 50, and a quality assurance (QA) program which ensures appropriate maintenance of such plans (Appendix B to

10 CFR Part 50). In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 and 10 CFR 50.54(a). Accordingly, the staff has concluded that existing TS 3/4.7.8.1, 3/4.7.8.3 and 3/4.7.9, may be relocated from the TS to the LCS and licensee's QA Plan, as appropriate. The control of these provisions under the terms of the license condition and 10 CFR 50.59 is acceptable.

The above relocated requirements are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR §50.59, §50.54(a), §50.55a, Part 50 Appendix R, and 10 CFR Part 20. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS.

b. <u>Less Restrictive Requirements</u>

The licensee has adopted a significant portion of the NUREG-1432 provisions for the plant systems. Additional specifications and clarifications beyond the existing TS have been made throughout this section in order to make the section more user-friendly, consistent with current staff positions and the current licensing basis for SONGS. In particular, the following changes are incorporated in the improved SONGS TS:

- In improved TS 3.7.1, the revised Main Steam Safety Valve (MSSV) set point tolerances previously approved in SONGS 2/3 License Amendments 114/103 of November 23, 1994 (i.e., MSSVs will be set within +/-1% and may have an as-found lift setting of +1/-3%) have been incorporated. In addition, SR 3.7.1.1 is modified by a note that allows entry into and operation in Mode 3, and entry into Mode 2 prior to performance of the SR, where the conditions are appropriate for its conduct.

- In improved TS 3.7.2, the completion time to restore an inoperable main steam isolation valve (MSIV) is increased from 4 to 8 hours because the MSIVs are different from containment isolation valves that allow communication with containment atmosphere, in that they isolate a closed system that penetrates containment and provide a potential additional means for containment isolation. Also, if that completion time is not met, then 6 hours is provided to be in Mode 2, rather than 2 hours, to allow time for possible MSIV repair and to avoid the potential for a transient caused by an unnecessary shutdown. The increase in completion times will not significantly increase the probability or consequences of any accident previously analyzed.

- In improved TS 3.7.4, the completion time to restore one atmospheric dump valve (ADV) line to operable status, when both are inoperable, has been increased from 6 to 24 hours to account for: the block valve that can be closed to isolate an ADV making repairs possible, and allowing time for the repairs; and the availability of the steam bypass system and MSSVs.

- In improved TS 3.7.5, the completion times for the required actions to restore one auxiliary feedwater (AFW) train to operable status have been increased from 6 hours to 48 hours when both motor-driven pumps are inoperable, and to 24 hours when one motor-driven and the turbine driven pump are inoperable. The NRC staff has found this acceptable due to: the existence of two 100 percent motor-driven AFW pumps and one 100 percent turbine-driven AFW pump, which can feed either SG; and the redundant design capabilities of the AFW system. This change is reflected in NUREG-1432.

- In improved TS 3.7.6, the completion time for restoring condensate storage tank level is increased from 4 hours to 7 days, based upon verification of backup source operability within 4 hours, and every 12 hours thereafter. This change is acceptable because the existing TS do not account for a backup source of condensate water and it is consistent with NUREG-1432.

- The frequency of SR 3.7.11.4, to test the ability of the control room emergency air cleanup system (CREACUS) train to maintain pressure in the control room has been changed to the refueling cycle "on a staggered test basis," and is consistent with the guidance provided in the Standard Review Plan (NUREG-0800), Section 6.4.

- The duration of the CREACUS train test in SR 3.7.11.1 has been reduced from 10 hours to 15 minutes, consistent with NUREG-1432. This reduction is acceptable because: (1) the emergency filters are not credited for removing radioactivity in the dose analyses, (2) the emergency ventilation heaters are not required to maintain relative humidity below 70 percent at the filters in the emergency recirculation unit which are credited in the dose analysis, and (3) an operating time of 15 minutes is sufficient time to initiate flow through the system, establish and maintain the proper system parameters, and ensure operability.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. <u>More Restrictive Requirements</u>

Improved TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)," is an addition to the SONGS existing TS. The MFIVs are able to isolate main feedwater flow to the secondary side of the steam generators following a high-energy line break. Closure of the MFIVs terminates flow to both steam generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks inside containment, and reducing cooldown effects for steam line breaks. The MFIVs meet Criterion 3 of the NRC Policy Statement, in that they

are part of the primary success path that functions to mitigate the design basis accident. The NUREG-1432 MFIV TS has been utilized with slight modifications to reflect SONGS-specific design differences. For example, the SONGS units do not have MFIV bypass valves. In addition, the completion time for required action A, to "close or isolate the inoperable MFIV," has been changed from 72 hours to 7 days based upon plant design differences and supporting PRA analysis.

Improved TS 3.7.5 has a new condition not currently in the existing TS. Improved TS 3.7.5 now has required actions when one AFW train is inoperable in Mode 4, to ensure at least two trains of decay heat removal are operable. This is a more restrictive and conservative requirement than currently exists.

The staff has reviewed these more restrictive requirements and concludes that they result in an enhancement to the improved TS. Therefore, these more restrictive requirements are acceptable.

d. Administrative Changes

The requirements of the following existing TS have been reorganized and moved.

Existing TS Requirements

SR 3.7.11.2 and 3.7.14.2, Ventilation Filter Testing Surveillances
TS 3/4.7.8.2, Spray and/or Sprinkler Systems

Ventilation Filter Testing Program

The details of SR 3.7.11.2 and SR 3.7.14.2 on ventilation filter testing have been relocated to the Ventilation Filter Testing Program (VFTP), in the Administrative Controls Section 5.5.2.12 of the improved TS.

Spray and/or Sprinkler Systems

Existing TS 3/4.7.8.2 is retained within the improved TS and moved to the "Containment Systems" section, where it has become TS 3.6.6.1, Containment Spray and Cooling Systems, and 3.6.6.2, Containment Cooling System. The requirements of the existing TS have been retained and the changes that have been made are discussed above in Section 2.3.6, "Containment Systems."

Auxiliary Feed Water SRs

The change from 24 to 72 hours in the improved TS SR notes (relative to NUREG-1432) for SRs 3.7.5.2, 3.7.5.3, and 3.7.5.4 are based upon the existing TS having an SR 3.0.4 (4.0.4) exception without a time constraint, which effectively permits 72 hours to accomplish the SRs. The notes in the improved TS provide a similar, more specific, effect to the SR 3.0.4 exception; the 72 hours is the plant specific requirement.

The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the improved TS changes are purely administrative and are acceptable.

2.3.8 Electrical (Section 3.8)

a. Relocated Requirements

The licensee proposed to relocate the following TS to other licensee-controlled documents:

Existing TS Number	Title
3/4.8.4.1	Containment Penetration Conductor Overcurrent Protective Devices
3/4.8.4.2	Motor Operated Valves Thermal Overload Protective Devices

Containment Penetration Conductor Overcurrent Protective Devices

Existing TS 3/4.8.4.1 describes primary containment penetration conductor overcurrent protective devices. The electrical equipment protective devices are provided in the plant design to minimize the potential for equipment faults to cause overcurrent conditions which would result in a failure of associated containment penetrations for the electrical cables. The likelihood of such an overcurrent condition occurring coincident with or as a result of a design basis accident or transient, that would challenge the containment integrity, is quite low. These devices are not considered part of the primary success path to prevent or mitigate such design basis accidents and transients, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Therefore, these requirements specified in the existing TS do not satisfy the criteria for technical specifications, and may be relocated to the LCS, and incorporated by reference in the UFSAR, such that future changes to these provisions may be made pursuant to 10 CFR 50.59.

Motor-Operated Valves Thermal Overload Protective Devices

Existing TS 3/4.8.4.2 describes motor-operated valve thermal overload protective devices. These devices protect the equipment from potential damage to maintain the capability of the equipment, but are not relied to mitigate a design basis accident or transient. Further, such design details as thermal overload protection devices are inherently considered in the operability of valves which are relied on for the safety functions specified in LCOs (for example, ECCS and containment isolation valves). These design details are specified in the associated Bases and the UFSAR, while the general requirements for thermal overload protection have been relocated to the LCS.

DG Surveillances

The following SRs have been relocated since they are more appropriate for administrative procedures and are not essential for determining operability or whether the LCO has been met:

- SR 4.8.1.1.2.a.6, for insuring DG standby alignment to the emergency buses
- SR 4.8.1.1.2.d.1, for inspecting the DGs in accordance with manufacturers' recommendations, once per refueling outage
- SR 4.8.1.1.2.d.9, for verifying that the auto-connected loads to each DG do not exceed 4700 KW
- SR 4.8.1.1.2.d.14, for verifying the operability of the K23 lockout relay which prevents the DG from starting
- SR 4.8.1.1.3, for reporting to the NRC the valid and non-valid DG failures. Deletion of this requirement from the existing TS is consistent with changes outlined in Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," issued on August 30, 1994.

The above relocated requirements relating to the electrical systems are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these LCO requirements may be relocated from the TS to the LCS, and the SR requirements may be relocated to plant procedures.

b. Less Restrictive Requirements

In the improved TS, required actions in TS 3.8.1, A.C. Sources--Operating are revised to:

- provide an option to avoid unnecessary testing of the remaining operable DG if it can be determined within 24 hours that the cause of the inoperable DG does not exist on the operable DG. The allowance to confirm operability of the other DG is in conformance with RG 1.93 (and GL 84-15), which indicate that 24 hours is reasonable to confirm that the operable DG is not affected by a similar problem as the inoperable DG.
- require that loads supported by an inoperable DG are declared inoperable when their redundant required loads are inoperable. The four-hour time to declare inoperability allows a reasonable time to restore capability before subjecting the plant to a the possibility of a shutdown transient and reflects the low probability of a DBA occurring during this period.

- not require starting both operable DGs when two offsite sources are inoperable, not require demonstrating the operability of two offsite power sources when two DGs are inoperable, and not require verifying supported feature(s) are operable when one DG is inoperable. At the same time, the improved TS now require declaring redundant features inoperable in 12 hours after discovery of two inoperable offsite sources. These changes are made to avoid the possibility of unnecessary transient conditions resulting from the cycling of OPERABLE equipment, during a limited time period, permitting efforts to be devoted toward attempts at corrective action.

- reflect the battery surveillance requirements in NUREG-1432, as revised to be consistent with proposed changes in the industry standard for battery testing (IEEE-450) and to accommodate 24-month refueling cycles. The scope and frequency for the battery service test and performance discharge test are less restrictive in the improved SONGS TS than the existing requirements. However, the improved SONGS TS include additional requirements for the performance discharge test, as well as a provision for an alternate modified performance discharge test. SR 3.8.4.7 verifies at least once per refueling interval that the battery capacity is adequate by means of a battery service test. SR 3.8.4.7 is modified by a note that allows SR 3.8.4.8, a battery discharge test, to be performed in lieu of SR 3.8.4.7 once every 48 months since SR 3.8.4.8 may represent a more severe test. The requirements of SR 4.8.2.1.b.2 and 3 to verify, within 7 days after a battery discharge with the terminal voltage below 110 volts and/or battery overcharge with terminal voltage above 150 volts, that there is no visible corrosion at the battery terminals and connectors, or the connector resistance is less than the specified limits, and that the average electrolyte specific gravity and temperatures of representative is within limits, has been deleted (the 92 day requirement of these same SRs is retained). This is acceptable because the set of battery testing requirements reflected in the improved SONGS TS will minimize testing that can be detrimental to the life of the batteries, but will provide a more effective measure of the capability of the DC electrical power system. On this basis, the staff concludes that the proposed changes are acceptable.

In improved TS 3.8.5, "DC Sources-Shutdown," required action A.1 has been added as an option to allow declaring inoperable the required features associated with the inoperable DC electrical power subsystem. This is appropriate because the remaining trains with DC power available may be capable of supporting systems that allow continuation of core alterations and fuel movement. The applicability of improved TS 3.8.5 has been expanded to include "during movement of irradiated fuel assemblies," and is appropriate to ensure that features required to mitigate a fuel handling accident, including instrumentation to monitor the refueling condition, are operable.

In improved TS 3.8.6, "Battery Cell Parameters," the Condition A required actions have been modified from restoring battery operability in 2 hours to verifying pilot cell electrolyte level and float voltage meet Category C (in Table 3.8.6-1) values in one hour, all battery cells meet Category C values in

24 hours, and restoring battery parameters to Category A and B limits within 31 days. The verifications provide assurance that during the time needed to restore battery parameters to Category A and B limits, the battery will be capable of performing its intended function, and the plant will not be unnecessarily cycled because of rigid adherence to an unrealistically stringent requirement.

In improved TS 3.8.8, "Inverters-Shutdown," required action A.1 has been added as an option to allow declaring inoperable the required features associated with the inoperable inverter. This is appropriate because the remaining operable inverters may be capable of supporting sufficient required features to allow continuation of core alterations, fuel movement, and operations with the potential for positive reactivity additions.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. Administrative Changes

The licensee proposed the following reorganization of the TS:

Existing TS	Improved TS	Title
3/4.8.1.1	3.8.1	A.C. Sources--Operating
3/4.8.1.2	3.8.2	A.C. Sources--Shutdown
3/4.8.1.1 & 2 (parts of each)	3.8.3	Diesel Fuel Oil, Lube Oil, and Starting Air
3/4.8.2.1	3.8.4	D.C. Sources--Operating
3/4.8.2.2	3.8.5	D.C. Sources--Shutdown
3/4.8.2.1 & 2 (parts of each)	3.8.6	Battery Cell Parameters
3/4.8.3.1	3.8.7	Inverters--Operating
3/4.8.3.2	3.8.8	Inverters--Shutdown
3/4.8.3.1	3.8.9	Onsite Power Distribution Systems--Operating
3/4.8.3.2	3.8.10	Onsite Power Distribution Systems--Shutdown

SR 4.8.1.1.2.d.8 has become SR 3.8.1.14 and SR 3.8.1.15. SR 3.8.1.14 verifies that each DG operates with the maximum KVAR loading for 24 hours and has been revised to be consistent with Regulatory Guide 1.9, Revision 3. SR 3.8.1.15 verifies that each DG starts and achieves, within 10 seconds or less, the specified voltage and frequency within acceptable limits.

SRs 4.8.1.2.1 and 4.8.1.2.2 are combined as SR 3.8.2.1, which is modified by a note to indicate that the listed SRs are not required to be performed on the operable DGs. This precludes paralleling the DGs with the offsite power

network, or otherwise unnecessarily rendering the DG or required offsite circuit inoperable.

Improved TS 3.8.3, "Diesel Fuel Oil, Lube Oil and Starting Air" specification requirements are part of TS 3/4.8.1.1 and TS 3/4.8.1.2 in existing specifications. NUREG-1432 places these requirements in a separate TS to ensure DG fuel oil, lube oil and air capacity and quality are sufficient to maintain the DGs operable under design conditions, and to provide for an appropriate time to return these requirements to within limits when the LCO is not met prior to declaring the DG inoperable. Existing SR 4.8.1.1.2.d.12, to transfer fuel oil from each fuel storage tank via the cross-connect lines to the day tank every refueling outage, has been deleted since it is unnecessary (it is not required in the design safety analysis), and is an overly prescriptive detail that is not required to be in TS.

The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the improved TS changes are purely administrative and are acceptable.

2.3.9 Refueling Operations (Section 3.9)

a. Relocated Requirements

The licensee proposed to relocate some or all of the following TS to other licensee-controlled documents:

Existing TS Number	Title
3/4.9.1	Boron Concentration
3/4.9.3	Decay Time
3/4.9.5	Communication
3/4.9.6	Refueling Machine
3/4.9.7	Fuel Handling Machine - Spent Fuel Pool Building

Cycle Specific Parameters

The licensee is relocating cycle specific parameters to a Core Operating Limits Report (COLR). The discussion of the relocation of the cycle specific parameters to the COLR is contained in the Administrative Controls discussion in section 2.5. Cycle specific parameters are relocated from existing TS 3/4.9.1 "Boron Concentration.

Decay Time

Existing TS 3/4.9.3 provides the minimum time requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel. This requirement ensures that sufficient time has elapsed to allow radioactive decay of the short lived fission products. The operations required prior to moving irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals) require well in excess of 24 hours (and approaches the 72 hours in the existing TS) to complete before irradiated fuel can be moved. Therefore, the requirement is not relied on for fuel handling accidents and has been relocated from the specifications to the LCS.

Communication

Existing TS 3/4.9.5 provides the requirement for communications capability to ensure that personnel can be promptly informed of significant changes in facility status and reactivity conditions during CORE ALTERATIONS. The refueling system design accident or transient response does not take credit for communications and is only designed to ensure safe refueling operations. Therefore, the requirements have been relocated to the LCS, incorporated by reference in the UFSAR, and will be controlled in accordance with 10 CFR 50.59.

Refueling Machine and Fuel Handling Machine - Spent Fuel Pool

Existing TS 3/4.9.6 and 3/4.9.7 provide loading restrictions for the refueling machines corresponding to assumptions in the analysis for fuel handling accidents and dropping heavy loads in the spent fuel pool. These loading restrictions are consistent with the design capabilities of the associated cranes. However, the bounding accident analyses demonstrate acceptable radiological consequences even if the fuel bundles or heavy loads are dropped, based on the secondary containment capability and filtration capacity of the safety-related ventilation systems. Consequently, the loading restrictions for the refueling machines do not constitute a primary success path to prevent or mitigate fuel handling accidents and therefore, do not satisfy the TS criteria. The loading restrictions, including procedures for defeating crane interlocks and stops for specific analyzed loading conditions, will be relocated to the LCS.

The above relocated requirements relating to the refueling operations are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS.

b. Less Restrictive Requirements

Existing SR 4.9.2, which serves to demonstrate operability of the source range monitors, has been revised and replaced by two SRs: 3.9.2.1, to conduct a channel check every 12 hours, and 3.9.2.2, to perform a channel calibration every 24 months. The requirements to conduct a channel functional test 8 hours prior to the start of core alterations and every 7 days have been deleted. The functional test is not applicable since the instruments do not actuate equipment, and only provide indication in the control room. In addition, the performance of the channel check every 12 hours is sufficient to ensure operability and the check 8 hours prior to core alterations is an administrative burden that is unnecessary and provides no commensurate increase in safety. These changes are, therefore, acceptable.

Existing SR 4.9.3, determining the status/operability of containment penetrations, has been replaced by two SRs (3.9.3.1 and 3.9.3.2) with the existing requirements retained, with the exception of the requirement to verify containment penetrations closed/isolated 72 hours prior to core alterations. The existing requirement to verify the status of containment penetrations every 7 days is deemed sufficient, and there is not a safety gain commensurate with the increased administrative burden of requiring this check within 72 hours prior to core alterations. The requirement to determine the status of containment penetrations prior to core alterations is part of plant procedures, so there is no reduction in safety by this deletion. The existing containment purge isolation system TS 3/4.9.9 has been relocated to improved

TS 3.3.8 and SR 3.9.3.2 with no resulting decrease in requirements.

SR 3.9.6.1 verifies the refueling water level every 24 hours during core alterations, but does not include the requirement in existing SR 4.9.10 to verify refueling water within two hours prior to commencement of core alterations. This is acceptable because meeting the SR is required prior to entry into the applicability of the TS, without the requirement being explicitly stated (per SR 3.0.4). Also, this prescriptive detail is unnecessary in improved TS 3.9.6 because if the refueling water level is not within limits, the required action stipulates that fuel movement must cease within containment; this is more stringent than the existing TS which only require that fuel movement cease within the reactor vessel.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More restrictive Requirements

There are no more-restrictive requirements added to section 3.9 of the improved TS.

d. Administrative Changes

Other than the administrative changes required to reformat the existing TS into the improved TS format, there are no additional administrative changes to section 3.9 of the improved TS.

2.4 Design Features (Section 4.0)

a. Relocated Requirements

In accordance with the guidance in NUREG-1432, the licensee has proposed to relocate all or portions of the following existing TS to other licensee-controlled documents:

Existing TS	Title
5.1.3	Site Boundary for Gaseous Effluents
5.1.4	Site Boundary for Liquid Effluents
5.2.1	Containment Configuration
5.2.2	Design Pressure and Temperature
5.4.1	Reactor Coolant System Design Pressure and Temperature
5.4.2	Reactor Coolant System Volume
5.5.1	Meteorological Tower Location
5.7.1	Component Cyclic or Transient Limit

Existing Figures	Title
5.6-3	Fuel Storage Patterns for Region II Racks
5.6-4	Fuel Storage Patterns for Region II Racks Reconstitution Station

Site Boundary for Gaseous Effluents and Site Boundary for Liquid Effluents

The improved TS will retain the figures showing the Exclusion Area Boundary and Low Population Zone, but delete the subsections and figures on "Site Boundary for Gaseous Effluents" (existing TS 5.1.3) and "Site Boundary for Liquid Effluents" (existing TS 5.1.4). The deleted portions do not meet 10 CFR 50.36(c)(4) criteria for retention, and related limits are covered by 10 CFR Part 20 and 10 CFR Part 100 requirements. These site characteristics are not relied on to avert an immediate threat to the public health and safety. They are reflected as elements of the original site suitability evaluation (under 10 CFR Part 100), and in procedures for demonstrating compliance with various environmental protection regulations, the Emergency Plan, and the Security Plan. Accordingly, these site characteristics are depicted in the UFSAR for which 10 CFR 50.59 provides adequate controls. The staff has, therefore, concluded that this proposed change is acceptable.

Design Information

These improved TS sections contain the same material as found in the existing TS, except for the following subsections that are deleted: "Containment" (existing TS 5.2 including 5.2.1 and 5.2.2), and "Reactor Coolant System" (existing TS 5.4 including 5.4.1 and 5.4.2). Specific requirements for the performance of the associated systems, structures and components are specified in the improved TS limiting conditions for operation, in accordance with the criteria in the Final Policy Statement. The additional detailed information

currently in the Design Features section for containment and RCS is not relied on to satisfy the limiting conditions for operation, or relied on to avert an immediate threat to public health and safety. The information contained in the Design Features section of the TS is described in the UFSAR, so that any changes to these design features would have to be evaluated under §50.59 before they could be made.

Meteorological Tower Location

The licensee proposes to eliminate the existing description which specifies the location of the meteorological tower by a reference to Figure 5.1-1. The meteorological data provided by the tower is important to the emergency response capabilities which are adequately controlled by the Emergency Plan requirements under Appendix E to 10 CFR Part 50. The licensee indicated that the inclusion of Figure 5.1-1 in the UFSAR, described above, will ensure that any change to the tower's location would be adequately evaluated under 10 CFR 50.59. For these reasons, meteorological tower location was removed from the Design Features section in NUREG-1432. The staff, therefore, concludes that these changes are acceptable.

Component Cyclic or Transient Limits

The licensee proposes to relocate the information related to cyclic limits, including TS 5.7.1 and Table 5.7-1 of the existing Design Features to the LCS. These details are contained in the UFSAR and any changes are subject to 10 CFR 50.59. While the cyclic and transient limits are important to the design analysis specified by Section III of the ASME Code, those limits relate to the maintenance of long-term fatigue usage factors; these limits do not constitute conditions of immediate importance with regard to design basis accidents or transients which present a challenge to fission product barriers. For this reason, these limits were removed from NUREG-1432. The table will be relocated to the LCS and a new program, Specification 5.5.2.4, "Component Cyclic and Transient Limit," will be implemented to provide controls to track cyclic and transient occurrences to ensure that components are maintained within the design limits. The staff has, therefore, concluded that this proposed change is acceptable.

Fuel Storage Patterns for Region II Racks

The Design Features section of the improved TS retains the specifications on minimum fuel assembly center-to-center storage distance, K_{eff} limits, and the maximum enrichment for fuel in the fuel storage racks. These specifications meet the 10 CFR 50.36(c)(4) requirements that design features are to include "geometric arrangements, which, if altered or modified, would have a significant effect on safety...". Storage configurations that meet these specifications satisfy 10 CFR 50.36(c)(4) requirements. Existing figures 5.6-3 and 5.6-4 on "Fuel Storage Patterns for Region II Racks," which provide redundant requirements to those retained in the improved TS, are not required in the Design Features section of the TS, and are not included in NUREG-1432. Therefore, the staff concludes that the figures on the fuel storage patterns can be relocated to the LCS, and that 10 CFR 50.59 provides adequate controls

for maintaining specific fuel storage patterns that are consistent with the requirements retained in the improved TS.

The above relocated requirements relating to design features are not required to be in the TS under 10 CFR 50.36(c)(4) or Section 182a of the Atomic Energy Act and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS.

b. Less Restrictive Requirements

There are no less-restrictive requirements added to section 4.0 of the improved TS.

c. More Restrictive Requirements

There are no more-restrictive requirements added to section 4.0 of the improved TS.

d. Administrative Changes

Controls for component cyclic and transient limits are moved to improved TS Section 5.5. The spent fuel pool storage figures on fuel burnup versus initial enrichment have been moved to improved TS 3.7.18, "Spent Fuel Assembly Storage." Other changes to the design features section are administrative and editorial. The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the staff concludes that these changes are administrative in nature and are acceptable.

2.5 Administrative Controls (Section 5.0)

The licensee has proposed the NUREG-1432 specifications for Section 5.0 with some plant-specific differences to reflect the current licensing basis.

a. Relocated Requirements

In accordance with the guidance in NUREG-1432 the licensee has proposed to relocate all or portions of the following existing TS to other licensee-controlled documents:

Existing TS Section	Title
6.2.2.d	SRO Present During Core Alterations
6.2.2.e	Fire Brigade Requirements
6.4	Training
6.5	Review and Audits
6.6	Reportable Occurrence Action
6.8.1.d	Security Plan Implementation
6.8.1.e	Emergency Plan Implementation
6.8.2	Review and Approval Process
6.8.3	Temporary Change Process
6.8.4.b	In Plant Radiation Monitoring
6.8.4.f	Radiological Environmental Monitoring Program
6.9	Adoption of Core Operating Limits Report Relocating Cycle Specific Limits
6.9.2	ECCS Actuation Special Report (relocating reporting requirements in TS 3/4.5.2 and 3/4.5.3 on ECCS Subsystems)
6.10	Record Retention
6.10.2.h	Inservice Inspection Program Requirements
6.11	Radiation Protection Program
6.13	Process Control Program

Figures and Tables	Title
Figure 6.2-1	Offsite Organization
Figure 6.2-2	Unit Organization
Table 6.2-1	Minimum Shift Crew Composition
Figure 6.2-3	Control Room Area

SRO Present During Core Alterations

The licensee proposes that the requirement (existing TS 6.2.2.d) that an SRO be present during fuel handling and supervise all core alterations not be retained in the TS. This is required by 10 CFR 50.54(m)(2)(iv) and need not be controlled by the TS to assure safe operation of SONGS. The current regulation states,

> "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding

a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person."

This requirement is specified in the Administrative Instructions for 10 CFR 50.54. The staff concludes that the regulatory requirements provide sufficient control of these provisions and removing this requirement from the TS is acceptable.

Fire Brigade Requirements

The Fire Protection Program (NUREG-1432, Section 5.7.2.18) provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire. Existing TS 6.2.2.e specifies the composition of the fire brigade. Fire Protection Program requirements are being relocated to the LCS and QA Plan, as discussed in section 2.3.7 above. The fire brigade administrative control was originally developed to ensure the capability to provide for alternate/dedicated safe shutdown in accordance with 10 CFR Part 50, Appendix R. As such, it allows for the ability to place the unit in a safe condition in the event of a fire. The details of this specification are defined and implemented by the Fire Protection Program and are not necessary to be stated in the TS. The relocation of this administrative control from TS is also consistent with the guidance in NUREG-1432 and in GL 86-10, "Implementation of Fire Protection Requirements." The NUREG does not include fire brigade requirements since the requirements of Appendix R pertain regardless of whether these details are controlled by the TS. The staff concludes that fire brigade requirements can be relocated from the TS to the LCS, and be controlled through the Fire Protection Program and 10 CFR 50.59.

Training

The licensee proposes that the requirements (existing TS 6.4) on training be deleted from the improved SONGS TS on the basis that they are adequately addressed by other Section 5.0 administrative controls and the UFSAR, as well as in the regulations. The staff finds that improved SONGS TS 5.3, "Unit Staff Qualifications," provides adequate requirements to assure an acceptable, competent operating staff. Each member of the SONGS staff is required to meet or exceed the minimum qualifications of specific Regulatory Guides or ANSI Standards acceptable to the NRC staff. Improved SONGS TS 5.3 describes the details of the required qualifications. The UFSAR describes the details of the SONGS training program.

Additionally, improved SONGS TS 5.2, "Organization," details SONGS staff requirements. Improved TS 5.2.2 and 10 CFR 50.54(m) describe the minimum shift crew composition and delineate which positions require a reactor operator (RO) or a senior reactor operator (SRO) license. Training and requalification of those positions are as specified in 10 CFR Part 55.

The licensee has proposed to relocate the remaining provisions to the UFSAR

and appropriate plant procedures, as previously described. The staff concludes that these provisions do not need to be controlled by the TS under the Commission's regulations. Therefore, control of these provisions under 10 CFR 50.59 is acceptable. The staff also concludes that the regulatory requirements in 10 CFR 50.54(m) and 10 CFR Part 55 provide sufficient control of these provisions, and removing them from the TS is acceptable.

Review and Audits

The licensee proposes that the review and audit functions (existing TS 6.5) be deleted or relocated to licensee controlled documents on the basis that they are adequately controlled elsewhere. These TS provisions are not necessary to assure safe operation of SONGS, given the requirements in the Quality Assurance (QA) Program implementing 10 CFR 50.54 and 10 CFR Part 50, Appendix B, to control the requirements for all review and audit functions, except with respect to those associated with the security and emergency plans described later.

The security and emergency plan review and audit functions are relocated to their respective plans. Such an approach would result in an equivalent level of regulatory authority while providing for a more appropriate change control process. The level of safety of plant operation is unaffected, and NRC and SCE resources associated with processing license amendments to this administrative control are optimized. The following points summarize the reasons for removing the review and audit requirements from the improved SONGS TS.

- The on-site review function, composition, alternate membership, meeting frequency, quorum, responsibilities, authority, and records are all covered in equivalent detail in ANSI N18.7-1976. These requirements are in the QA Program and change control is provided by 10 CFR 50.54(a).

- The off-site review group is also addressed, although with less detail, in ANSI N18.7-1976. The QA Program includes the requirements for the off-site review group. Therefore, duplicating the review and audit function of the off-site review group in the improved SONGS TS is unnecessary.

- Audit requirements are specified in the QA Program to satisfy 10 CFR Part 50, Appendix B, Criterion XVIII. Audits are also covered by ANSI N18.7, ANSI N45.2, 10 CFR 50.54(t), 10 CFR 50.54(p), and 10 CFR Part 73. Therefore, duplication of these regulatory requirements does not enhance the level of safety of the SONGS plant, nor is a restatement of the provisions relating to audits necessary to assure safe operation of SONGS.

The licensee has proposed to relocate those provisions that are not otherwise covered by regulatory requirements to the Topical Quality Assurance Manual (TQAM), the security plan or the emergency plan. The staff concludes that sufficient regulatory controls exist under 10 CFR 50.54(a) for changes to the TQAM, the security plan and the emergency plan such that removing these

provisions from the TS, and relocating them to the TQAM is acceptable.

Reportable Occurrence Action

The licensee proposes that the requirement in existing TS 6.6, that the Commission be notified of all reportable events, not be retained in the TS. Requirements are provided in 10 CFR 50.73(a)(2) for the licensee to submit a Licensee Event Report (LER) for all reportable events specified in 10 CFR 50.73. The reports are required to be submitted within 30 days and will contain the same type of information required by existing TS 6.6.1.a. The above requirements are included in the licensee procedures that implement 10 CFR 50.72, 10 CFR 50.73, and the LCS. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

Security Plan Implementation and Emergency Plan Implementation

The licensee proposes to relocate the requirements to establish, implement, and maintain procedures related to the Emergency Plan (existing TS 6.8.1.e) and Security Plan (existing TS 6.8.1.d). Since the Security Plan requirements are specified in 10 CFR 50.54, 73.40, 73.55, and 73.56 and the Emergency Plan requirements are specified in 10 CFR 50.54 and 10 CFR Part 50, Appendix E, Section V, the staff concluded in Generic Letter 93-07, "Relocation of Administrative Controls for Emergency and Security Plans," that it is acceptable to remove these requirements from the TS and relocate them to their respective plans.

The requirements in TS 6.8.1.d for the review of the security program and implementing procedures and in TS 6.8.1.e for the review of the station emergency plan and implementing procedures will be included in their respective plans. Further changes in these review requirements must be made in accordance with 10 CFR 50.54(p) for the Security Plan and 10 CFR 50.54(q) for the Emergency Plan. The staff concludes that, in conjunction with this change to the plans, the extensive requirements for emergency planning in 10 CFR 50.47, 50.54, and Part 50 Appendix E; and for security in 10 CFR 50.54 and 10 CFR Part 73; and drills, exercises, testing, and maintenance of the program in 10 CFR 73.55, provide adequate assurance that the objective of the previous TS for a periodic review of the program and changes to the programs will be met. The staff concludes that other regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

Review and Approval Process and Temporary Change Process

The licensee is proposing to relocate both the review and approval process (existing TS 6.8.2) and the temporary change process (existing TS 6.8.3) for procedures to the TQAM.

The requirement for procedure control is mandated by 10 CFR Part 50, Appendix B, Criterion II and Criterion V. ANSI N18.7-1976, which is an NRC staff-endorsed document used in the development of many licensee QA plans,

also contains specific requirements related to procedures. The licensee has committed to follow ANSI N18.7-1976 as a means to comply with 10 CFR Part 50, Appendix B. ANSI N18.7-1976, Section 5.2.2 discusses procedure adherence. This section clearly states that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. ANSI N18.7-1976 also discusses temporary changes to procedures, and requires review and approval of procedures to be defined. ANSI N18.7-1976, Section 5.2.15, describes the review, approval, and control of procedures. This section describes the requirements for the licensee's QA Program to provide measures to control and coordinate the approval and issuance of documents, including changes thereto, which prescribe all activities affecting quality. The section further states that each procedure shall be reviewed and approved prior to initial use. The required reviews are also described. ANSI N45.2-1971, Section 6, also specifies that the QA Program describe procedure requirements.

The licensee will continue to implement a QA Program in accordance with the requirements of 10 CFR Part 50, Appendix B, which provides appropriate controls for the review and approval of procedure changes. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

<u>In Plant Radiation Monitoring</u>

The In Plant Radiation Monitoring Program (NUREG-1432, Section 5.7.2.5) provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program was developed to minimize radiation exposure to plant personnel (post-accident). The In Plant Radiation Monitoring Program administrative control is not relied upon to detect a degradation of the reactor coolant system pressure boundary. The licensee has proposed to relocate these provisions to the LCS and appropriate supporting plant procedures. The staff concludes that the control of these provisions under 10 CFR 50.59 is acceptable.

<u>Radiological Environmental Monitoring Program</u>

The Radiological Environmental Monitoring Program (existing TS 6.8.4.f) requires that procedures be prepared for monitoring the radiation and radionuclides in the environs of SONGS consistent with the guidance specified in 10 CFR Part 50, Appendix I. These procedures were developed to ensure that radioactive effluents are restricted to levels as low as reasonably achievable, and have no impact on plant nuclear safety. The details and description of the program are already contained in the Offsite Dose Calculation Manual (ODCM), as specified by existing TS 6.8.4.b, and consistent with GL 89-01, "Relocation of Radiological Effluent Technical Specifications." The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

Core Operating Limits Report

The licensee has proposed relocating cycle-specific core operating limits to the Core Operating Limits Report (COLR). The COLR is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with improved TS 5.7.1.5. The requirement for operating within the limits is retained in the improved TS, while the specific limit can be relocated to the COLR as long as the methodology for determining the limits is approved by the staff. Plant operation within these limits is addressed in the following TS:

1. Improved TS 3.1.4, "Moderator Temperature Coefficient";

2. Improved TS 3.1.7, "Regulating CEA Insertion Limits";

3. Improved TS 3.1.8, "Part Length Control Element Assembly Insertion Limits";

4. Improved TS 3.2.1, "Linear Heat Rate";

5. Improved TS 3.2.4, "Departure From Nucleate Boiling Ratio";

6. Improved TS 3.2.5, "Axial Shape Index";

7. Improved TS 3.9.1, "Boron Concentration".

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the Topical Reports identified in improved TS 5.7.1.5. The core operating limits are established such that all applicable limits of the safety analysis are met. The above listed specifications are revised to state that the values of the cycle-specific parameters shall be maintained within the limits defined in the COLR.

The use of a COLR is being adopted in accordance with the guidance and recommendation of GL 88-16 and implemented with the adoption of the improved TS. Changes to the COLR are made in accordance with the staff approved topical reports and methodologies referenced in the administrative controls section, as described above. Changes in format and content must have prior approval from the NRC, as stipulated in improved TS 5.7.1.5. Other changes would be governed by the 10 CFR 50.59 process. The staff concludes that the use of the COLR as specified above, provides sufficient control of the cycle-specific core operating limits, and removing them from the TS is acceptable.

ECCS Actuation Special Report

The licensee proposes to delete TS 3.5.2, "Required Action b," to submit a special report for ECCS actuation, from the TS. Requirements are provided in 10 CFR 50.73(a)(2)(iv) for the licensee to submit a Licensee Event Report in the event of an ECCS actuation. The report is required to be submitted within 30 days and will contain the same type of information as the special report.

The above requirements are included in the SONGS Administrative Instruction for 10 CFR 50.72 and 10 CFR 50.73. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

Record Retention

The licensee proposes that the requirements on record retention (existing TS 6.10) be deleted or relocated from the improved SONGS TS on the basis that they are adequately addressed by the QA Program (10 CFR Part 50, Appendix B, Criteria XVII) and the related TQAM. Facility records document appropriate station operations and activities. Retention of these records provides documentation retrievability for review of compliance with requirements and regulations. Post-compliance review of records does not directly assure operation of the facility in a safe manner, as activities described in these documents have already been performed. In addition, numerous other regulations such as 10 CFR Part 20, Subpart L, and 10 CFR 50.71 require the retention of certain records related to operation of the nuclear plant. The staff concludes that these regulatory requirements provide sufficient control of these recordkeeping provisions and removing them from the TS is acceptable.

Inservice Inspection Program

The Inservice Inspection Program required by ASME Section XI is relocated since the requirement is duplicated in 10 CFR 50.55a. These regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

Radiation Protection Program

The licensee proposes to relocate the TS program description for the Radiation Protection Program (existing TS 6.11). The Radiation Protection Program requires procedures to be prepared for personnel radiation protection consistent with the requirements of 10 CFR Part 20. The requirement to have procedures to implement Part 20 is also contained within 10 CFR 20.1101(b). Periodic review of these procedures is addressed under 10 CFR 20.1101(c). The program requirements specified above are described in UFSAR.

The licensee has proposed to relocate these provisions to the LCS with supporting plant procedures. The staff concludes that the requirements of 10 CFR Part 20 provide sufficient control of these provisions, and that 10 CFR 50.59 provides adequate controls for changes to the implementing provisions in the LCS and plant procedures.

Process Control Program

The licensee proposes to relocate the TS program description for the Process Control Program (PCP)(existing TS 6.13). The PCP is described in the LCS and TQAM. The PCP implements the requirements of 10 CFR Part 20, 10 CFR Part 61, and 10 CFR Part 71. The staff concludes that the regulatory controls for the TQAM provide sufficient control of these requirements and removing these

provisions from the TS is acceptable.

Minimum Shift Composition, Organizational and Control Room Figures

The licensee proposes that the Minimum Shift Crew Composition Table (6.2-3), Offsite and Unit Organization Figures (6.2-1 and 6.2-2), and the Control Room Area Figure (6.2-3) not be retained in the TS. Requirements are provided in 10 CFR 50.54(k), (l) and (m) for shift complement regarding licensed operators. The regulations describe the minimum shift composition for operating modes, as well as for cold shutdown and refueling. The requirements in this specification and the associated table are located in the SONGS Administrative Instructions for 10 CFR 50.54 and in the Emergency Plan. Additionally, new Specifications 5.1.3 and 5.2.2 of the improved SONGS TS specify when licensed and non-licensed operators are required to be in the control room. The organization figures and control room area figure are unnecessary since the organization and control room are adequately described in the licensee's UFSAR. Repeating the organizational structure and control room configuration only increases the administrative burden on the facility with no resulting benefit. In addition, removal of the organizational figures is consistent with NUREG-1432 and Generic Letter 88-06, "Removal of Organization Charts from Technical Specification Administrative Control Requirements." Plant safety is not compromised by these proposed deletions from the TS. Therefore, control of these provisions under 10 CFR 50.59 is acceptable. The staff concludes that the regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable.

The above relocated requirements are not required to be in the TS under 10 CFR 50.36(c)(5), and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 and 10 CFR 50.54(a) and the other regulations set forth above. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the LCS, plant procedures, TQAM, ODCM, COLR, Security Plan, or Emergency Plan, as applicable.

b. Less Restrictive Requirements

Startup Report

The requirement to submit a startup report has been deleted from the improved SONGS TS. The report was a summary of plant startup and power escalation testing following receipt of the operating license, an increase in licensed power level, the installation of nuclear fuel with a different design or manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of SONGS. The report provided a mechanism for the staff to review the appropriateness of licensee activities after-the-fact, but contained no requirement for staff approval. The approved 10 CFR Part 50, Appendix B, QA Plan, and Startup Test Program (UFSAR) provide assurance that the listed activities are adequately performed and that appropriate corrective actions, if required, are taken.

There is no requirement for the staff to approve the report, and the startup report is not necessary to assure operation of SONGS in a safe manner. Therefore, the removal of this requirement is acceptable.

Staff Work Hour Limitations

Improved TS 5.2.2.e requires that the licensee provide administrative procedures to control overtime worked by personnel who perform safety-related functions; this revises the existing TS which provide specific overtime limits.

On February 18, 1982, the NRC published "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors," (47 FR 23836). In June 1982, the NRC revised the policy slightly and subsequently disseminated the revision in GL 82-12, "Nuclear Power Plant Staff Working Hours," which recommended that licensees incorporate specific overtime limits in the technical specifications to minimize the potential for operator errors resulting from fatigue. The staff subsequently determined that few events at U.S. nuclear plants have been attributed to inadequate control of working hours, and that licensees can adequately control working hours with administrative procedures. This approach is consistent with Action Item I.A.1.3.1, "Limit Overtime," of NUREG-0737, "Clarification of TMI Action Plan Requirements."

The Staff has determined, on a generic basis, that specific overtime limits need not be specified in TS; this change is being incorporated on a generic basis in a revision to the improved STS (NUREG 1432). The staff concludes that control of this matter through administrative procedures provides reasonable assurance that personnel overtime will not jeopardize safe plant operation, and that specific overtime limits are not required to be in the technical specifications under 10 CFR 50.36(c)(5). The staff further concludes that the specific overtime limits and associated procedures to minimize the potential for operator fatigue can be described in the UFSAR, or other licensee controlled documents incorporated in the UFSAR by reference, for which future changes can be made pursuant to 10 CFR 50.59. Accordingly, this change is acceptable for SONGS.

The above less restrictive requirements have been reviewed by the staff and found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

c. More Restrictive Requirements

Safety Function Determination Program and Bases Control Program

The licensee has adopted the Safety Function Determination Program (improved TS 5.5.6) and Technical Specification Bases Control Program (improved TS 5.5.4), two more restrictive conditions than are required by the existing TS. The Safety Function Determination Program is included to support

implementation of the support system operability characteristics of the improved TS. The Bases Control Program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the improved TS Bases.

The staff has reviewed this more restrictive requirement and concludes that it results in an enhancement to the improved TS. Therefore, this more restrictive requirement is acceptable.

d. Administrative Changes

Safety Limit Violation

The licensee has proposed to move the following existing administrative control provision to another section of the improved STS:

Existing TS Section	Title	Improved TS Section
6.7	Safety Limit Violation	2.2

In accordance with the guidance in NUREG-1432, the licensee has proposed to move or reorganize all or portions of existing TS 6.7, "Safety Limit Violation," within the improved TS. Existing TS 6.7 specifies the actions to be taken in the event a safety limit is violated. The improved TS addresses this item within Specification 2.2, "SL Violations." Thus the safety limit violation actions in the existing TS are effectively retained within the improved TS. This change is considered an administrative change in the location of the requirements within the TS and is, therefore, acceptable.

Programs

The following programs have been moved to the improved TS administrative controls section from other areas in the existing TS. Some of the retained requirements are relocated details associated with surveillance requirements. These programs ensure that provisions necessary to satisfy the policy statement guidance are included in the TS, and that procedures outside the TS are adequately defined.

- Steam Generator Tube Surveillance Program
- Diesel Fuel Oil Testing Program
- Ventilation Filter Testing Program
- Pre-stressed Concrete Containment Tendon Surveillance Program
- Inservice Testing Program
- Reactor Coolant Pump Flywheel Inspection Program
- Radioactive Effluent Controls Program
- Component Cyclic or Transient Limit Program
- Explosive Gas and Storage Tank Radioactivity Monitoring Program

Specification 5.7, "High Radiation Area," (existing TS 6.12) has been revised to ensure consistency with the changes to 10 CFR Part 20, "Standards for Protection Against Radiation," and 10 CFR 50.36a, "Technical specifications on

effluents from nuclear power reactors," which became effective on June 20, 1991, and October 1, 1992, respectively.

The above changes result in comparable restrictions to the current requirements, or they represent an enhanced presentation of the existing TS intent. Accordingly, the improved TS changes are purely administrative and are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendment. [The State official had no comments.] [The state official provided comments ... .] [This action will performed when the staff issues the final TS.]

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 49434). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in record keeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The improved SONGS TS provide clearer, more readily understandable requirements to ensure safe operation of the plant. The staff concludes that they satisfy the guidance in the Commission's policy statement with regard to the content of technical specifications, and conform to the model provided in NUREG-1432 with appropriate modifications for plant-specific considerations. The staff further concludes that the improved SONGS TS satisfy Section 182a of the Atomic Energy Act, 10 CFR 50.36 and other applicable standards. On this basis, the staff concludes that the proposed improved SONGS TS are acceptable.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and, (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Table 1

Principal Contributor: T. R. Tjader

Date: August 10, 1995

Table 1

Summary of
Relocated SONGS Technical Specifications

Existing TS	Title	Relocation Document	Relocation Control
2.2	3 LSSS (Instrumentation)	LCS	§50.59
4.0.5	Inservice Inspection and Testing Programs	LCS	§50.55a §50.59
3/4.1.3.2	Position Indicator Channels--Operating	Bases & LCS	§50.59
3/4.1.3.3	Position Indicator Channels--Shutdown	Bases & LCS	§50.59
3.1.1.3	Moderator Temperature Coefficient, Cycle Specific Parameter	COLR	§50.59
3.1.3.6	Regulating CEA Insertion Limits, Cycle Specific Parameter	COLR	§50.59
3.1.3.7	Part Length CEA Insertion Limits, Cycle Specific Parameter	COLR	§50.59
3/4.2.1	Linear Heat Rate, Cycle Specific Parameter	COLR	§50.59
3/4.2.4	DNBR Cycle Specific Parameter	COLR	§50.59
3.2.7	Axial Shape Index, Cycle Specific Parameter	COLR	§50.59
3/4.3.1	Loss of Load Trip	LCS	§50.59
3/4.3.1	Steam Generator High Level Trip	LCS	§50.59
3/4.3.2(10)	Toxic Gas Isolation Signal	LCS & UFSAR	§50.59
3/4.3.3.1	Radiation Monitoring Instrumentation	LCS & UFSAR	§50.59
3/4.3.3.2	Incore Detectors	LCS & UFSAR	§50.59
3/4.3.3.3	Seismic Instrumentation	LCS & UFSAR	§50.59
3/4.3.3.4	Meteorological Instrumentation	LCS & UFSAR	§50.59
3/4.3.3.7	Fire Detection Instrumentation	LCS & UFSAR	§50.59
3/4.3.3.9	Explosive Gas Monitoring Instrumentation	LCS & UFSAR	§50.59
3/4.3.3.10	Vibration and Loose-Parts Monitoring System	LCS & UFSAR	§50.59

Existing TS	Title	Relocation Document	Relocation Control
3/4.3.4	Turbine Overspeed Protection	LCS & UFSAR	§50.59
Tables 3.3-2 & 3.3.5	Response Time Tests	LCS	§50.59
3/4.4.6	Chemistry	LCS	§50.59
3.4.9	Structural Integrity	UFSAR	§50.59 & §50.55(a)
3/4.4.10	Reactor Coolant Gas Vent System	LCS	§50.59
3/4.5.1	Safety Injection Tank SRs 4.5.1.c & 4.5.1.e	LCS	§50.59
3/4.5.2 & 3	ECCS Subsystems Reporting Requirement	LCS	§50.59
Table 3.6-1	Containment Isolation Valves	LCS	§50.59
3/4.7.2	Steam Generator Pressure/Temperature Limitations	LCS	§50.59 & §50.55(a)
3/4.7.6	Snubbers	LCS	§50.59 & §50.55(a)
3/4.7.7	Sealed Source Contamination	LCS	§50.59 & 10CFR20
3/4.7.8	Fire Suppression System	LCS	§50.59
3/4.7.8.3	Fire Hose Stations	LCS	§50.59
3/4.7.9	Fire Rated Assemblies	LCS	§50.59
3/4.8.4.1	Containment Penetration Conductor Overcurrent Protective Devices	LCS	§50.59
3/4.8.4.2	Motor Operated Valves Thermal Overload Protective Devices	LCS	§50.59
3/4.9.1	Boron Concentration, Cycle Specific Parameter	COLR	§50.59
3/4.9.3	Decay Time	LCS	§50.59
3/4.9.5	Communication	LCS	§50.59
3/4.9.6	Refueling Machine	LCS	§50.59
3/4.9.7	Fuel Handling Machine--Spent Fuel Pool Building	LCS & UFSAR	§50.59
5.1.3	Site Boundary for Gaseous Effluents	UFSAR	§50.59

Existing TS	Title	Relocation Document	Relocation Control
5.1.4	Site Boundary for Liquid Effluents	UFSAR	§50.59
5.2.1	Containment Configuration	UFSAR	§50.59
5.2.2	Design Pressure and Temperature	UFSAR	§50.59
5.4.1	Reactor Coolant System Design Pressure and Temperature	UFSAR	§50.59
5.4.2	Reactor Coolant System Volume	UFSAR	§50.59
5.5.1	Meteorological Tower Location	UFSAR	§50.59
5.7.1	Component Cyclic or Transient Limit	UFSAR	§50.59
Figures 5.6-3 & 5.6-4	Fuel Storage Patterns for Region II Racks	LCS	§50.59
6.2.2.d	SRO Present During Core Alterations	LCS	§50.59 §50.54
6.2.2.e	Fire Brigade	LCS	§50.59
6.4	Training	LCS	§50.59
6.5	Review and Audits	TQAM	§50.54(a)
6.6	Reportable Occurrence Action	LCS	§50.59
6.8.1.d	Security Plan Implementation	Security Plan	§50.54(p)
6.8.1.e	Emergency Plan Implementation	Emergency Plan	§50.54(q)
6.8.2	Review and Approval Process	TQAM	§50.54(a)
6.8.3	Temporary Change Process	TQAM	§50.54(a)
6.8.4.b	In Plant Radiation Monitoring	LCS	§50.59
6.8.4.f	Radiological Environmental Monitoring Program	ODCM	Part 50 App. I
6.9.2	ECCS Actuation Special Report	LCS	§50.59 §50.72 §50.73
6.10	Record Retention	LCS	§50.59
6.10.2.h	Inservice Inspection Program Requirements	LCS & ASME Sec. XI	§50.59 §50.55(a)
6.11	Radiation Protection Program	LCS	§50.59 & 10CFR20

Existing TS	Title	Relocation Document	Relocation Control
6.13	Process Control Program	LCS & TQAM	§50.59 & §50.54(a)
Figure 6.2-1	Offsite Organization	LCS	§50.59 & §50.54
Figure 6.2-2	Unit Organization	LCS	§50.59 & §50.54
Table 6.2-1	Minimum Shift Crew Composition	LCS	§50.59 & §50.54
Figure 6.2-3	Control Room Area	LCS	§50.59

REFERENCE L

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

February 9, 2000

RECEIVED
FEB 14 2000
NUCLEAR LICENSING

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

SUBJECT: NRC INSPECTION REPORT NO. 50-361/2000-01; 50-362/2000-01

Dear Mr. Ray:

This refers to the inspection conducted on January 10-14, 2000, at the San Onofre Nuclear Generating Station, Units 2 and 3 facilities. The purpose of the inspection was to review your solid radioactive waste management and radioactive material transportation programs. The enclosed report presents the results of this inspection. The programs reviewed were implemented properly.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Gail M. Good

Gail M. Good, Chief
Plant Support Branch
Division of Reactor Safety

Docket Nos.: 50-361
50-362
License Nos.: NPF-10
NPF-15

Enclosure:
NRC Inspection Report No.
50-361/2000-01; 50-362/2000-01

COM
3/3 BR

cc w/enclosure:
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-361 50-362
License Nos.:	NPF-10 NPF-15
Report No.:	50-361/2000-01 50-362/2000-01
Licensee:	Southern California Edison Co.
Facility:	San Onofre Nuclear Generating Station, Units 2 and 3
Location:	5000 S. Pacific Coast Hwy. San Clemente, California
Dates:	January 10-14, 2000
Inspector(s):	James S. Dodson, Radiation Specialist Plant Support Branch
Approved By:	Gail M. Good, Chief Plant Support Branch, Division of Reactor Safety
Attachment:	Supplemental Information

EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Units 2 and 3
NRC Inspection Report No. 50-361/2000-01; 50-362/2000-01

The NRC conducted an inspection of the solid radioactive waste management and radioactive material transportation programs. Areas reviewed included: the solid radioactive waste management program, radioactive material transportation program, facilities and equipment, staff knowledge and performance, staff training and qualifications, and nuclear oversight activities.

Plant Support

- The licensee met regulatory requirements associated with the solid radioactive waste management program. Radioactive material was correctly stored and controlled. Radioactive waste was correctly classified and stabilized for burial. Waste manifests were prepared in accordance with regulatory requirements (Section R1.1).

- The licensee met regulatory requirements for the packaging and shipping of radioactive materials and radioactive waste. Packages were properly marked and labeled, and radioactive material transport vehicles were properly placarded. Shipping documentation, emergency response information, and instructions were prepared in accordance with regulatory requirements (Section R1.2).

- There were no significant changes to the solid radwaste facilities, equipment, and the process control program. Housekeeping and material condition were acceptable (Section R2).

- The individuals responsible for training and oversight were knowledgeable of regulatory and procedural requirements. Radwaste supervisors, foreman, and technicians were knowledgeable of transfer, packaging, waste classification, marking, labeling, storage, documentation, vendor supplied computer software operation, and radioactive material transportation regulations (Section R4).

- The licensee provided solid radwaste and transportation personnel with the appropriate training and retraining (Section R5).

- The nuclear oversight organization and health physics division management provided effective oversight of radioactive waste management and transportation activities. Field observations and self assessments were comprehensive and provided adequate depth to identify problems and provide oversight of radwaste management and transportation activities (Section R7).

Report Details

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Solid Radioactive Waste Management Program

a. Inspection Scope (86750)

The inspector interviewed licensee personnel and reviewed the following program areas:

- Waste storage and container accountability
- Waste stream sampling results
- Waste classification
- Waste characteristics
- Waste shipment manifests

b. Observations and Findings

Waste Storage and Container Accountability

During tours of the radiological controlled areas, the inspector confirmed that radioactive waste was stored in accordance with commitments in the Updated Final Safety Analysis Report, Chapter 11.4. The inspector verified that randomly selected radioactive material containers were properly labeled and confirmed that the licensee's tracking system listed the correct location and status of the containers.

Waste Stream Sampling

The inspector reviewed the analysis results and the associated evaluations for the identified waste streams. The inspector determined that sampling and analyses were completed at the required intervals. Analyses were performed by a vendor laboratory and the licensee as required by procedure. The scaling factors used in the vendor-supplied computer code were verified with current analysis results as required by procedure.

Waste Classification

The licensee used a vendor supplied computer software code to perform the calculations necessary to classify radioactive waste. The inspector reviewed sample results from eleven randomly selected radioactive waste shipments and confirmed that the waste shipments were properly classified in accordance with 10 CFR 61.55.

Waste Characteristics

Through record review and observations, the inspector confirmed that the licensee met the structural integrity requirements of 10 CFR 61.56 (b)(1) by using high integrity containers. No adverse findings related to the licensee's radioactive waste characteristics had been identified by burial site representatives.

Manifests

The inspector reviewed eleven randomly selected shipping documentation packages and confirmed that the licensee prepared manifests included the information required by 10 CFR Part 20, Appendix G. The shipment manifests included a certification that the transported material was properly classified, described, packaged, marked, labeled, and that it was in proper condition for transport. The certification was signed and dated by an authorized licensee representative.

c. Conclusions

The licensee met regulatory requirements associated with the solid radioactive waste management program. Radioactive material was correctly stored and controlled. Radioactive waste was correctly classified and stabilized for burial. Waste manifests were prepared in accordance with regulatory requirements.

R1.2 Radioactive Material Transportation Program

a. Inspection Scope (86750)

The inspector interviewed licensee personnel and reviewed selected examples of the following materials: packaging; loading, storage, blocking, and bracing; vehicle placarding; driver instructions; emergency response information; radiation surveys; shipping paper documentation; and package marking and labeling.

b. Observations and Findings

Packaging

The inspector reviewed A_2 values for selected radionuclides in the licensee's waste classification computer data base and confirmed that they matched the values in 49 CFR 173.435. The licensee maintained records that documented Type B packages used by the licensee were designed to meet the applicable requirements specified in 10 CFR 71.12.

Radiation Surveys

Radiation surveys were conducted by the inspector during tours of the radioactive waste processing and storage facilities to ensure that external radiation levels were within the allowable limits of 49 CFR 173.441. The inspector verified that radioactive waste

package external radiation levels were within allowable limits for randomly selected packages.

Package Marking, Labeling, and Loading and Vehicle Placarding

The inspector conducted field observations and reviewed randomly selected shipping documentation packages. The inspector determined that packages prepared for transport were properly marked and labeled and that radioactive material transport vehicles were properly placarded in accordance with 49 CFR 172.504 and 172.506.

Shipping Papers and Documentation

The inspector reviewed eleven randomly selected examples of shipping documentation and confirmed that the licensee provided the shipping papers and information required by 49 CFR Part 172, Subpart C, and the emergency response information required by 49 CFR Part 172, Subpart G.

Additionally, the inspector verified that shipping permits, licenses, certificates of compliance, user lists, and shipping regulations were current. No problems were noted.

c. Conclusions

The licensee met regulatory requirements for the packaging and shipping of radioactive materials and radioactive waste. Packages were properly marked and labeled, and radioactive material transport vehicles were properly placarded. Shipping documentation and emergency response information and instructions were prepared in accordance with regulatory requirements.

R2 Status of Radiological Protection and Chemistry Facilities and Equipment

a. Inspection Scope (86750)

The inspector reviewed the Updated Final Safety Analysis Report, Chapter 11.4, and toured the auxiliary building, on-site radwaste facilities, truck bay and truck bay storage areas, multipurpose handling facility, and south services repair center.

b. Observations and Findings

The licensee made no significant changes to solid radwaste facilities. There were no changes in equipment or the Process Control Program. The inspector noted no deviations from commitments in the Updated Final Safety Analysis Report, Chapter 11.4.

During the tours of the auxiliary building and on-site radwaste facilities, the inspector noted that the housekeeping was acceptable.

To selectively review the material conditions in the licensee's radwaste facilities, the inspector conducted a walkdown of accessible radwaste tanks and resin transfer system components including, pumps, valves, and associated piping. Material condition was acceptable.

c. Conclusions

There were no significant changes to the solid radwaste facilities, equipment, and the process control program. Housekeeping and material condition were acceptable.

R4 Staff Knowledge and Performance

a. Inspection Scope (86750)

The inspector interviewed a nuclear oversight auditor, health physics self-assessment supervisor, radwaste general foreman, radwaste shipping and receiving supervisor, two radioactive material control technicians, radwaste supervisor, and the radwaste training instructor involved in the radioactive material transportation program.

b. Observations and Findings

The nuclear oversight auditor who conducted quality assurance audits was knowledgeable of regulatory and procedural requirements for solid radioactive waste management and transportation. The radwaste training instructor had a good understanding of procedural, regulatory, and training/retraining requirements. The radwaste supervisors, foreman, and technicians responsible for shipping were knowledgeable of radioactive waste classification, packaging, marking, labeling, storage, documentation, vendor supplied computer software operation, and radioactive material transportation regulations.

c. Conclusions

The individuals responsible for training and oversight were knowledgeable of regulatory and procedural requirements. Radwaste supervisors, foreman, and technicians were knowledgeable of transfer, packaging, waste classification, marking, labeling, storage, documentation, vendor supplied computer software operation, and radioactive material transportation regulations.

R5 Staff Training and Qualification

a. Inspection Scope (86750)

The inspector reviewed training lesson plans and verified current and past training records for the nuclear oversight auditors, rawaste supervisors, radwaste foreman, radioactive material control technicians, radwaste support personnel, and the radwaste training instructor.

b. Observations and Findings

Training lesson plans and records confirmed that the licensee provided the appropriate training and periodic retraining in Department of Transportation and NRC regulatory requirements. Additionally, the training and retraining programs included instructions and a review of procedures for personnel involved in the transfer, storage, packaging, and transport of radioactive material.

c. Conclusions

The licensee provided solid radwaste and transportation personnel with the appropriate training and retraining.

R7 Quality Assurance in Radiological Protection and Chemistry Activities

a. Inspection Scope (86750)

The inspector interviewed licensee personnel and reviewed the following items: nuclear oversight audit, nuclear oversight surveillance/observations, health physics division management field observations, self-assessments, and action requests.

b. Observations and Findings

There were no audits conducted by the licensee since the previous NRC inspection. There is an audit scheduled for March 2000 in the area of solid radioactive waste management and transportation. The licensee conducted 63 nuclear oversight field observations, 5 division self-assessment audits, and 18 division management field observations. The field observations and self assessments were comprehensive and provided adequate depth to identify problems and provide oversight of radwaste management and transportation activities. Problems identified were placed in the corrective action program.

The inspector reviewed a summary of action requests relating to solid radioactive waste and transportation and selected 14 action request packages for a detailed review. The inspector verified that the corrective actions were appropriate and completed in a timely manner.

c. Conclusions

The nuclear oversight organization and health physics division management provided effective oversight of radioactive waste management and transportation activities. Field observations and self assessments were comprehensive and provided adequate depth to identify problems and provide oversight of radwaste management and transportation activities.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at an exit meeting on January 14, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Krieger, Vice President
D. Nunn, Vice President
J. Madigan, Manager, Health Physics
K. Slagle, Manager, Nuclear Oversight
R. Sandstrom, Manager, Nuclear Training Division
E. Scherer, Manager, Nuclear Regulatory Affairs
G. Cook, Supervisor, Nuclear Regulatory Affairs
M. McBrearty, Engineer, Nuclear Regulatory Affairs
P. Elliott, Supervisor, Radwaste
M. Farmer, General Foreman, Radwaste
E. Bennett, Auditor, Nuclear Oversight
S. Stinson, Supervisor, Nuclear Training Division
C. Ahola, Supervisor, Radwaste Shipping and Receiving
S. Schofield, Supervisor, Health Physics Self Assessment
R. Morrison, Training Instructor
A. Gray, Supervisor, Radwaste
A. Martinez, General Foreman, Health Physics

NRC

J. Kramer, Resident Inspector

INSPECTION PROCEDURES USED

86750	Solid Radioactive Waste Management and Transportation of Radioactive Material

ITEMS OPENED AND CLOSED

Opened and Closed

None

PARTIAL LIST OF DOCUMENTS REVIEWED

Summary list of action requests relating to the inspection areas (8/98 to 1/7/2000)

Action Requests

AR-990700158-01
AR-981101041-01
AR-981101041-05
AR-981101041-03
AR-980801365-01

Nuclear Oversight Division Observations (8/98 to 1/7/2000)

Health Physics Division Observations (8/98 to 1/7/2000)

Health Physics Division Self Assessment (3Q98, 4Q98, 1Q99, 2Q99, 3Q99)

Procedures

SO123-VII-20	Health Physics Program, Revision 5
SO123-VII-8	Control of Radioactive Material, Revision 8
SO123-VII-8.1	Solid Radioactive Waste Sampling for Classification and Typification, Revision 17
SO123-VII-8.1.2	Radioactive Materials Curie Content Determination, Revision 4
SO123-VII-8.1.4	Solid Radioactive Waste Packaging for Class A Unstable Material, Revision 8
SO123-VII-8.1.5	Loading of Radioactive Material for Shipment, Revision 1
SO123-VII-8.1.6	Radioactive Waste Package Accountability, Revision 4
SO123-VII-8.2.12	Shipment of Radioactive Waste for Land Disposal to the Envirocare Facility at Clive, Utah, Revision 0
SO123-VII-8.3.1	Multipurpose Handling Facility (MPHF) Operations, Revision 4
SO123-VII-8.16	Radioactive Equipment and Material Storage (REMS), Revision 4
SO123-VII-8.17	Radman Software Package, Revision 3

Training Lesson Plans

RC7919	Overview of Packaging & Transportation of Radioactive Waste, Revision 1

REFERENCE M

5.0.103.2 Programs

The following programs shall be established, implemented, and maintained.

5.0.103.2.1 Radiation Protection Program

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

5.0.103.2.2 Process Control Program (PCP)

The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes will be accomplished to ensure compliance with 10 CFR 20, 10 CFR 61, and 10 CFR 71; state regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

5.0.103.2.2.1 Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained in accordance with Topical Report SCE-1A, Quality Assurance Program. This documentation shall contain:

   1. Sufficient information to support the change(s) and appropriate analyses or evaluations justifying the change(s); and

   2. A determination that the change(s) maintain the overall conformance of the solidified waste product to the existing requirements of Federal, State, or other applicable regulations; and

   3. Documentation of the fact that the change has been reviewed and found acceptable pursuant to Topical Report SCE-1A, Quality Assurance Program.

b. Shall be effective after review pursuant to Topical Report SCE-1A, Quality Assurance Program and approval by the Vice President-Nuclear Generation or his designee.

5.0.103.2 Programs

The following programs shall be established, implemented, and maintained.

5.0.103.2.1 Radiation Protection Program

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

5.0.103.2.2 Process Control Program (PCP)

The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes will be accomplished to ensure compliance with 10 CFR 20, 10 CFR 61, and 10 CFR 71; state regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

5.0.103.2.2.1 Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained in accordance with Topical Report SCE-1A, Quality Assurance Program. This documentation shall contain:

   1. Sufficient information to support the change(s) and appropriate analyses or evaluations justifying the change(s); and

   2. A determination that the change(s) maintain the overall conformance of the solidified waste product to the existing requirements of Federal, State, or other applicable regulations; and

   3. Documentation of the fact that the change has been reviewed and found acceptable pursuant to Topical Report SCE-1A, Quality Assurance Program.

b. Shall be effective after review pursuant to Topical Report SCE-1A, Quality Assurance Program and approval by the Vice President-Nuclear Generation or his designee.

REFERENCE N

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 & 3
DESIGN CHANGE PACKAGE
TITLE SHEET
JOB NO. 10079

DCP NO.	790.1M
REV	1

ORIGINATING DISCIPLINE MECHANICAL

RESPONSIBLE ENGINEER JENIFER HEDRICK DATE 7-16-84

EFFECTIVITY DATE OF PACKAGE 8-7-84

UNIT NO Comm. Q-CLASS IV SEISMIC CAT. III HIGHEST Q-CLASS OF DWGS IN DCP I

SAR CHANGE REQUIRED? [ ] YES [X] NO 3007/13/84 REPAIR MODE/OPERATIONS PRIORITY CODE 0.1

FHA CHANGE REQUIRED? [ ] YES [X] NO 3007/13/84 STATION MANUAL CHANGE REQUIRED? [ ] YES [X] NO

TECH SPEC CHANGE REQUIRED? [ ] YES [X] NO 3007/13/84 TECHNICAL MANUAL CHANGE REQUIRED? [ ] YES [X] NO

STARTUP SYSTEM AFFECTED: GHA GKB

BASIS OF CHANGE:

[ ] FCR NO. ______ [ ] NCR NO. ______ [ ] SPR NO. ______
STARTUP PROBLEM REPORT NO

[ ] SUPPLIER, NAME & DATE OF LETTER

[ ] CLIENT, DATE & LOG NO. OF LETTER

[X] OTHER BE-8481, 7-12-84

RECEIVED
CDM SITE

CHANGE CATEGORY: [X] WITHIN SCOPE [ ] SCOPE CHANGE [ ] BACK CHARGE

[X] TREND NO. 2850 [X] OTHER ICR 790

TITLE OF PROBLEM/CHANGES:

Modifications to the radwaste loading dock area / supplement DCP 790.1M Revision 0 to show a change in truck door AR-309 vendors, editorial changes to the instrument index and the addition of fire dampers.

APPROVALS:

BPC GROUP SUPERVISOR PDW/Gary Thomas DATE 8-6-84

PFC REQUIRED? YES [X] NO [ ] SCE GROUP LEADER A. D. Sietos DATE 8-3-84

PROJECT ENGINEER E. Hatter DATE 8-7-84

QA/QC Closenwald DATE 8/7/84

NOTIFICATION OF CHANGE COMPLETION:

RECEIVED
AUG 07 1984
SONGS 2 & 3
Project Control Center
Bechtel Power Corp.

FIELD CONSTRUCTION MANAGER/(DESIGNEE) ______ DATE ______

HAZARDS REQUIREMENTS MET AND APPROVED (SEE DCP HAZARDS REQUIREMENTS SHEET)

ENGINEERING GROUP SUPERVISOR ______ DATE ______

DCP# 790.1M Rev / Sheet / of 25 (01)

PF-2353 (10079) 6/84

Design Change Basis

1. Description of Change:

Make design changes to the Radwaste building Loading Dock area to allow the following:

1. Radwaste processing in the loading area with the main truck door closed.

2. Maintenance of the Radwaste building pressure boundary with the main truck door open.

These will be accomplished by making the following changes:

A. Provide motorized automatic dampers interlocked with the truck door open/close position. The dampers will be equipped with indicating lights and annuciators.

B. Padlock interior roll up doors and one leaf of each interior swing door.

C. Modify the main truck door to add a self closing type personnel door.

D. The indicating lights and annunciators will be located on a new panel (2/3 L 508).

(SEE FOLLOWING PAGE FOR ADDITIONAL INFORMATION)

RECEIVED
JAN 16 1984
CDM SITE

2. Technical Basis of Change:

The Radwaste Loading Dock area was originally designed to maintain the Radwaste Pressure Boundary only with the main truck door closed. This modification will automatically maintain the Radwaste Pressure Boundary with the main truck door open. In addition, adequate ventilation will be provided to allow radwaste processing in the loading area when the main door is closed. Letter, ES-21185, dated May 13, 1983 and letter, ES-21559, dated June 31, 1983 provided SCX guidance to make these design changes. The fix is to be implemented during first refueling.

Reference #31

2-3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION V
1450 MARIA LANE, SUITE 210
WALNUT CREEK, CALIFORNIA 94596

MAR 15 1982

MAR 17 REC'D

Docket No. 50-361

Southern California Edison Company
P. O. Box 800
2244 Walnut Grove Avenue
Rosemead, California 91770

Attention: Dr. L. T. Papay, Vice President
Advanced Engineering

Gentlemen:

Subject: NRC Inspection of San Onofre Unit No. 2 82-10

This refers to the routine inspection conducted by Mr. A. Chaffee of this office on January 19, 1982 to February 12, 1982 of activities authorized by NRC Construction Permit No. CPPR-97, and to the discussion of our findings held by Mr. Chaffee with Mr. H. Ray and other members of the Southern California Edison Company staff at the conclusion of the inspection.

Areas examined during this inspection are described in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

No items of noncompliance with NRC requirements were identified within the scope of this inspection.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

Should you have any questions concerning this inspection, we will be glad to discuss them with you.

Sincerely,

D. M. Sternberg

D. M. Sternberg, Chief
Reactor Operation Projects Branch

Enclosure:
NRC Inspection Report
No. 50-361/82-10

cc w/o enclosure:
R. Dietch, SCE

Phyllis Harvey 7-6-2-

C820212G- 90 -7

U. S. NUCLEAR REGULATORY COMMISSION

AD12-RB
5023

REGION V

Report No. 50-361/82-10

Docket No. 50-361 License No. CPPR-97-1 Safeguards Group ______

Licensee: Southern California Edison Company
2244 Walnut Grove Avenue
Rosemead, California 91770

Facility Name: San Onofre Nuclear Generating Station Unit 2

Inspection at: San Onofre, San Diego County, California

Inspection conducted: January 19, 1982 through February 12, 1982

Inspectors: [signature] for	March 15, 1982
A. E. Chaffee, Senior Resident Inspector, Unit 2	Date Signed
	Date Signed
Approved by: [signature]	March 16, 1982
B. Zwetzig, Chief, Reactor Projects Section I, Reactor Operations Project Branch	Date Signed

Summary:

Inspection on January 19, 1982 through February 12, 1982 (Report No. 50-361/82-10)

Areas Inspected: Routine, unannounced inspection of applicant's preoperational test program, TMI modifications, follow-up on Bulletin 80-06, safety committee activities, independent inspection effort, and general plant observations. This inspection involved 68 inspection-hours onsite by one NRC inspector.

Results: Of the six areas inspected, no items of noncompliance or deviations were identified.

SEE
AD12-RB
3-22-82
3-15-82

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MAR 24 1982
CDM/GO

RV Form 219 (2)

Phyll Hennig 7-4-82-9

DETAILS

1. Persons Contacted

   a. Southern California Edison

   *P. Croy, Configuration Control and Compliance Manager
   *H. Morgan, Station Operations Manager
   *H. Ray, Station Manager**
   *W. Marsh, Acting Station Health Physics Manager
   *C. Kergis, Unit 2/3 Operations Quality Assurance Engineer
   *D. Schone, Project Quality Assurance Supervisor
   *M. Speer, Compliance Engineer
   *F. Briggs, Compliance Engineer
   *R. Rosenblum, Unit 2 Startup Engineering Supervisor
   *B. Katz, Station Technical Manager
   *W. Moody, Deputy Station Manager
   *P. King, Project Operations Quality Assurance Supervisor

   b. Consultants

   *J. Hummer, ASTA

   The inspector also interviewed and talked with others of the applicant's employees during the course of the inspection. These included operators, startup engineers and Quality Assurance personnel.

   *Denotes those persons who attended the exit interview.

2. Plant Status

   The applicant reported Unit 2 construction to be 99 percent complete as of February 3, 1982. The applicant anticipates commencement of initial fuel loading in February.

3. NP TMI Modifications

   a. TMI Item I.A.1.1, "Shift Technical Advisors" (STA) (Closed)

   Five STA candidates completed their qualification on February 5, 1982. The STA training records appear to document satisfactory completion of the approved STA Training Program. This item is closed.

Phyll Hervey 7-6-82 - 97

b. <u>TMI Item I.C.6, "Guidance on procedures for verifying correct performance of operating activities" (Closed)</u>

Since the previous inspection (see Inspection Report 50-361/82-04), the inspector reviewed the following additional documentation pertaining to the open items in this area:

PK Operating Instruction SO23-0-24, "Redundant and Operability Testing Requirements," Revision 1, February 2, 1982

PK Operating Instruction SO23-0-13, "Work Authorizations," Revision 5, February 2, 1982

PK Special Procedure SO23-SPO-4.0, "Use of Not Turned Over Labels," Revision New, February 2, 1982

PØ Station Order SO123-0-114, "Equipment Control Program," Revision 0, January 27, 1982

Based on the above review and discussions with applicant personnel, the inspector determined that the open items addressed in Inspection Report 82-04, have been corrected.

All systems required for fuel loading have been turned over to the station and, the I.C.6 program is presently being implemented for these systems. In addition, the applicant has committed that prior to leaving each phase of the startup program, any additional systems required for the next phase will have been turned over to the station and the I.C.6 program implemented. This process will be reviewed during subsequent inspections. (OI 50-361/82-10-02)

The inspector will also review the implementation of SO23-SPO-4.0, "Use of Not Turned Over Labels" during subsequent inspections. (OI 50-361/82-10-03)

c. <u>TMI Item I.D.1, "Control Room Design" (Open)</u>

The inspector reviewed the following documentation relevent to the two items remaining open in this area prior to fuel loading:

PR Engineering Procedure SO23-V-4.21, "Plant Monitoring System Software Control and Documentation," Revision New, January 28, 1982

Phyllis Young 7-6-82 97

Operating Instruction S023-3-2.13, "Core Protection/Control Element Assembly Calculator Operation," Revision 2, January 28, 1982

Based on the above review and discussions with applicant personnel, the inspector determined the following:

(1) The Core Protection Calculator Index required by NUREG 0712, "Safety Evaluation Report SONGS 2 and 3," (the SER), Supplement 1, was provided in Operating Instruction S023-3-2.13, dated January 28, 1982.

(2) The administrative controls for software changes to the plant computer were provided by Engineering Procedure S023-V-4.21, dated January 28, 1982. With the completion of the above items, all pre-fuel load requirements of TMI Item I.D.1, as stated in Supplement 1 to the SER are complete.

d. TMI Item II.E.4.2, "Containment Isolation Dependability" (Closed)

The applicant approved and issued Operations Surveillance Test S023-3-3.26, "Once a Day Surveillance (Modes 1-4)," Revision 1, dated February 2, 1982. Supplement 1 to the SER requires a daily check of the closed position of the containment purge valves, and S023-3-3.26 provides for implementation of this requirement. This closes the remaining identified deficiency in this area.

e. TMI Item II.F.1, "Additional Accident Monitoring Instrumentation," Subitem 2F (Closed)

Subitem 2F, Containment Hydrogen Monitor

The inspector reviewed the following:

Safety Evaluation Report SONGS 2 and 3, NUREG 0712, Supplement 1

NUREG 0737, Clarification of TMI Action Plan Requirements

Design Change Package 50J

Instrument and Control Loop Verification Data Sheets

Based on the above review and discussions with applicant personnel, the inspector concluded that the containment H_2 monitor is complete. TMI Subitem II.F.1.2F, is closed.

No items of noncompliance or deviations were identified.

Phyllis Henry 7-6-82 - 9J6

4. IEB 80-06, "Engineered Safety Feature (ESF) Reset Controls" (Closed)

The inspector reviewed the following documentation in addition to that reviewed during the previous inspection (see Inspection Report 50-361/82-04):

IR

PR

Operating Instruction SO23-1-4.1, "Containment Emergency Cooling," Revision 3, February 7, 1982

Operating Instruction SO23-3-2.10, "Main Steam Isolation Valve Operation," Revision 4, February 7, 1982

Emergency Operating Instruction SO23-3-5.9, "Steam Line Rupture," Revision 5, February 7, 1982

Emergency Operating Instruction SO23-3-5.15, "Recovery From Inadvertent Safety Injection/Containment Isolation," Revision 5, February 7, 1982

Operating Instruction SO23-3-2.9, "Containment Spray/Iodine Removal System Operation," Revision 3, February 7, 1982

Emergency Operating Instruction SO23-3-5.6, "Loss of Coolant Accident," Revision 4, February 7, 1982

Emergency Operating Instruction SO23-3-5.29, "Steam Generator Tube Rupture," Revision 4, February 7, 1982

Based on the above review, discussions with applicant personnel, and discussions with the Office of Nuclear Reactor Regulation, the inspector determined the following:

(1) The applicant appears to have satisfactorily included appropriate caution statements in the applicable procedures, as discussed in Supplement 4 to the SER.

(2) The applicant has added an action statement to appropriate procedures to prevent the outlet valve on the Volume Control Tank from opening upon reset of a safety injection actuation signal.

(3) The applicant has not yet revised the applicable procedures as necessary to prevent re-energizing the non-Class 1E backup pressurizer heaters to a Class 1E bus upon safety injection actuation signal reset. Such revisions are required prior to exceeding 5 percent power by Supplement 5 of the SER. This item will be examined at a future inspection (OI 50-361/82-10-06).

This item is closed for issuance of a low power license.

No items of noncompliance or deviations were identified.

Phyll Harvey 7-6 12-

5. Safety Committee Activities (Open)

The inspector reviewed the following documents pertaining to this area:

PR

Engineering and Construction Interim QA Procedure 40-9-21, "Nuclear Safety Group Review and Audit Responsibilities for SONGS 2 and 3," Revision 1, February 1, 1982

PR Administrative Procedure SO123-III-1.0, "Station Documents - Preparation, Revision and Review," Revision 3, January 27, 1982

SP Station Order SO123-G-1, "Organization and Responsibilities of the Facility Staff and the On-Site Review Committee," Revision 0, January 27, 1982

SP Station Order SO123-A-109, "Station Documents," Revision 2, January 27, 1982

Based on the above review the inspector noted the followings

(1) The offsite review committee (Nuclear Safety Group) functions are delineated in Procedure E and C 40-9-21. This procedure appears to be in conformance with the presently proposed Technical Specifications.

(2) The implementation of the NSG functions will not be fully realized until after issuance of the operating license. This is due to the nature of the required reviews. For example, 50.59 reviews do not commence until after issuance of the license. Full implementation of the NSG function will be reviewed at a subsequent inspection. (OI 50-361/82-10-04)

(3) The Onsite Review Committee has as yet not had its first official meeting. The Procedure Review Committee, however, has been meeting several times a month to review procedures; and this committee is basically the forerunner to the Onsite Review Committee (OSRC). Full implementation of the OSRC activities will occur and be inspected after issuance of the operating license. (OI 50-361/82-10-05)

(4) The licensee has developed procedures to define the responsibilities and authority of the OSRC, and these procedures appear to conform to the provisions of the presently proposed Technical Specifications. The procedures however, have not yet been distributed for use, due to the large number of procedure changes occuring at this time. Distribution of the procedures will be confirmed at a future inspection. (OI-82-10-1)

No items of noncompliance or deviations were identified.

Phyll Henry 7-8 12-97

6. Plant Tour

The inspector made several plant tours and noted that the overall cleanliness of the containment had been improved in preparation for fuel loading.

No items of noncompliance or deviations were identified.

7. Preoperational Test Witnessing

The inspector observed selected portions of Preoperational Test 2PE-358-01, "Plant Protection System Time Response." During the performance of this test, the inspector verified that the test was conducted with an approved procedure, that test equipment was installed and functional, and that the test data were collected and recorded.

No items of noncompliance or deviations were identified.

8. Independent Inspection Effort

The inspector monitored an emergency plan implementation drill. This drill included manning the Technical Support Center (TSC) and Control Room in order to deal with an accident scenario which included a partial core melt down. It appeared to the inspector that, overall, the drill was beneficial and provided useful training. The inspector did note, however, that one copy of the proposed Technical Specifications present in the Technical Support Center was not the latest revision. The applicant assured the inspector that this deficiency would be corrected promptly. All other publications in the TSC appeared to be up-to-date. The inspector also discussed with applicant personnel their responsibility for maintaining a continuous open circuit with the NRC Operations Center during an event. While there was some initial confusion regarding this matter, it appears the applicant now has a clear understanding of his responsibilities in this area.

No items of noncompliance or deviations were identified.

9. Exit Interview

The inspector met with the applicant's representatives (denoted in Paragraph 1) at the conclusion of the inspection on February 12, 1982, and presented the results of the inspection.

ENCLOSURE 2

Enclosure 2:

This enclosure includes replacement San Onofre Units 2 and 3 Operating License (OL) pages to supercede those provided in the SCE to NRC amendment request letter dated March 21, 2001 (Proposed change Number [PCN]-517).

In addition to showing the PCN-517 OL changes these replacement OL pages identify those NRC license amendments which closed other license conditions not being proposed for closure in PCN-517.

Please replace PCN-517 Attachments A, B, C, D, E, and F with the following pages.

Attachment A

(Existing Facility Operating License)

SONGS Unit 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS AND ELECTRIC COMPANY
THE CITY OF RIVERSIDE, CALIFORNIA
THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2

FACILITY OPERATING LICENSE

License No NPF-10

Facility Operating License No. NPF-10 was issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in The Final Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess, use, and operate the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibrations, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2. Transshipment of Unit 1 fuel between Units 1 and 2 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 181, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Antitrust Conditions</u>

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) <u>Intentionally Deleted</u>

---

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

This paragraph intentionally deleted.

(6) High Burnup Fission Gas Release (Section 4.2.2.2. SER)

Prior to beginning the cycle of reactor operation during which peak fuel pellet burnups will achieve greater than 20,000 megawatt days per metric ton of uranium SCE shall provide revised analyses using fission gas release models acceptable to the NRC staff.

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Prior to operation for more than five (5) effective full power years, SCE shall provide at report describing its reexamination of the Technical Specification requirements for steam generator/RCS delta T and SDCS initiation temperature limits. The report must either demonstrated that the current Technical Specification limits and still suitably conservative, or propose and justify revised limits.

(8) Control Room Pressurization Capability (Section 6.4, SER, SSER #5)

By November 1, 1982, SCE shall complete the modifications required to achieve a positive pressure of 1/8" water gauge in the control room. Tests shall be performed on the modified system to verify the 1/8" positive pressure.

(9) Seismic Trip System (Section 7.2.5, SSER #4)

Prior to initially exceeding five (5) percent power, the seismic trip system shall be operable.

(10) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Prior to startup following the first refueling outage, the volume control tank outlet valve control logic shall be modified to ensure that the valve does not change position following safety injection actuation signal reset.

(11) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

By May 15, 1982, SCE shall submit a proposal, including a proposed implementation schedule, for meeting Revision 2 of Regulatory Guide 1.97.

(12) Control System Failures (Section 7.7, SSER #4)

a. By April 1, 1983, SCE shall provide an evaluation of control system failures caused by high energy line break, and by failures of any power sources, sensor, or sensor impulse lines which provide power or signals to two or more control systems. Implementation of any corrective action resulting from this evaluation shall be completed on a schedule acceptable to the NRC.

(13) Diesel Generator Modifications (Section 8.3.1, SER)

Prior to startup following the first refueling outage. SCE shall install a heavy duty turbocharger gear drive assembly on the emergency diesel generators.

(14) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(15) Turbine Disc Inspection (Section 10.2.2, SER)

Prior to startup following the second refueling outage, the bores of the low pressure turbine disc shall be inspected for ultrasonic indications.

(16) Radioactive Waste System (Section 11.1, SER, SSER #5)

"Wet" solid radwaste shall not be shipped from the facility until the NRC has approved the waste solidification Process Control Program.

(17) Purge System Monitors (Section 11.3, SER, SSER #5)

Prior to startup following the first refueling outage, equipment having the capability to continuously monitor and sample the containment purge exhaust directly from the purge stack shall be operable.

(18) Initial Test Program (Section 14, SER)

SCE shall conduct the post-fuel loading initial test program (set forth in Section 14 of the San Onofre Units 2 and 3 Final Safety Analysis Report, as amended) without making any major modifications to this program unless such modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.

b. Modification of test objectives, methods, or acceptance criteria for any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.

c. Performance of any test at a power level different than that described in the test procedure.

d. Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).

(19) NUREG-0737 Conditions (Section 22)

Each of the following conditions shall be completed to the satisfaction of the NRC. Each item references the related subpart of Section 22 of the SER and/or its supplements.

a. Shift Technical Advisor (I.A.1.1, SSER #1)

SCE shall provide a fully trained on-shift technical advisor to the shift supervisor (watch engineer).

b. Shift Manning (I.A.1.3, SSER #1, SSER #5)

Deleted.

c. Independent Safety Engineering Group (1.B.1.2, SSER #1)

SCE shall have an on-site independent safety engineering group.

d. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

By May 1, 1982, SCE shall provide emergency procedure guidelines. Emergency procedures based on guidelines approved by the NRC shall be implemented prior to startup following the first refueling outage.

e. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Prior to fuel loading, SCE shall implement a system for verifying the correct performance of operating activities, and shall keep the system in effect thereafter.

f. Control Room Design Review (I.D.1, SSER #1)

Prior to exceeding five (5) percent power, SCE shall:

1. Prioritize the control room annunciator windows.

2. Delete master acknowledge capabilities of the annunciator system.

3. Incorporate a second flash note/audible scheme into the annunciator system to alert the operator of an alarm returned to normal.

4. Identify changes required to correct control room lighting for optimum operator performance.

5. Revise control room labeling according to a hierarchical scheme.

6. Label Foxboro containment spray controller.

7. Replace RC loop hot leg temperature scales with appropriate scale divisions.

8. Eliminate 10X multiplier from RC loop hot leg and cold leg temperature.

9. Make all labels flush with the face of the instrument bezel.

10. Incorporate normal and abnormal operating range indications on applicable instruments.

11. Replace Dymo tape with permanent labels or markers.

12. Color code all component bezels.

13. Add channel identification to emergency feedwater controls.

14. Label dual function vertical scales to identify each scale.

15. Provide increase/decrease labels for the containment spray chemical controllers.

16. Incorporate the requirement to replace burned-out lamps in the procedures.

17. Add phone jacks to the control room back-panel areas.

Prior to startup following the first refueling outage, SCE shall complete the changes required to correct control room lighting for optimum operator performance.

g. Special Low Power Testing and Training (I.G.1, SSER #1)

By April 16, 1982, SCE shall provide detailed test procedures and a safety analysis.

h. Reactor Coolant System Vents (II.B .1), SSER #1 , SSER #4)

By May 1, 1982, SCE shall provide procedures or procedure guidelines for reactor coolant gas vent system operation and testing.

i. Deleted

j. Safety Valve Test Requirements (II.D.1, SSER #1)

SCE shall conform to the results of the EPRI test program. By April 1, 1982, SCE shall provide confirmation of the adequacy of the San Onofre 2 RCS safety valves based on a preliminary review of generic test program results. By July 1, 1982, SCE shall provide evidence supported by test of safety valve functionality for expected operating and accident (non-ATWS) conditions. The testing shall demonstrate that the valves will open and reclose under the expected flow conditions. By July 1, 1982, SCE shall provide an evaluation of the adequacy of the associated piping and supports at San Onofre 2.

k. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

Prior to exceeding five (5) percent power, the safety valve position indication system shall be environmentally and seismically qualified consistent with the component or system to which it is attached, and documentation of this shall be provided.

l. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #1)

Prior to exceeding five (5) percent power, SCE shall conduct a 48-hour endurance test of all auxiliary feedwater pumps.

m. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

Prior to exceeding five (5) percent power, SCE shall implement procedures to preclude the automatic reapplication of pressurizer heaters to Class IE buses upon SIAS reset.

n. Additional Monitoring Instrumentation (II.F.1, SSER #1, SSER #4)

Prior to exceeding five (5) percent power, the mid/high range noble gas monitors and iodine and particulate isokinetic samplers shall be operable.

o. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

Prior to startup following the first refueling outage, the following items shall be completed:

1. The subcooling monitors shall be modified to include the maximum unheated junction thermocouple temperature and the representative core exit thermocouple input.

2. Incore detector assemblies (core exit thermocouples and associated cabling) shall be environmentally qualified and shall have seismic and environmentally qualified Class 1E connectors.

3. Qualified cables shall be installed for the core exit thermocouples.

4. A safety parameter display system shall be provided.

5. The heated junction thermocouple probe and associated process instrumentation shall be installed.

p. Voiding in the Reactor Coolant System (II.K.2.17, SSER #1, SSER #5)

By May 1, 1982, SCE shall provide the results of the Combustion Engineering Owners Group analysis of the potential for RCS voiding during anticipated transients.

q. Revised Model for Small-Break LOCAs (II.K.3.30, SSER #1, SSER #4, SSER #5

By May 1, 1982, SCE shall provide the results of the Combustion Engineering Owners Group effort on model justification or a revised analytical model.

r. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31), SSER #1)

Within one year after model revisions are approved by the NRC, SCE shall provide a supplemental plant-specific analysis to verify compliance with 10 CFR 50.46, using the revised models developed under item II.K.3.30.

s. Improving Licensee Emergency Preparedness (III.A.2, SSER#1, SSER #5)

1. By April 1, 1982, SCE shall provide a functional description of the upgraded emergency support facilities. (Technical Support Center, Operations Support Center and Emergency Operations Facility).

2. By January 1, 1983, the upgraded emergency support facilities shall be operational.

3. SCE shall maintain interim emergency support facilities (Technical Support Center, Operations Support Center and the Emergency Operations Facility) until the upgraded facilities are completed.

(20) Surveillance Program (Section 1.12, SSER #5)

Prior to entering any operational mode for the first time, including initial fuel loading, SCE shall:

a. Have completed a review of the surveillance procedures applicable to the change of mode, and determined that the procedures demonstrate the operability of the required systems With respect to all acceptance criteria defined in the Technical Specifications.

b. Have dispatched written certification to the NRC Regional Administrator, Region V, that the actions defined in a, above, have been completed for the mode or modes to be entered.

(21) <u>Laboratory Instrumentation (Section 1.12, SSER #5)</u>

Prior to initial entry into operating Mode 2, the laboratory instrumentation described in Sections 11.5.2.2.2 and 12.5.2.2.1 of the Final Safety Analysis Report shall be calibrated and shall be capable of analyzing sample types and geometries necessary to support facility operation. In addition, at that time there shall also be approved, written procedures governing laboratory operations and analyses.

(22) <u>Design Verification Program (Section 3.7.4, SSER #5)</u>

Prior to exceeding five (5) percent power, SCE shall provide the final report of the Design Verification Program being conducted by the General Atomic Company, and NRC approval of the results must be obtained.

(23) <u>Emergency Preparedness Conditions</u>

a. Conditions of ASLB Initial Decision of May 14, 1982

Within five (5) months of initially exceeding five (5) percent power, SCE shall:

i. Demonstrate that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (See Initial Decision, Section IV, Paragraph D1.12, pp. 136-140), at no less than that level of readiness, pending development of satisfactory capability of offsite response organizations (see Initial Decision, Section IV, Paragraph D.27, pp. 145-146, and Section V, Paragraph B, pp. 213-214).

ii. Provide an assessment of whether public information regarding emergency planning should also be presented in Spanish (see Initial Decision, Section IV, Paragraph F.32, pp. 168, and Section V, Paragraph C.2, pp. 215).

iii. Provide revised plans demonstrating that the "extended" Emergency Planning Zone (EPZ) concept has been deleted from the San Onofre onsite and offsite plans and the Plume Exposure Pathway EFZ boundary has been extended, along with siren coverage, to Dana Point and all of San Juan Capistrano (see Initial Decision, Section IV, Paragraph D.25, pp. 98, and Section V, Paragraph C.5, pp. 216; see also Order (making Clarifying Change in Initial Decision) dated May 25, 1982).

b. Completion of Emergency Preparedness Requirements

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's proposed rules, 44 CFR 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR 50.54(s) (2) will apply.

c. Condition of ASLB of February 1, 1983 (Medical Services)

By September 17, 1983, or six months from the date that the Nuclear Regulatory Commission issues its determination of the medical services questions certified by it, whichever is the shorter period of time, SCE shall demonstrate that SCE and offsite jurisdictions have developed and stand ready to implement arrangements for medical services as required by 10 CFR 50.47(b) (12) (See Initial Decision, Section III, pp. 43-47, and Section V Paragraph D. pp. 216-217, and Stipulation and Order Modifying License Condition, February 1, 1983).

d. Conditions of ALAB-717, March 4, 1983

i. By July 2, 1983, SCE shall provide evidence that it has undertaken further efforts to assemble and to keep current as reasonably complete a list as possible of housebound people within the plume emergency planning zone who would require transportation assistance in the event of an evacuation.

ii. By July 2, 1983, SCE shall provide evidence that a training program has been developed and initiated to assist Orange County Transit District bus drivers in the discharge of their responsibilities in the event of a radiological emergency at San Onofre.

(24) RCS Depressurization System (PORV's)

By June 30, 1983, SCE shall provide a complete response to the NRC letter of March 27, 1982, requesting additional information relative to the capability of San Onofre 2 and 3 for rapid depressurization and decay heat removal without power operating relief valves (PORV's).

(25) Qualification of Auxiliary Feedwater Pump Motor Bearings

By October 30, 1982, SCE shall submit a proposed hardware modification and schedule for implementation that will increase the reliability of the AFW motor-driven pumps in the event of a break in the high energy line feeding the steam-driven pump. In the interim, prior to the installation of a hardware modification acceptable to the NRC staff, SCE shall perform an augmented in-service inspection of the steam line in accordance with SCE's letter of July 12, 1982.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which may contain Safeguards Information protected under 10 CFR 73.21, are entitled: "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Physical Security Plan," with revisions submitted through April 22, 1988; "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Security Force Training and Qualification Plan," with revisions submitted through October 22, 1986; and "San Onofre Nuclear Generating Station, Units 1, 2, and 3, Safeguards Contingency Plan," with revisions submitted through December 29, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. SCE shall report any violations of the requirements contained in Section 2, items C(1), C(3) through C(13), C(15) through C(22), and F of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region IV, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on February 16, 2022.

Attachment B

(Existing Facility Operating License)

SONGS Unit 3

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

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS & ELECTRIC COMPANY
THE CITY OF RIVERSIDE, CALIFORNIA
THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

FACILITY OPERATING LICENSE

License No. NPF-15

Facility Operating License No. NPF-15 was issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report, as amended, through Amendment 30, and the Environmental Report, as amended, through Amendment 6.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3. Transshipment of Unit 1 fuel between Units 1 and 3 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 172, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Intentionally Deleted

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

This paragraph intentionally deleted.

The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(6) High Burnup Fission Gas Release (Section 4.2.2.2, SER)

Prior to beginning the cycle of reactor operation during which peak fuel pellet burnups will achieve greater than 200,000 megawatt days per metric ton of uranium, SCE shall provide revised analyses using fission gas release models acceptable to the NRC staff.

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Prior to operation for more than five (5) effective full power years, SCE shall provide a report describing its reexamination of the Technical Specification requirements for steam generator/reactor coolant system delta temperature and shutdown cooling system initiation temperature limits that are presently provided for overpressure protection. The report must either demonstrate that the current Technical Specification limits are still suitably conservative, or propose and justify revised limits.

(8) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Prior to startup following the first refueling outage, the volume control tank outlet valve control logic shall be modified to ensure that the valve does not change position following safety injection actuation signal reset. In the interim, SCE shall maintain emergency procedures that require the volume control tank outlet valve to be placed in the manual mode prior to SIAS reset.

(9) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Prior to startup following the first refueling outage, SCE shall comply with the recommendations of Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," as described in the SCE letter of May 13, 1982.

(10) Control System Failures (Section 7.7, SER, SSER #4)

By April 1, 1983, SCE shall provide an evaluation, for NRC staff review and approval, of control system failures caused by high energy line break, and by failures of any power sources, sensors, or sensor impulse lines which provide power or signals to two or more control systems. Implementation of any corrective action resulting from this evaluation shall be completed on a schedule acceptable to the NRC.

(11) Diesel Generator Modifications (Section 8.3.1, SER)

Prior to startup following the first refueling outage. SCE shall install a heavy duty turbocharger gear drive assembly on the emergency diesel generators.

(12) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(13) Turbine Disc Inspection (Section 10.2.2, SER)

Prior to startup following the second refueling outage, the bores of the low pressure turbine disc shall be inspected for flaws using ultrasonic testing. The results of the inspection shall be submitted to the NRC staff.

(14) Radioactive Waste System (Section 11.1, SER, SSER #5)

"Wet" solid radwaste shall not be shipped from the facility until the NRC has approved the waste solidification Process Control Program.

(15) Purge System Monitors (Section 11.3, SER, SSER #5)

Prior to startup following the first refueling outage, equipment having the capability to continuously monitor and sample the containment purge exhaust directly from the purge stack shall be operable.

(16) Initial Test Program (Section 14, SER)

SCE shall conduct the post-fuel loading initial test program (set forth in Section 14 of the San Onofre Units 2 and 3 Final Safety Analysis Report, as amended, through Amendment 30) without making any major modifications to this program unless such modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.

b. Modification of test objectives, methods, or acceptance criteria for any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.

c. Performance of any test at a power level different than that described in the test procedure.

d. Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).

(17) NUREG-0737 Conditions (Section 22)

Each of the following conditions shall be completed to the satisfaction of the NRC. Each item references the related subpart of Section 22 of the SER and/or its supplements.

a. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Emergency procedures based on guidelines approved by the NRC shall be implemented prior to startup following the first refueling outage that occurs six months or more after NRC approval of the guidelines.

b. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Prior to fuel loading, SCE shall implement a system for verifying the correct performance of operating activities, and shall keep the System in effect thereafter.

c. Control Room Design Review (I.D.1, SSER #1)

The control room modifications identified as required in Section 22, Item I.D.1 of Supplement No. 1 to the SER shall be installed and made operational on the schedules identified for each modification in Supplement No. 1 to the SER.

d. Deleted

e. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

The safety valve position indication system shall be environmentally and seismically qualified consistent with the component or system to which it is attached, and documentation of this shall be maintained.

f. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #11)

Prior to exceeding five (5) percent power, SCE shall conduct a 48-hour endurance test of all auxiliary feedwater pumps. The results of the test shall be submitted to the NRC staff.

g. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

SCE shall maintain in effect procedures to preclude the automatic reapplication of pressurizer heaters to Class IE buses upon SIAS reset.

h. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

Prior to fuel loading, the following items shall be completed, and shall be maintained thereafter:

1. The subcooling monitors shall be modified to include the maximum unheated junction thermocouple temperature and the representative core exit thermocouple input.

2. Incore detector assemblies (core exit thermocouples and associated cabling) shall be environmentally qualified and shall have seismic and environmentally qualified Class 1E connectors.

3. Qualified cables shall be installed for the core exit thermocouples.

4. The heated junction thermocouple probe and associated process instrumentation shall be installed.

Prior to startup following the first refueling outage, the heated junction thermocouple system and the safety parameter display system shall be operable and shall be maintained operable thereafter.

i. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31, SSER #1

Within one year after model revisions are approved by the NRC, SCE shall provide a supplemental plant-specific analysis to verity compliance with 10 CFR 50.46, using the revised models developed under Item II.K.3.30.

j. Improving Licensee Emergency Preparedness (III.A.2, SSER #1, SSER #5

1. By January 1, 1983, the upgraded emergency support facilities shall be operational.

2. SCE shall maintain interim emergency support facilities (Technical Support Center, Operations Support Center and the Emergency Operations Facility) until the upgraded facilities are completed.

(18) Emergency Preparedness Conditions

a. Conditions of ASLB Initial Decision of May 14, 1982

By February 17, 1983, SCE shall:

1. Provide evidence that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (see Initial Decision, Section IV, Paragraph D.1-12, pp. 136-140), at no less than that level of readiness, pending development of satisfactory capability of offsite response Paragraph D.27, pp. 145-146, Section V, Paragraph B, pp. 213-214).

2. Provide an assessment of whether public information regarding emergency planning should also be presented in Spanish (see Initial Decision, Section IV, Paragraph F.32, pp. 168, and Section V, Paragraph C.2, pp. 215).

3. Provide revised plans demonstrating that the "extended" Emergency Planning Zone (EPZ) concept has been deleted from the San Onofre onsite and offsite plans and the Plume Exposure Pathway EPZ boundary has been extended, along with siren coverage, to Dana Point and all of San Juan Capistrano (see Initial Decision, Section IV, Paragraph D.25, pp. 98, and Section V, Paragraph C.5, pp. 216; see also Order (Making Clarifying Change in Initial Decision) dated May 25, 1982).

b. Completion of Emergency Preparedness Requirements

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's proposed rules, 44 CFR 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR 50.54(s) (2) will apply.

c. Deleted Amendment No. 8 dated September 16, 1983

d. Conditions of ALAB 717, March 4, 1983

i. By July 2, 1983, SCE shall provide evidence that it has undertaken further efforts to assemble and to keep current as reasonably complete a list as possible of housebound people within the plume emergency planning zone who would require transportation assistance in the event of an evacuation.

ii. By July 2, 1983, SCE shall provide evidence that a training program has been developed and initiated to assist Orange County Transit District bus drivers in the discharge of their responsibilities in the event of a radiological emergency at San Onofre.

(19) RCS Depressurization System (PORV's)

By June 30, 1983, SCE shall provide a complete response to the NRC letter of March 27, 1982, including information relative to the capability of San Onofre 3 for rapid depressurization and decay heat removal without power operated relief valves (PORVs).

(20) Qualification of Auxiliary Feedwater (AFW) Pump Motor Bearings

Prior to startup following the first refueling outage, SCE shall install and make operational the lubrication oil cooling system for the auxiliary feedwater pump motor bearings described in SCE's letter of March 7, 1983. Prior to installation of the lube oil cooling system, SCE shall perform daily visual inspection of the steam lines in the AFW pump room in accordance with SCE's letter of July 12, 1982.

(21) Surveillance Program (Section 1.12, SSER #5)

Prior to entering any operational mode for the first time, including initial fuel loading, SCE shall:

a. Have completed a review of the surveillance procedures applicable to the change of mode, and determined that the procedures demonstrate the operability of the required systems with respect to all acceptance criteria defined in the Technical Specifications.

b. Have dispatched written certification to the NRC Regional Administrator, Region V, that the actions defined in (a), above, have been completed for the mode or modes to be entered.

(22) Auxiliary Building Ventilation System.

SCE shall complete all modifications to the auxiliary building ventilation system described in the November 5, 1982 letter from H. Ray, SCE, to R. Engelken, NRC, on the schedule proposed In the November 5, 1982 letter.

(23) Fuel Assembly Shoulder Gap Clearance (SCE letter of July 25, 1983)

Prior to entering Startup (Mode 2) after each refueling, SCE shall either provide a report that demonstrates that the existing fuel element assembly (FEA) has sufficient available shoulder gap clearance for at least the next cycle of operation, or identify to the NRC and implement a modified FEA design that has adequate shoulder gap clearance for at least the next cycle of operation. The commitment will apply until the NRC concurs that the shoulder gap clearance provided is adequate for the design life of the fuel.

(24) Isolation Capability for Primary EOF

By January 1, 1984 the primary EOF ventilation system shall be modified to provide isolation capability as described in the SCE letter of July 22, 1983.

(25) Correction of CPC Software Error

At the first outage of sufficient duration (7 days in Mode 5) after February 2, 1984, SCE shall correct the software error in the Core Protection Calculators discussed in the SCE letters dated March 7, 1893 and July 22, 1983.

(26) Until the first refueling outage, SCE shall provide a monthly report describing any occurrences resulting in the degradation (including, but not limited to component failures, maintenance errors, and operator errors) of the auxiliary feedwater system. The report shall identify the cause of such occurrences. The report does not relieve the licensee from any existing requirements for Licensee Event Reports (LERs).

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which may contain Safeguards Information protected under 10 CFR 73.21, are entitled: "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Physical Security Plan," with revisions submitted through April 22, 1988; "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Security Force Training and Qualification Plan," with revisions submitted through October 22, 1986; and "San Onofre Nuclear Generating Station, Units 1, 2, and 3, Safeguards Contingency Plan," with revisions submitted through December 29, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

G. SCE shall report any violations of the requirements contained in Section 2, items C(1), C(3) through C(11), C(13) through C(22), and F of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region IV, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on November 15, 2022.

K. In accordance with the Commission's direction in its Statement of Policy, Licensing and Regulatory Policy and Procedures for Environmental Protection; Uranium Fuel Cycle Impacts, October 29, 1982, this license is subject to the final resolution of the pending litigation involving Table S-3. See Natural Resources Defense Council v. NRC, No. 74-1586 (D. C. Cir., April 27, 1982).

ATTACHMENT 1
TO
NPF-15

The following item must be completed prior to initial criticality:

> The deficiency identified by the SCE letter, dated July 19, 1982, to R.H. Engelken from Dr. L.T. Papay regarding discrepant inputs to the Core Protection Calculator from Reactor Coolant Pump shaft speed and Control Element Assembly position indication shall be corrected.

Attachment C

(Proposed Amendments to Facility Operating License)
(Redline and Strikeout)

SONGS Unit 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS AND ELECTRIC COMPANY
THE CITY OF RIVERSIDE, CALIFORNIA
THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2

FACILITY OPERATING LICENSE

License No NPF-10

Facility Operating License No. NPF-10 was issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in The Final Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess, use, and operate the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibrations, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2. Transshipment of Unit 1 fuel between Units 1 and 2 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 181, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

~~(4) Intentionally Deleted~~

(4) Containment Tendon Surveillance

Deleted by Amendment No. 37

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

Deleted by Amendment No. 60

~~This paragraph intentionally deleted.~~

(6) High Burnup Fission Gas Release (Section 4.2.2.2. SER)

Deleted by Amendment No.

~~Prior to beginning the cycle of reactor operation during which peak fuel pellet burnups will achieve greater than 20,000 megawatt days per metric ton of uranium SCE shall provide revised analyses using fission gas release models acceptable to the NRC staff.~~

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Deleted by Amendment No.

~~Prior to operation for more than five (5) effective full power years, SCE shall provide at report describing its reexamination of the Technical Specification requirements for steam generator/RCS delta T and SDCS initiation temperature limits. The report must either demonstrated that the current Technical Specification limits and still suitably conservative, or propose and justify revised limits.~~

(8) Control Room Pressurization Capability (Section 6.4, SER, SSER #5)

Deleted by Amendment No.

~~By November 1, 1982, SCE shall complete the modifications required to achieve a positive pressure of 1/8" water gauge in the control room. Tests shall be performed on the modified system to verify the 1/8" positive pressure.~~

(9) Seismic Trip System (Section 7.2.5, SSER #4)

Deleted by Amendment No.

~~Prior to initially exceeding five (5) percent power, the seismic trip system shall be operable.~~

(10) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, the volume control tank outlet valve control logic shall be modified to ensure that the valve does not change position following safety injection actuation signal reset.~~

(11) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Deleted by Amendment No.

~~By May 15, 1982, SCE shall submit a proposal, including a proposed implementation schedule, for meeting Revision 2 of Regulatory Guide 1.97.~~

(12) Control System Failures (Section 7.7, SSER #4)

Deleted by Amendment No.

~~a. By April 1, 1983, SCE shall provide an evaluation of control system failures caused by high energy line break, and by failures of any power sources, sensor, or sensor impulse lines which provide power or signals to two or more control systems. Implementation of any corrective action resulting from this evaluation shall be completed on a schedule acceptable to the NRC.~~

(13) Diesel Generator Modifications (Section 8.3.1, SER)

Deleted by Amendment No.

~~Prior to startup following the first refueling outage. SCE shall install a heavy duty turbocharger gear drive assembly on the emergency diesel generators.~~

(14) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(15) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No.

~~Prior to startup following the second refueling outage, the bores of the low pressure turbine disc shall be inspected for ultrasonic indications.~~

(16) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No.

~~"Wet" solid radwaste shall not be shipped from the facility until the NRC has approved the waste solidification Process Control Program.~~

(17) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, equipment having the capability to continuously monitor and sample the containment purge exhaust directly from the purge stack shall be operable.~~

(18) Initial Test Program (Section 14, SER)

Deleted by Amendment No.

~~SCE shall conduct the post-fuel loading initial test program (set forth in Section 14 of the San Onofre Units 2 and 3 Final Safety Analysis Report, as amended) without making any major modifications to this program unless such modifications have been identified and have received prior NRC approval. Major modifications are defined as:~~

~~a. Elimination of any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.~~

b. ~~Modification of test objectives, methods, or acceptance criteria for any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.~~

c. ~~Performance of any test at a power level different than that described in the test procedure.~~

d. ~~Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).~~

(19) NUREG-0737 Conditions (Section 22)

~~Each of the following conditions shall be completed to the satisfaction of the NRC. Each item references the related subpart of Section 22 of the SER and/or its supplements.~~

a. Shift Technical Advisor (I.A.1.1, SSER #1)

Deleted by Amendment No.

~~SCE shall provide a fully trained on-shift technical advisor to the shift supervisor (watch engineer).~~

b. Shift Manning (I.A.1.3. SSER #1, SSER #5)

Deleted by Amendment No. 147

~~Deleted.~~

c. Independent Safety Engineering Group (1.B.1.2, SSER #1)

Deleted by Amendment No.

~~SCE shall have an on-site independent safety engineering group.~~

d. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No.

~~By May 1, 1982, SCE shall provide emergency procedure guidelines. Emergency procedures based on guidelines approved by the NRC shall be implemented prior to startup following the first refueling outage.~~

e. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Deleted by Amendment No.

~~Prior to fuel loading, SCE shall implement a system for verifying the correct performance of operating activities, and shall keep the system in effect thereafter.~~

f. Control Room Design Review (I.D.1, SSER #1)

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, SCE shall:~~

~~1. Prioritize the control room annunciator windows.~~

~~2. Delete master acknowledge capabilities of the annunciator system.~~

~~3. Incorporate a second flash note/audible scheme into the annunciator system to alert the operator of an alarm returned to normal.~~

~~4. Identify changes required to correct control room lighting for optimum operator performance.~~

~~5. Revise control room labeling according to a hierarchical scheme.~~

~~6. Label Foxboro containment spray controller.~~

~~7. Replace RC loop hot leg temperature scales with appropriate scale divisions.~~

~~8. Eliminate 10X multiplier from RC loop hot leg and cold leg temperature.~~

~~9. Make all labels flush with the face of the instrument bezel.~~

~~10. Incorporate normal and abnormal operating range indications on applicable instruments.~~

~~11. Replace Dymo tape with permanent labels or markers.~~

~~12. Color code all component bezels.~~

~~13. Add channel identification to emergency feedwater controls.~~

~~14. Label dual function vertical scales to identify each scale.~~

~~15. Provide increase/decrease labels for the containment spray chemical controllers.~~

~~16. Incorporate the requirement to replace burned-out lamps in the procedures.~~

~~17. Add phone jacks to the control room back-panel areas.~~

~~Prior to startup following the first refueling outage, SCE shall complete the changes required to correct control room lighting for optimum operator performance.~~

g. Special Low Power Testing and Training (I.G.1, SSER #1)

Deleted by Amendment No.

~~By April 16, 1982, SCE shall provide detailed test procedures and a safety analysis.~~

h. Reactor Coolant System Vents (II.B .1), SSER #1 , SSER #4)

Deleted by Amendment No.

~~By May 1, 1982, SCE shall provide procedures or procedure guidelines for reactor coolant gas vent system operation and testing.~~

i. ~~Deleted~~
Post-Accident Sampling System (NUREG-0737 Item II.B.3)

Deleted by Amendment No. 178

j. <u>Safety Valve Test Requirements (II.D.1, SSER #1)</u>

Deleted by Amendment No.

~~SCE shall conform to the results of the EPRI test program. By April 1, 1982, SCE shall provide confirmation of the adequacy of the San Onofre 2 RCS safety valves based on a preliminary review of generic test program results. By July 1, 1982, SCE shall provide evidence supported by test of safety valve functionality for expected operating and accident (non-ATWS) conditions. The testing shall demonstrate that the valves will open and reclose under the expected flow conditions. By July 1, 1982, SCE shall provide an evaluation of the adequacy of the associated piping and supports at San Onofre 2.~~

k. <u>Direct Indication of Safety Valve Position (II.D.3, SSER #1)</u>

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, the safety valve position indication system shall be environmentally and seismically qualified consistent with the component or system to which it is attached, and documentation of this shall be provided.~~

l. <u>AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #1)</u>

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, SCE shall conduct a 48-hour endurance test of all auxiliary feedwater pumps.~~

m. <u>Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)</u>

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, SCE shall implement procedures to preclude the automatic reapplication of pressurizer heaters to Class IE buses upon SIAS reset.~~

n. <u>Additional Monitoring Instrumentation (II.F.1, SSER #1, SSER #4)</u>

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, the mid/high range noble gas monitors and iodine and particulate isokinetic samplers shall be operable.~~

o. <u>ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)</u>

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, the following items shall be completed:~~

~~1. The subcooling monitors shall be modified to include the maximum unheated junction thermocouple temperature and the representative core exit thermocouple input.~~

~~2. Incore detector assemblies (core exit thermocouples and associated cabling) shall be environmentally qualified and shall have seismic and environmentally qualified Class IE connectors.~~

~~3. Qualified cables shall be installed for the core exit thermocouples.~~

~~4. A safety parameter display system shall be provided.~~

~~5. The heated junction thermocouple probe and associated process instrumentation shall be installed.~~

p. Voiding in the Reactor Coolant System (II.K.2.17, SSER #1, SSER #5)

Deleted by Amendment No.

~~By May 1, 1982, SCE shall provide the results of the Combustion Engineering Owners Group analysis of the potential for RCS voiding during anticipated transients.~~

q. Revised Model for Small-Break LOCAs (II.K.3.30, SSER #1, SSER #4, SSER #5

Deleted by Amendment No.

~~By May 1, 1982, SCE shall provide the results of the Combustion Engineering Owners Group effort on model justification or a revised analytical model.~~

r. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31), SSER #1)

Deleted by Amendment No.

~~Within one year after model revisions are approved by the NRC, SCE shall provide a supplemental plant-specific analysis to verify compliance with 10 CFR 50.46, using the revised models developed under item II.K.3.30.~~

s. Improving Licensee Emergency Preparedness (III.A.2, SSER#1, SSER #5)

Deleted by Amendment No.

~~1. By April 1, 1982, SCE shall provide a functional description of the upgraded emergency support facilities. (Technical Support Center, Operations Support Center and Emergency Operations Facility).~~

~~2. By January 1, 1983, the upgraded emergency support facilities shall be operational.~~

~~3. SCE shall maintain interim emergency support facilities (Technical Support Center, Operations Support Center and the Emergency Operations Facility) until the upgraded facilities are completed.~~

(20) Surveillance Program (Section 1.12, SSER #5)

Deleted by Amendment No.

~~Prior to entering any operational mode for the first time, including initial fuel loading, SCE shall:~~

~~a. Have completed a review of the surveillance procedures applicable to the change of mode, and determined that the procedures demonstrate the operability of the required systems with respect to all acceptance criteria defined in the Technical Specifications.~~

~~b. Have dispatched written certification to the NRC Regional Administrator, Region V, that the actions defined in a, above, have been completed for the mode or modes to be entered.~~

(21) Laboratory Instrumentation (Section 1.12, SSER #5)

Deleted by Amendment No.

~~Prior to initial entry into operating Mode 2, the laboratory instrumentation described in Sections 11.5.2.2.2 and 12.5.2.2.1 of the Final Safety Analysis Report shall be calibrated and shall be capable of analyzing sample types and geometries necessary to support facility operation. In addition, at that time there shall also be approved, written procedures governing laboratory operations and analyses.~~

(22) Design Verification Program (Section 3.7.4, SSER #5)

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, SCE shall provide the final report of the Design Verification Program being conducted by the General Atomic Company, and NRC approval of the results must be obtained.~~

(23) Emergency Preparedness Conditions

Deleted by Amendment No.

~~a. Conditions of ASLB Initial Decision of May 14, 1982~~

~~Within five (5) months of initially exceeding five (5) percent power, SCE shall:~~

~~i. Demonstrate that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (See Initial Decision, Section IV, Paragraph D1.12, pp. 136-140), at no less than that level of readiness, pending development of satisfactory capability of offsite response organizations (see Initial Decision, Section IV, Paragraph D.27, pp. 145-146, and Section V, Paragraph B, pp. 213-214).~~

~~ii. Provide an assessment of whether public information regarding emergency planning should also be presented in Spanish (see Initial Decision, Section IV, Paragraph F.32, pp. 168, and Section V, Paragraph C.2, pp. 215).~~

~~iii. Provide revised plans demonstrating that the "extended" Emergency Planning Zone (EPZ) concept has been deleted from the San Onofre onsite and offsite plans and the Plume Exposure Pathway EFZ boundary has been extended, along with siren coverage, to Dana Point and all of San Juan Capistrano (see Initial Decision, Section IV, Paragraph D.25, pp. 98, and Section V, Paragraph C.5, pp. 216; see also Order (making Clarifying Change in Initial Decision) dated May 25, 1982).~~

~~b. Completion of Emergency Preparedness Requirements~~

~~In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's proposed rules, 44 CFR 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR 50.54(s) (2) will apply.~~

~~c. Condition of ASLB of February 1, 1983 (Medical Services)~~

~~By September 17, 1983, or six months from the date that the Nuclear Regulatory Commission issues its determination of the medical services questions certified by it, whichever is the shorter period of time, SCE shall demonstrate that SCE and offsite jurisdictions have developed and stand ready to implement arrangements for medical services as required by 10 CFR 50.47(b) (12) (See Initial Decision, Section III, pp. 43-47, and Section V Paragraph D. pp. 216-217, and Stipulation and Order Modifying License Condition, February 1, 1983).~~

~~d. Conditions of ALAB-717, March 4, 1983~~

~~i. By July 2, 1983, SCE shall provide evidence that it has undertaken further efforts to assemble and to keep current as reasonably complete a list as possible of housebound people within the plume emergency planning zone who would require transportation assistance in the event of an evacuation.~~

~~ii. By July 2, 1983, SCE shall provide evidence that a training program has been developed and initiated to assist Orange County Transit District bus drivers in the discharge of their responsibilities in the event of a radiological emergency at San Onofre.~~

(24) RCS Depressurization System (PORV's)

Deleted by Amendment No.

~~By June 30, 1983, SCE shall provide a complete response to the NRC letter of March 27, 1982, requesting additional information relative to the capability of San Onofre 2 and 3 for rapid depressurization and decay heat removal without power operating relief valves (PORV's).~~

(25) Qualification of Auxiliary Feedwater Pump Motor Bearings

Deleted by Amendment No.

~~By October 30, 1982, SCE shall submit a proposed hardware modification and schedule for implementation that will increase the reliability of the AFW motor-driven pumps in the event of a break in the high energy line feeding the steam-driven pump. In the interim, prior to the installation of a hardware modification acceptable to the NRC staff, SCE shall perform an augmented in-service inspection of the steam line in accordance with SCE's letter of July 12, 1982.~~

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which may contain Safeguards Information protected under 10 CFR 73.21, are entitled: "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Physical Security Plan," with revisions submitted through April 22, 1988; "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Security Force Training and Qualification Plan," with revisions submitted through October 22, 1986; and "San Onofre Nuclear Generating Station, Units 1, 2, and 3, *Safeguards Contingency Plan," with revisions submitted through December 29, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. SCE shall report any violations of the requirements contained in Section 2, items C(1), C(3)~~ through C(13), C(15) through C(22)~~, and F of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region IV, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on February 16, 2022.

Attachment D

(Proposed Amendments to Facility Operating License)
(Redline and Strikeout)

SONGS Unit 3

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

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS & ELECTRIC COMPANY
THE CITY OF RIVERSIDE, CALIFORNIA
THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

FACILITY OPERATING LICENSE

License No. NPF-15

Facility Operating License No. NPF-15 was issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report, as amended, through Amendment 30, and the Environmental Report, as amended, through Amendment 6.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3. Transshipment of Unit 1 fuel between Units 1 and 3 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 172, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

~~(4) Intentionally Deleted~~

(4) Containment Tendon Surveillance

Deleted by Amendment No. 26

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

Deleted by Amendment No. 49

~~This paragraph intentionally deleted.~~

The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(6) High Burnup Fission Gas Release (Section 4.2.2.2, SER)

Deleted by Amendment No.

~~Prior to beginning the cycle of reactor operation during which peak fuel pellet burnups will achieve greater than 200,000 megawatt days per metric ton of uranium, SCE shall provide revised analyses using fission gas release models acceptable to the NRC staff.~~

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Deleted by Amendment No.

~~Prior to operation for more than five (5) effective full power years, SCE shall provide a report describing its reexamination of the Technical Specification requirements for steam generator/reactor coolant system delta temperature and shutdown cooling system initiation temperature limits that are presently provided for overpressure protection. The report must either demonstrate that the current Technical Specification limits are still suitably conservative, or propose and justify revised limits.~~

(8) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, the volume control tank outlet valve control logic shall be modified to ensure that the valve does not change position following safety injection actuation signal reset. In the interim, SCE shall maintain emergency procedures that require the volume control tank outlet valve to be placed in the manual mode prior to SIAS reset.~~

(9) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, SCE shall comply with the recommendations of Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," as described in the SCE letter of May 13, 1982.~~

(10) Control System Failures (Section 7.7, SER, SSER #4)

Deleted by Amendment No.

~~By April 1, 1983, SCE shall provide an evaluation, for NRC staff review and approval, of control system failures caused by high energy line break, and by failures of any power sources, sensors, or sensor impulse lines which provide power or signals to two or more control systems. Implementation of any corrective action resulting from this evaluation shall be completed on a schedule acceptable to the NRC.~~

(11) Diesel Generator Modifications (Section 8.3.1, SER)

Deleted by Amendment No.

~~Prior to startup following the first refueling outage. SCE shall install a heavy duty turbocharger gear drive assembly on the emergency diesel generators.~~

(12) <u>Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)</u>

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(13) <u>Turbine Disc Inspection (Section 10.2.2, SER)</u>

Deleted by Amendment No.

~~Prior to startup following the second refueling outage, the bores of the low pressure turbine disc shall be inspected for flaws using ultrasonic testing. The results of the inspection shall be submitted to the NRC staff.~~

(14) <u>Radioactive Waste System (Section 11.1, SER, SSER #5)</u>

Deleted by Amendment No.

~~"Wet" solid radwaste shall not be shipped from the facility until the NRC has approved the waste solidification Process Control Program.~~

(15) <u>Purge System Monitors (Section 11.3, SER, SSER #5)</u>

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, equipment having the capability to continuously monitor and sample the containment purge exhaust directly from the purge stack shall be operable.~~

(16) <u>Initial Test Program (Section 14, SER)</u>

Deleted by Amendment No.

~~SCE shall conduct the post-fuel loading initial test program (set forth in Section 14 of the San Onofre Units 2 and 3 Final Safety Analysis Report, as amended, through Amendment 30) without making any major modifications to this program unless such modifications have been identified and have received prior NRC approval. Major modifications are defined as:~~

~~a. Elimination of any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.~~

~~b. Modification of test objectives, methods, or acceptance criteria for any test identified in Section 14 of the Final Safety Analysis Report, as amended, as being essential.~~

~~c. Performance of any test at a power level different than that described in the test procedure.~~

~~d. Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).~~

(17) NUREG-0737 Conditions (Section 22)

~~Each of the following conditions shall be completed to the satisfaction of the NRC. Each item references the related subpart of Section 22 of the SER and/or its supplements.~~

a. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No.

~~Emergency procedures based on guidelines approved by the NRC shall be implemented prior to startup following the first refueling outage that occurs six months or more after NRC approval of the guidelines.~~

b. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Deleted by Amendment No.

~~Prior to fuel loading, SCE shall implement a system for verifying the correct performance of operating activities, and shall keep the System in effect thereafter.~~

c. Control Room Design Review (I.D.1, SSER #1)

Deleted by Amendment No.

~~The control room modifications identified as required in Section 22, Item I.D.1 of Supplement No. 1 to the SER shall be installed and made operational on the schedules identified for each modification in Supplement No. 1 to the SER.~~

d. ~~Deleted~~
Post Accident Sampling System (NUREG-0737 Item II.B.3)

Deleted by Amendment No. 169

e. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

Deleted by Amendment No.

~~The safety valve position indication system shall be environmentally and seismically qualified consistent with the component or system to which it is attached, and documentation of this shall be maintained.~~

f. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #11)

Deleted by Amendment No.

~~Prior to exceeding five (5) percent power, SCE shall conduct a 48-hour endurance test of all auxiliary feedwater pumps. The results of the test shall be submitted to the NRC staff.~~

g. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

Deleted by Amendment No.

~~SCE shall maintain in effect procedures to preclude the automatic reapplication of pressurizer heaters to Class IE buses upon SIAS reset.~~

h. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

Deleted by Amendment No.

~~Prior to fuel loading, the following items shall be completed, and shall be maintained thereafter:~~

~~1. The subcooling monitors shall be modified to include the maximum unheated junction thermocouple temperature and the representative core exit thermocouple input.~~

~~2. Incore detector assemblies (core exit thermocouples and associated cabling) shall be environmentally qualified and shall have seismic and environmentally qualified Class 1E connectors.~~

~~3. Qualified cables shall be installed for the core exit thermocouples.~~

~~4. The heated junction thermocouple probe and associated process instrumentation shall be installed.~~

~~Prior to startup following the first refueling outage, the heated junction thermocouple system and the safety parameter display system shall be operable and shall be maintained operable thereafter.~~

i. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31, SSER #1

Deleted by Amendment No.

~~Within one year after model revisions are approved by the NRC, SCE shall provide a supplemental plant-specific analysis to verity compliance with 10 CFR 50.46, using the revised models developed under Item II.K.3.30.~~

j. Improving Licensee Emergency Preparedness (III.A.2, SSER #1, SSER #5

Deleted by Amendment No.

~~1. By January 1, 1983, the upgraded emergency support facilities shall be operational.~~

~~2. SCE shall maintain interim emergency support facilities (Technical Support Center, Operations Support Center and the Emergency Operations Facility) until the upgraded facilities are completed.~~

(18) Emergency Preparedness Conditions

Deleted by Amendment Nos. 8 and

~~a. Conditions of ASLB Initial Decision of May 14, 1982~~

~~By February 17, 1983, SCE shall:~~

~~1. Provide evidence that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (see Initial Decision, Section IV, Paragraph D.1-12, pp. 136-140), at no less than that level of readiness, pending development of satisfactory capability of offsite response Paragraph D.27, pp. 145-146, Section V, Paragraph B, pp. 213-214).~~

~~2. Provide an assessment of whether public information regarding emergency planning should also be presented in Spanish (see Initial Decision, Section IV, Paragraph F.32, pp. 168, and Section V, Paragraph C.2, pp. 215).~~

~~3. Provide revised plans demonstrating that the "extended" Emergency Planning Zone (EPZ) concept has been deleted from the San Onofre onsite and offsite plans and the Plume Exposure Pathway EPZ boundary has been extended, along with siren coverage, to Dana Point and all of San Juan Capistrano (see Initial Decision, Section IV, Paragraph D.25, pp. 98, and Section V, Paragraph C.5, pp. 216; see also Order (Making Clarifying Change in Initial Decision) dated May 25, 1982).~~

~~b. Completion of Emergency Preparedness Requirements~~

~~In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's proposed rules, 44 CFR 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR 50.54(s) (2) will apply.~~

~~c. Deleted Amendment No. 8 dated September 16, 1983~~

~~d. Conditions of ALAB 717, March 4, 1983~~

~~i. By July 2, 1983, SCE shall provide evidence that it has undertaken further efforts to assemble and to keep current as reasonably complete a list as possible of housebound people within the plume emergency planning zone who would require transportation assistance in the event of an evacuation.~~

~~ii. By July 2, 1983, SCE shall provide evidence that a training program has been developed and initiated to assist Orange County Transit District bus drivers in the discharge of their responsibilities in the event of a radiological emergency at San Onofre.~~

(19) RCS Depressurization System (PORV's)

Deleted by Amendment No.

~~By June 30, 1983, SCE shall provide a complete response to the NRC letter of March 27, 1982, including information relative to the capability of San Onofre 3 for rapid depressurization and decay heat removal without power operated relief valves (PORVs).~~

(20) Qualification of Auxiliary Feedwater (AFW) Pump Motor Bearings

Deleted by Amendment No.

~~Prior to startup following the first refueling outage, SCE shall install and make operational the lubrication oil cooling system for the auxiliary feedwater pump motor bearings described in SCE's letter of March 7, 1983. Prior to installation of the lube oil cooling system, SCE shall perform daily visual inspection of the steam lines in the AFW pump room in accordance with SCE's letter of July 12, 1982.~~

(21) Surveillance Program (Section 1.12, SSER #5)

Deleted by Amendment No.

~~Prior to entering any operational mode for the first time, including initial fuel loading, SCE shall:~~

~~a. Have completed a review of the surveillance procedures applicable to the change of mode, and determined that the procedures demonstrate the operability of the required systems with respect to all acceptance criteria defined in the Technical Specifications.~~

~~b. Have dispatched written certification to the NRC Regional Administrator, Region V, that the actions defined in (a), above, have been completed for the mode or modes to be entered.~~

(22) Auxiliary Building Ventilation System.

Deleted by Amendment No.

~~SCE shall complete all modifications to the auxiliary building ventilation system described in the November 5, 1982 letter from H. Ray, SCE, to R. Engelken, NRC, on the schedule proposed In the November 5, 1982 letter.~~

(23) Fuel Assembly Shoulder Gap Clearance (SCE letter of July 25, 1983)

Deleted by Amendment No.

~~Prior to entering Startup (Mode 2) after each refueling, SCE shall either provide a report that demonstrates that the existing fuel element assembly (FEA) has sufficient available shoulder gap clearance for at least the next cycle of operation, or identify to the NRC and implement a modified FEA design that has adequate shoulder gap clearance for at least the next cycle of operation. The commitment will apply until the NRC concurs that the shoulder gap clearance provided is adequate for the design life of the fuel.~~

(24) Isolation Capability for Primary EOF

Deleted by Amendment No.

~~By January 1, 1984 the primary EOF ventilation system shall be modified to provide isolation capability as described in the SCE letter of July 22, 1983.~~

(25) Correction of CPC Software Error

Deleted by Amendment No.

~~At the first outage of sufficient duration (7 days in Mode 5) after February 2, 1984, SCE shall correct the software error in the Core Protection Calculators discussed in the SCE letters dated March 7, 1893 and July 22, 1983.~~

(26) Auxiliary Feedwater System Monthly Reports

Deleted by Amendment No.

~~(26) Until the first refueling outage, SCE shall provide a monthly report describing any occurrences resulting in the degradation (including, but not limited to component failures, maintenance errors, and operator errors) of the auxiliary feedwater system. The report shall identify the cause of such occurrences. The report does not relieve the licensee from any existing requirements for Licensee Event Reports (LERs).~~

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which may contain Safeguards Information protected under 10 CFR 73.21, are entitled: "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Physical Security Plan," with revisions submitted through April 22, 1988; "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Security Force Training and Qualification Plan," with revisions submitted through October 22. 1986; and "San Onofre Nuclear Generating Station, Units 1, 2, and 3, *Safeguards Contingency Plan," with revisions submitted through December 29, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

G. SCE shall report any violations of the requirements contained in Section 2, items C(1), C(3) ~~through C(11), C(13) through C(22),~~ and F of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region IV, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on November 15, 2022.

K. Deleted by Amendment No.

Attachment 1 to NPF-15 - Deleted by Amendment No.

~~K. In accordance with the Commission's direction in its Statement of Policy, Licensing and Regulatory Policy and Procedures for Environmental Protection; Uranium Fuel Cycle Impacts, October 29, 1982, this license is subject to the final resolution of the pending litigation involving Table S-3. See Natural Resources Defense Council v. NRC, No. 74-1586 (D. C. Cir., April 27, 1982).~~

~~ATTACHMENT 1~~

~~TO~~

~~NPF-15~~

~~The following item must be completed prior to initial criticality:~~

~~The deficiency identified by the SCE letter, dated July 19, 1982, to R.H. Engelken from Dr. L.T. Papay regarding discrepant inputs to the Core Protection Calculator from Reactor Coolant Pump shaft speed and Control Element Assembly position indication shall be corrected~~

Attachment E

(Proposed Amendments to Facility Operating License)

SONGS Unit 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS AND ELECTRIC COMPANY
THE CITY OF RIVERSIDE, CALIFORNIA
THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2

FACILITY OPERATING LICENSE

License No NPF-10

Facility Operating License No. NPF-10 was issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in The Final Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess, use, and operate the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibrations, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2. Transshipment of Unit 1 fuel between Units 1 and 2 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 181, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Containment Tendon Surveillance

Deleted by Amendment No. 37

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

Deleted by Amendment No. 60

(6) High Burnup Fission Gas Release (Section 4.2.2.2. SER)

Deleted by Amendment No.

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Deleted by Amendment No.

(8) Control Room Pressurization Capability (Section 6.4, SER, SSER #5)

Deleted by Amendment No.

(9) Seismic Trip System (Section 7.2.5, SSER #4)

Deleted by Amendment No.

(10) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Deleted by Amendment No.

(11) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Deleted by Amendment No.

(12) Control System Failures (Section 7.7, SSER #4)

Deleted by Amendment No.

(13) Diesel Generator Modifications (Section 8.3.1, SER)

Deleted by Amendment No.

(14) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(15) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No.

(16) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No.

(17) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No.

(18) Initial Test Program (Section 14, SER)

Deleted by Amendment No.

(19) NUREG-0737 Conditions (Section 22)

Deleted by Amendment No.

a. Shift Technical Advisor (I.A.1.1, SSER #1)

Deleted by Amendment No.

b. Shift Manning (I.A.1.3. SSER #1, SSER #5)

Deleted by Amendment No. 147

c. Independent Safety Engineering Group (1.B.1.2, SSER #1)

Deleted by Amendment No.

d. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No.

e. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Deleted by Amendment No.

f. Control Room Design Review (I.D.1, SSER #1)

Deleted by Amendment No.

g. Special Low Power Testing and Training (I.G.1, SSER #1)

Deleted by Amendment No.

h. Reactor Coolant System Vents (II.B .1), SSER #1 , SSER #4)

Deleted by Amendment No.

i. Post-Accident Sampling System (NUREG-0737 Item II.B.3)

Deleted by Amendment No. 178

j. Safety Valve Test Requirements (II.D.1, SSER #1)

Deleted by Amendment No.

k. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

Deleted by Amendment No.

l. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #1)

Deleted by Amendment No.

m. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

Deleted by Amendment No.

n. Additional Monitoring Instrumentation (II.F.1, SSER #1, SSER #4)

Deleted by Amendment No.

o. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

Deleted by Amendment No.

p. Voiding in the Reactor Coolant System (II.K.2.17, SSER #1, SSER #5)

Deleted by Amendment No.

q. Revised Model for Small-Break LOCAs (II.K.3.30, SSER #1, SSER #4, SSER #5

Deleted by Amendment No.

r. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31), SSER #1)

Deleted by Amendment No.

s. Improving Licensee Emergency Preparedness (III.A.2, SSER#1, SSER #5)

Deleted by Amendment No.

(20) Surveillance Program (Section 1.12, SSER #5)

Deleted by Amendment No.

(21) Laboratory Instrumentation (Section 1.12, SSER #5)

Deleted by Amendment No.

(22) Design Verification Program (Section 3.7.4, SSER #5)

Deleted by Amendment No.

(23) Emergency Preparedness Conditions

Deleted by Amendment No.

(24) RCS Depressurization System (PORV's)

Deleted by Amendment No.

(25) Qualification of Auxiliary Feedwater Pump Motor Bearings

Deleted by Amendment No.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which may contain Safeguards Information protected under 10 CFR 73.21, are entitled: "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Physical Security Plan," with revisions submitted through April 22, 1988; "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Security Force Training and Qualification Plan," with revisions submitted through October 22, 1986; and "San Onofre Nuclear Generating Station, Units 1, 2, and 3, *Safeguards Contingency Plan," with revisions submitted through December 29, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

G. SCE shall report any violations of the requirements contained in Section 2, items C(1), C(3), and F of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region IV, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on February 16, 2022.

Attachment F

(Proposed Amendments to Facility Operating License)

SONGS Unit 3

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

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS & ELECTRIC COMPANY
THE CITY OF RIVERSIDE, CALIFORNIA
THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

FACILITY OPERATING LICENSE

License No. NPF-15

Facility Operating License No. NPF-15 was issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report, as amended, through Amendment 30, and the Environmental Report, as amended, through Amendment 6.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3. Transshipment of Unit 1 fuel between Units 1 and 3 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 172, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Containment Tendon Surveillance

Deleted by Amendment No. 26

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

Deleted by Amendment No. 49

(6) High Burnup Fission Gas Release (Section 4.2.2.2, SER)

Deleted by Amendment No.

The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Deleted by Amendment No.

(8) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Deleted by Amendment No.

(9) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Deleted by Amendment No.

(10) Control System Failures (Section 7.7, SER, SSER #4)

Deleted by Amendment No.

(11) Diesel Generator Modifications (Section 8.3.1, SER)

Deleted by Amendment No.

(12) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(13) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No.

(14) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No.

(15) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No.

(16) Initial Test Program (Section 14, SER)

Deleted by Amendment No.

(17) NUREG-0737 Conditions (Section 22)

Deleted by Amendment No.

a. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No.

b. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Deleted by Amendment No.

c. Control Room Design Review (I.D.1, SSER #1)

Deleted by Amendment No.

d. Post Accident Sampling System (NUREG-0737 Item II.B.3)

Deleted by Amendment No. 169

e. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

Deleted by Amendment No.

f. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #11)

Deleted by Amendment No.

g. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

Deleted by Amendment No.

h. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

Deleted by Amendment No.

i. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31, SSER #1

Deleted by Amendment No.

j. Improving Licensee Emergency Preparedness (III.A.2, SSER #1, SSER #5

Deleted by Amendment No.

(18) Emergency Preparedness Conditions

Deleted by Amendment Nos. 8 and

(19) RCS Depressurization System (PORV's)

Deleted by Amendment No.

(20) Qualification of Auxiliary Feedwater (AFW) Pump Motor Bearings

Deleted by Amendment No.

(21) Surveillance Program (Section 1.12, SSER #5)

Deleted by Amendment No.

(22) Auxiliary Building Ventilation System.

Deleted by Amendment No.

(23) Fuel Assembly Shoulder Gap Clearance (SCE letter of July 25, 1983)

Deleted by Amendment No.

(24) Isolation Capability for Primary EOF

Deleted by Amendment No.

(25) Correction of CPC Software Error

Deleted by Amendment No.

(26) Auxiliary Feedwater System Monthly Reports

Deleted by Amendment No.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which may contain Safeguards Information protected under 10 CFR 73.21, are entitled: "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Physical Security Plan," with revisions submitted through April 22, 1988; "San Onofre Nuclear Generating Station, Units 1, 2, and 3 Security Force Training and Qualification Plan," with revisions submitted through October 22, 1986; and "San Onofre Nuclear Generating Station, Units 1, 2, and 3, *Safeguards Contingency Plan," with revisions submitted through December 29, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. SCE shall report any violations of the requirements contained in Section 2, items C(1), C(3) and F of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region IV, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on November 15, 2022.

K. Deleted by Amendment No.

Attachment 1 to NPF-15 - Deleted by Amendment No.