ATTACHMENT 1

VOLUME 14

SAN ONOFRE NUCLEAR GENERATING STATION

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0 ADMINISTRATIVE CONTROLS

LIST OF ATTACHMENTS

1. **ITS 5.1 – Responsibility**
2. **ITS 5.2 – Organization**
3. **ITS 5.3 – Unit Staff Qualifications**
4. **ITS 5.4 – Procedures**
5. **ITS 5.5 – Programs and Manuals**
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7. **ITS 5.8 – High Radiation Area**

Note: There is no ITS 5.6

ATTACHMENT 1

ITS 5.1, RESPONSIBILITY

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

plant manager

5.1.1 5.1.1 The ~~corporate officer with direct responsibility for the plant~~ shall be responsible for overall unit operation and maintenance of Units 2 and 3 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence. LA01 M01

See ITS 5.2

INSERT 1

5.1.2 5.1.2 ~~The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.~~ A04 LA01 A02

Not used.

5.1.3 5.1.3 The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. ~~A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS).~~ During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior ~~Reactor~~ Operator's (S~~R~~O) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active S~~R~~O license or ~~Reactor~~ Operator's license shall be designated to assume the Control Room command function. A02 A03

(continued)

5.1

M01 **INSERT 1**

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

plant manager

5.1.1 5.1.1 The ~~corporate officer with direct responsibility for the plant~~ shall be responsible for overall unit operation and maintenance of Units 2 and 3 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence. LA01

See ITS 5.2

INSERT 1 M01

5.1.2 5.1.2 ~~The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.~~ A04 LA01 A02

Not used.

5.1.3 5.1.3 The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. ~~A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS).~~ During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior ~~Reactor~~ Operator's (S~~R~~O) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active S~~R~~O license or ~~Reactor~~ Operator's license shall be designated to assume the Control Room command function. A02 A03

(continued)

5.1

M01

INSERT 1

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 5.1.2 states, in part, "A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis." CTS 5.1.3 states, in part, "A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS)." This statement refers to the Control Room Supervisor (CRS) being responsible for the Control Room command function. ITS 5.1.1 and ITS 5.1.3 do not include these requirements. This changes the CTS by deleting the requirement to issue these management directives annually and the reference to what and where the control room area encompasses.

The purpose of CTS 5.1.2 and CTS 5.1.3 is to specify the responsibilities of the Shift Manager and Control Room Supervisor. Since the corporate officer is responsible for overall unit operation, as directed by CTS 5.1.1, then there is no need for these requirements in the Technical Specifications. In addition, the control room command function requirement has not changed. Furthermore, the location that describes what the control room area encompasses (i.e., the LCS) is also not required in the ITS since this description is more properly located in plant requirements (like the LCS) and the ITS does not need to specifically delineate where the description is located. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 5.1.3 uses the term "Senior Reactor Operator" and "Reactor Operator." ITS 5.1.3 uses the term "Senior Operator" and "Operator." This changes the CTS by modifying the terms for the two licensed individuals.

The new terms are being used since they are consistent with 10 CFR 55.4 and 10 CFR 50.54(m). These two regulations define the licensed individuals as "Senior Operator" and "Operator," not "Senior Reactor Operator" and "Reactor Operator." Therefore, this change is acceptable and is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 5.1.2 states, "The Shift Manager shall be responsible for the ultimate…general public." This portion is being relocated to the UFSAR as described in DOC LA01. CTS 5.1.2 also states, "A management directive to this effect…annual basis." This portion is being deleted as described in DOC A02.

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

With these changes, all CTS 5.1.2 has either been deleted or relocated. ITS 5.1.2 will state "Not used." This changes the CTS by replacing the current wording with "Not used."

The purpose of CTS 5.1.2 is to specify the responsibilities of the Shift Manager and Control Room Supervisor. These requirements are not being included in the ITS as justified in DOC A02 and LA01. The proposed change adds, "Not Used," to replace the previous CTS 5.1.2 wording. This change is acceptable because retaining the numbering consistent with the CTS will avoid the unnecessary administrative burden of changing Section numbers in plant procedures. This change is designated as administrative because it does not technically change the Specifications.

MORE RESTRICTIVE CHANGES

M01 ITS 5.1.1 requires that the plant manager or his designee approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety. The CTS does not include this requirement. This changes the CTS by adding an approved requirement for the plant manager (see DOC LA01 for the title change to plant manager) with direct responsibility for the plant or his designee.

The purpose of ITS 5.1.1 requirement is to provide additional assurance that the plant manager has direct responsibility for overall operation. This change is acceptable because having the plant manager or his designee approve actions affecting nuclear safety is consistent with 5.2.1.b requirement that the plant manager be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. This change is designated more restrictive because it adds a requirement for the plant manager or his designee to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.1.1 provides the responsibilities of the corporate officer. CTS 5.1.2 provides the responsibilities of the shift manager. ITS 5.1.1 provides the responsibilities of the plant manager. This changes the CTS by moving the specific organizational titles to the UFSAR and replacing it with a generic title.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

allowance to relocate the specific organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairman, dated November 10, 1994. Furthermore, both CTS and ITS 5.2.1.a require the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications to be documented in the UFSAR. Also, this type of change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e). This change is designated as a less restrictive removal of detail change because information relating to meeting Technical Specification requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

---REVIEWER'S NOTES---

1. ~~Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

2. ~~The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

1

5.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 Not used.

Control Room Supervisor (CRS)

4

5.1.3 5.1.~~2~~ 3 The ~~[Shift Supervisor (SS)]~~ shall be responsible for the control room command function. During any absence of the CRS ~~[SS]~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior ~~Reactor~~ Operator (S~~R~~O) license shall be designated to assume the control room command function. During any absence of the CRS ~~[SS]~~ from the control room while the unit is in MODE 5 or 6, an individual with an active S~~R~~O license or ~~Reactor~~ Operator license shall be designated to assume the control room command function.

2
3
2
3

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. The Reviewers Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator," not "Senior Reactor Operator" and "Reactor Operator."

4. ITS 5.1.2 is being added as a "not used" Section so the subsequent Section 5.1 numbering (ITS 5.1.2 was changed to ITS 5.1.3) can remain consistent with the CTS Section 5.1 numbering. The numbering changes are being made to be consistent with the CTS to avoid the unnecessary administrative burden of changing Section numbers in plant procedures.

5. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.1, RESPONSIBILITY

There are no specific NSHC discussions for this Specification.

ATTACHMENT 2

ITS 5.2, ORGANIZATION

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

plant manager

5.2.1.b b. The ~~corporate officer with direct responsibility for the plant~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. LA01

5.2.1.c c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

5.2.1.d d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

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5.2.2 5.2.2 UNIT STAFF

The unit staff organization shall include the following:

5.2.2.a a. A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.

With both units shutdown or defueled, a total of three non-Licensed operators are required for the two units.

Not used.

5.2.2.b b. ~~At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.~~ A02

, and 5.2.2.g

5.2.2.c c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) ~~and~~ 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. L01

5.2.2.d d. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5.2.2.e e. Deleted

operations manager or assistant operations manager

5.2.2.f f. The ~~Manager, Plant Operations~~ (at time of appointment)~~, Shift Managers, and Control Room Supervisors~~ shall hold a Senior ~~Reactor~~ Operator's license. LA01 LA02

An individual

5.2.2.g g. ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support to the ~~Shift Manager~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~The STA~~ shall ~~have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents~~. A04 A03

unit operations shift crew

This individual

meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift

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ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.2.1.b

See ITS 5.1

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall unit operation and maintenance of Units 2 and 3 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.

5.1.3 The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS). During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator's (SRO) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator's license shall be designated to assume the Control Room command function.

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ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

plant manager

5.2.1.b b. The ~~corporate officer with direct~~ ~~responsibility for the plant~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. LA01

5.2.1.c c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

5.2.1.d d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

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ITS

5.2.2 5.2.2 UNIT STAFF

The unit staff organization shall include the following:

5.2.2.a a. A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.

With both units shutdown or defueled, a total of three non-Licensed operators are required for the two units.

Not used.

5.2.2.b b. ~~At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.~~ A02

and 5.2.2.g

5.2.2.c , c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(I) ~~and~~ 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. L01

5.2.2.d d. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5.2.2.e e. Deleted

operations manager or assistant operations manager

5.2.2.f f. The ~~Manager, Plant Operations~~ (at time of appointment)~~, Shift Managers, and Control Room Supervisors~~ shall hold a Senior ~~Reactor~~ Operator's license. LA01 LA02

An individual

5.2.2.g g. ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support to the ~~Shift Manager~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~The STA~~ shall ~~have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents~~. A04 A03

unit operations shift crew

This individual

meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift

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ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.2.1.b

See ITS 5.1

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall unit operation and maintenance of Units 2 and 3 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.

5.1.3 The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS). During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator's (SRO) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator's license shall be designated to assume the Control Room command function.

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DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 5.2.2.b requires that at least one licensed Reactor Operator (RO) shall be in the control room when fuel is in the reactor. It also requires that when the unit is in MODE 1, 2, 3 or 4 that at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area. The ITS does not include these requirements. This changes the CTS by deleting these requirements.

10 CFR 50.54(m)(2)(iii) states "When a nuclear power unit is in operational mode other than cold shutdown or refueling, as defined by a unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be at the controls at all times." 10 CFR 50.54(m)(2)(iv) states "Each licensee shall have present, during alteration of the core of a nuclear power plant (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 change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.54(m)(2)(iii) and 10 CFR 50.54(m)(2)(iv). This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 5.2.2.g states, in part, that the STA provides advisory technical support to the Shift Manager. Additionally it provides qualification requirements for the Shift Technical Advisor (STA), and requires the STA to have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents. ISTS 5.2.2.f (ITS 5.2.2.g) requires the individual to provide advisory technical support to the unit operation shift crew and to meet the qualification requirements of the Commission Policy on Engineering Expertise on Shift for qualification requirements instead of listing the specific qualification requirements.

The purpose of the CTS 5.2.2.g STA requirements is to specify the minimum qualification requirements for the STA. This change is acceptable because the qualification requirements included in the Commission Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04, dated February 13, 1986) encompass the current STA qualification requirements. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

A04 CTS 5.2.2.f uses the term "Senior Reactor Operator." ITS 5.2.2.f uses the term "Senior Operator." This changes the CTS by modifying the terms for the two licensed individuals.

The new term is being used since it is consistent with 10 CFR 55.4 and 10 CFR 50.54(m). These two regulations define the licensed individuals as "Senior Operator," not "Senior Reactor Operator." Therefore, this change is acceptable and is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.2.1.b provides the responsibilities of the corporate officer. CTS 5.2.2.f requires, in part, the Manager, Plant Operations to hold a Senior Reactor Operator's license (DOC LA02 discusses the requirement for the Shift Managers and the Control Room Supervisors to hold a Senior Reactor Operator's license). ITS 5.2.1.b provides the responsibilities of the plant manager and ITS 5.2.2.f discusses the requirement for the operations manager or assistant manager to hold a Senior Operator's license (the change in the license name from Senior Reactor Operator to Senior Operator is discussed in DOC A04). This changes the CTS by moving the specific organizational titles to the UFSAR and replacing them with a generic title.

The removal of this detail, which is related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairman, dated November 10, 1994. Furthermore, both CTS and ITS 5.2.1.a require the plant specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications to be documented in the UFSAR. Also, this type of change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e). This change is designated as a less restrictive removal of detail change because information relating to meeting Technical Specification requirement is being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.2.2.f requires, in part, that the Shift Manager and Control Room Supervisors hold a Senior Reactor Operator's license (DOC LA01 discusses the requirement for the Manager, Plant Operations to hold a Senior Reactor Operator's license). ITS 5.2.2.f does not require the Shift Managers and Control Room Supervisors to hold a Senior Reactor Operator's license. This changes CTS by moving the requirement for the Shift Managers and Control Room Supervisor to hold a Senior Operator license (the change in the license name from Senior Reactor Operator to Senior Operator is discussed in DOC A04) to the UFSAR.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The requirement for shift supervision to hold Senior Operator licenses is contained in 10 CFR 50.54(m), and does not need to be repeated in the Technical Specifications. The relocation of the details of the shift supervisor personnel that are required to hold Senior Operator licenses to the UFSAR is acceptable considering the controls provided by regulations. Furthermore, both CTS and ITS 5.2.1.a require the plant specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications to be documented in the UFSAR. Also, this type of change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e). This change is designated as a less restrictive removal of detail change because information relating to meeting Technical Specification requirement is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 - Relaxation of LCO Requirement)* CTS 5.2.2.g requires a Shift Technical Advisor (STA) to provide technical support to the operating crew. ITS 5.2.2.g includes the same requirement, but ITS 5.2.2.c allows the position to be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. This changes the CTS by allowing the STA position to be vacant for a short time due to unexpected circumstances.

The purpose of CTS 5.2.2.g is to ensure an individual, trained in areas of thermal hydraulics, reactor engineering and plant analysis, is onsite to provide expertise to the plant with regard to these fields. However, under unusual circumstances, such as an unexpected and sudden illness of the onsite individual, the STA may not be available. This change allows a short time, 2 hours, to not meet the requirement, provided immediate action is taken to fill the position (e.g., call in a replacement STA). This allowance is similar to that allowed in CTS 5.2.2.c for an unexpected absence in the shift operating crew requirements. Therefore, since the time allowed is short, and immediate action to rectify the problem is required, this change is considered acceptable. This change is designated as less

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DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

restrictive because a 2 hour allowance is provided to not meet the STA position requirement.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan],~~ UFSAR. 1

5.1.1, 5.2.1.b b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant~~,~~. 3

5.2.1.c c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety~~, and~~. 2 3

5.2.1.d d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4~~.~~; 3

~~---REVIEWER'S NOTE---~~
~~Two unit sites~~ With both units shutdown or defueled~~ require~~, a total of three non-licensed operators are required for the two units. 4
~~--~~

5.2.2.b b. Not used.

5.2 Organization

5.2.2 Unit Staff (continued)

5.2.2.c c b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) , ~~and~~ 5.2.2.a and 5.2.2. g ~~f~~ for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. 7 3 TSTF-511-A

5.2.2.d d c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. 7

~~d. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., [licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel]).~~ 8

~~The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.~~

~~Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.~~

~~Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.~~

5.2.2.e e. Not used. 7

5.2.2.f f ~~e~~. The operations manager or assistant operations manager (at time of appointment) shall hold a ~~n~~ Senior Operator's ~~SRO~~ license. 7 5 6

5.2.2.g g ~~f~~. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. 7

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

1. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

2. ISTS 5.2.1.c has been modified to allow more than one corporate officer to be responsible for these requirements. This information was previously approved in Amendment 127 for Unit 2 and Amendment 116 for Unit 3 (ML021990684).

3. Changes are made to use correct punctuation, correct typographical errors or to make corrections consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. Since SONGS is a dual unit site, the requirement specified in the Reviewer's Note has been retained, consistent with the CTS requirements.

5. ISTS 5.2.2.e states that the operations manager or assistant operations manager shall hold an SRO license. ITS 5.2.2.f modifies this statement to allow the operations manager or assistant operations manager to have a Senior Operator's license (changed from SRO as discussed in JFD 6) at the time of appointment. This information was previously approved in Amendment 127 for Unit 2 and Amendment 116 for Unit 3 (ML021990684).

6. The generic positions have been used. Also, terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator," not "Senior Reactor Operator" and "Reactor Operator."

7. The Specification number has been changed to be consistent with the Specification number in the SONGS CTS. SCE has decided not to renumber the CTS to be consistent with the ISTS because by doing so would result in the unnecessary administrative burden of changing TS numbers in plant procedures.

8. The working hour restrictions in ISTS 5.2.2.d have not been included since they have been deleted from the SONGS Units 2 and 3 CTS as documented in the NRC Safety Evaluation for Amendments 221 and 214, respectively, dated 10/20/09 (ADAMS Accession No. ML092880169). This is also consistent with TSTF-511.

9. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.2, ORGANIZATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 5.3, UNIT STAFF QUALIFICATIONS

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A01

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.

In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Multi-discipline supervisors shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.

c. Training: Complete the multi-discipline supervisor training program.

INSERT 1

A02

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INSERT 1

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

A01

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.

In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Multi-discipline supervisors shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.

c. Training: Complete the multi-discipline supervisor training program.

INSERT 1

A02

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INSERT 1

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

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DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 ITS 5.3.2 states "For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m)." The CTS does not include such a statement. This changes the CTS by clarifying the functions Senior Operators and Operators perform (i.e., those described in 10 CFR 50.54(m).

This change is acceptable because it clarifies the existing relationship between the Technical Specifications and regulations regarding licensed Senior Operator and Operator qualification requirements. SONGS Units 2 and 3 are already required to meet 10 CFR 50.54(m) requirements, since it is a regulation. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.3 5.3 Unit Staff Qualifications

~~---REVIEWER'S NOTE---
Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.
---~~ 2

5.3.1 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or~~ ANSI ~~Standard acceptable to the NRC staff]. [The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ 1

INSERT 1

5.3.2 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior ~~Reactor~~ Operator ~~(SRO)~~ and a licensed ~~Reactor~~ Operator ~~(RO)~~ are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m). 3

1

INSERT 1

N18.1-1971 for comparable positions, except: a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975; and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.

In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Multi-discipline supervisors shall meet or exceed the following qualifications:

a. Education: Minimum of a high school diploma or equivalent;

b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant; and

c. Training: Complete the multi-discipline supervisor training program.

JUSTIFICATION FOR DEVIATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

1. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

2. The Reviewers Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

3. Terms in 10 CFR 55.4 ad 10 CFR 50.54(m) are "Senior Operator" and "Operator," not "Senior Reactor Operator" and "Reactor Operator."

4. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 4

ITS 5.4, PROCEDURES

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A01

4

5.0 ADMINISTRATIVE CONTROLS

5.~~5~~ Procedures~~, Programs, and Manuals~~

4

5.4 5.~~5.1~~ Procedures

~~5.5.1.1 Scope~~

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

5.4.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

5.4.1.b b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

5.4.1.c c. Quality assurance for effluent and environmental monitoring ~~using the guidance in Regulatory Guide 4.15, Revision 1, 1979~~; LA01

5.4.1.d d. Fire Protection Program implementation; and

5.4.1.e e. Programs, as specified in Specification 5.5.2.

5.4.1.f f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

(continued)

A01

ITS

4

5.0 ADMINISTRATIVE CONTROLS

5.~~5~~ Procedures~~, Programs, and Manuals~~

4

5.4 5.~~5.1~~ Procedures

~~5.5.1.1 Scope~~

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

5.4.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

5.4.1.b b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

5.4.1.c c. Quality assurance for effluent and environmental monitoring ~~using the guidance in Regulatory Guide 4.15, Revision 1, 1979~~; LA01

5.4.1.d d. Fire Protection Program implementation; and

5.4.1.e e. Programs, as specified in Specification 5.5.2.

5.4.1.f f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

(continued)

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DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.5.1.1.c requires written procedures to be established, implemented and maintained for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979. ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the reference to the Regulatory Guide to the UFSAR.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for written procedures for effluent and environmental monitoring. Also, this change is acceptable because these types of procedural details will be adequately controlled in the UFSAR. This change is designated as less restrictive removal of detail because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.5.1

5.4 Procedures

5.5.1

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

5.5.1.1.a

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978~~,~~ ; 2

5.5.1.1.b

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33]~~,~~ ; 2 3 2

5.5.1.1.c

c. Quality assurance for effluent and environmental monitoring~~,~~ ; 2

5.5.1.1.d

d. Fire Protection Program implementation~~,~~ ; and 2

5.5.1.1.e

e. All programs specified in Specification 5.5 .2. 1

5.5.1.1.f

[f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation. 3

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.] 3

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. The Specification number has been changed to be consistent with the Specification number in the SONGS CTS. SCE has decided not to renumber the CTS to be consistent with the ISTS because by doing so would result in the unnecessary administrative burden of changing TS numbers in plant procedures.

2. Changes are made to use correct punctuation, correct typographical errors or to make corrections consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

4. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.4, PROCEDURES

There are no specific NSHC discussions for this Specification.

ATTACHMENT 5

ITS 5.5, PROGRAMS AND MANUALS

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A01

~~TS Bases Control~~ Programs and Manuals

5.~~4~~ 5

5.5 Programs and Manuals

~~5.0 ADMINISTRATIVE CONTROLS~~

5.5.2.2 ~~5.4~~ 5.5.2.2 Technical Specifications (TS) Bases Control Program

~~5.4.1~~ a. Changes to the Bases of the TS shall be made under appropriate administrative controls.

~~5.4.2~~ b. Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

~~a.~~ 1. A change in the TS incorporated in the license; or

~~b.~~ 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

~~5.4.3~~ c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

~~5.4.4~~ d. Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC within 6 months following every Unit 3 refueling, not to exceed 24 months. This schedule is consistent with SCE's submittal of UFSAR updates as allowed by the NRC approved exemption from 10 CFR 50.71(e) dated April 27, 1999.

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ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

See ITS 5.4

e. Programs, as specified in Specification 5.5.2.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

(continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2 Programs and Manuals

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

5.5.2.1 5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and the Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

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

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

~~3. Documentation of the fact that the change has been reviewed and found acceptable.~~

LA01

plant manager

b. Shall become effective upon ~~review and~~ approval by the ~~corporate officer with direct responsibility for the plant or designee~~.

LA05

(continued)

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.1.1 5.5.2.1.1 Licensee-initiated changes to the ODCM: (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 5.5.2.3 Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

(continued)

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ITS

A01

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.3 5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Controls Program surveillance frequency.

A03

5.5.2.4 5.5.2.4 Component Cyclic or Transient Limit Program

This program provides controls to track the UFSAR Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

(continued)

5.5.2.5 5.5.2.5 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel.

A07

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the flywheels each 10 years.

5.5.2.6 5.5.2.6 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to measure the values of the critical parameters;

c. Identification of process sampling points~~;~~ which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage

M05

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off-control point chemistry conditions; and

f. A procedure identifying (a) the authority responsible for interpretation of data and (b) the sequence and timing of administrative events, required to initiate corrective action.

5.5.2.7 5.5.2.7 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following methodology comparable with Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

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5.5.2.7 5.5.2.7 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for the concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank and fed into the gaseous radwaste vent system is less than the amount that would result in a whole body exposure of greater than or equal to 0.5 rem to any individual in the unrestricted area, in the event of an uncontrolled release of the tanks contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.2.8 5.5.2.8 Primary Coolant Sources Outside Containment Program

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), and the liquid radwaste

(continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.8 5.5.2.8 Primary Coolant Sources Outside Containment Program (continued)

system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path). The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

least once per 24 months A08

b. Integrated leak test requirements for each system at ~~refueling cycle intervals or less~~.

The provisions of SR 3.0.2 are applicable. A09

5.5.2.9 5.5.2.9 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containment, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. ~~Program itself is relocated to the LCS.~~

INSERT 1 M01

5.5.2.10 5.5.2.10 Inservice ~~Inspection and~~ Testing Program

This ~~program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components and Code Class CC and MC components including applicable supports. The~~ program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. ~~The program itself is located in the LCS.~~ LA02

INSERT 2 M02

5.5.2.11 5.5.2.11 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

(continued)

M01

INSERT 1

The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

M02

INSERT 2

The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.2.11 5.5.2.11 Steam Generator (SG) Program (continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.11 5.5.2.11 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.

In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

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A01

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.12

5.5.2.12 Ventilation Filter Testing Program (VFTP)

This Program establishes the required testing of the Engineered Safety Feature filter ventilation system "Control Room Emergency Air Cleanup System." The frequency of testing shall be in accordance with Regulatory Guide 1.52, Revision 2. As a minimum the VFTP program shall include the following:

5.5.2.12.a

a. Inplace testing of the high efficiency particulate air (HEPA) filters to demonstrate ~~acceptable~~ penetration and system bypass < 0.05 % when tested at the ~~appropriate~~ system flowrate of 35,705 cfm ± 10% in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and M07

5.5.2.12.b

b. Inplace testing of the charcoal adsorber to demonstrate ~~acceptable~~ penetration and system bypass < 0.05 % when tested at the ~~appropriate~~ system flowrate of 35,705 cfm ± 10% in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and M07

5.5.2.12.c

c. Laboratory testing of charcoal adsorber samples obtained in accordance with Regulatory Guide 1.52, Revision 2 and tested per the methodology of ASTM D3803-1989 at 30°C and 70% relative humidity to show ~~acceptable~~ < 1% methyl iodide penetration; and M07

5.5.2.12.d

d. Testing to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers, when tested at the ~~appropriate~~ system flowrate of 35,705 cfm ± 10% and a Delta P of 8.3 inches of water M07

5.5.2.12.a, 5.2.2.12.b

Note 1: Sample and injection points shall be qualified per ANSI N510-1975 unless manifolds have been qualified per ASME N510-1989. ~~HEPA testing will be conducted with DOP aerosol or suitable alternate.~~

(continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.12 5.5.2.12 Ventilation Filter Testing Program (VFTP) (continued)

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

5.5.2.13 5.5.2.13 Diesel Fuel Oil Testing Program

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

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

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

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

   3. a water and sediment content within limits.

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following ~~sampling and~~ addition to the storage tanks, ~~with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3~~; and, A04

.

c. Total particulate concentration of fuel oil is ≤ 10 mg/l when tested every 92 days, ~~in accordance with ASTM D-2276, Method A~~. LA03

~~5.5.2.14~~ ~~Deleted~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies. L01

(continued)

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

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5.5.2.15 5.5.2.15 Containment Leakage Rate Testing Program

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

M03

INSERT 3

~~NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the March 31, 1995 Type A Test shall be performed no later than March 30, 2010.~~

A02

5.5.2.15.b The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 48.0 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The containment design pressure is 60 psig.

M06

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

5.5.2.15.d Leakage rate acceptance criteria are:

5.5.2.15.d.1 a. The Containment overall leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for the Type A tests;

5.5.2.15.d.2 b. Air lock testing acceptance criteria are:

5.5.2.15.d.2.a) 1) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

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

M03

INSERT 3

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.15 5.5.2.15 Containment Leakage Rate Testing Program (Continued)

~~The provisions of Surveillance Requirement 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, t~~ A06

5.5.2.15.f Test frequencies specified in this Program may be extended consistent with the guidance provided in NEI 94-01, "Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has these provisions for test frequencies extension:

1. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

2. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for the recommended surveillance frequency for Type B and Type C testing may be extended by up to 25 percent of the test interval, not to exceed 15 months.

5.5.2.15.e The provisions of Surveillance Requirement 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.2.16 5.5.2.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACUS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

(continued)

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.16 5.5.2.16 Control Room Envelope Habitability Program (Continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following is an exception to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

Appropriate application of ASTM E-741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACUS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

(continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.17 5.5.2.17 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to verify that the remaining cells are above 2.07 V when a battery cell or cells have been found less than 2.13 V, and

c. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates. M04

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M04

INSERT 4

5.5.2.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI-04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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Programs and Manuals ~~SFDP~~

5.~~6~~ 5

5.5 Programs and Manuals

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.6~~ 5.5.2.14 Safety Function Determination Program (SFDP)

5.5.2.14 5.6.1 This program ensures loss of safety function is detected and appropriate actions taken. Upon ~~failure to meet two or more LCOs at the same time~~ entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. A05

5.5.2.14.a 5.6.2 The SFDP shall contain the following:

5.5.2.14.a.1 a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected.

5.5.2.14.a.2 b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists.

5.5.2.14.a.3 c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities.

5.5.2.14.a.4 d. Other appropriate limitations and remedial or compensatory actions.

5.5.2.14.b 5.6.3 A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and: A05

5.5.2.14.b.1 a. A required system redundant to system(s) supported by the inoperable support system is also inoperable ~~(Case A)~~; or LA04

5.5.2.14.b.2 b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable ~~(Case B)~~; or LA04

5.5.2.14.b.3 c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable ~~(Case C)~~. LA04

(continued)

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5.5 Programs and Manuals

Programs and Manuals ~~SFDP~~
5.~~6~~ 5

~~5.6 Safety Function Determination Program (SFDP)~~

5.6.3
(continued)

~~Generic Example:~~

~~Train A~~		~~Train B~~	
~~System i~~		~~System i~~	~~← Case C~~
~~+~~		~~+~~	
~~System ii~~	~~← (Support System) Inoperable~~	~~System ii~~	
~~+~~		~~+~~	
~~System iii~~		~~System iii~~	~~← Case A~~
~~+~~		~~+~~	
~~System iv~~		~~System iv~~	~~← Case B~~

LA04

5.5.2.14.c 5.6.4 The Safety Function Determination Program identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

A05

(continued)

ITS

A01

~~TS Bases Control~~ Programs and Manuals
5.~~4~~ 5

~~5.0 ADMINISTRATIVE CONTROLS~~ 5.5 Programs and Manuals

5.5.2.2 ~~5.4~~ 5.5.2.2 Technical Specifications (TS) Bases Control Program

~~5.4.1~~ a. Changes to the Bases of the TS shall be made under appropriate administrative controls.

~~5.4.2~~ b. Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

~~a.~~ 1. A change in the TS incorporated in the license; or

~~b.~~ 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

~~5.4.3~~ c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

~~5.4.4~~ d. Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC within 6 months following every Unit 3 refueling, not to exceed 24 months. This schedule is consistent with SCE's submittal of UFSAR updates as allowed by the NRC approved exemption for 10 CFR 50.71(e) dated April 27, 1999.

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5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

See ITS 5.4

e. Programs, as specified in Specification 5.5.2.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

(continued)

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5.5.2 Programs and Manuals

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

5.5.2.1 5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and the Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

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

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

~~3. Documentation of the fact that the change has been reviewed and found acceptable.~~ LA01

plant manager

b. Shall become effective upon ~~review and~~ approval by the ~~corporate officer with direct responsibility for the plant or designee~~. LA05

(continued)

5.5.2.1.1 5.5.2.1.1 Licensee-initiated changes to the ODCM: (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 5.5.2.3 Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

(continued)

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5.5.2.3 5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Controls Program surveillance frequency.

A03

5.5.2.4 5.5.2.4 Component Cyclic or Transient Limit Program

This program provides controls to track the UFSAR Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

(continued)

5.5.2.5 5.5.2.5 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel.

A07

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the flywheels each 10 years.

5.5.2.6 5.5.2.6 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to measure the values of the critical parameters;

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

M05

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off-control point chemistry conditions; and

f. A procedure identifying (a) the authority responsible for interpretation of data and (b) the sequence and timing of administrative events, required to initiate corrective action.

5.5.2.7 5.5.2.7 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following methodology comparable with Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

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5.5.2.7 5.5.2.7 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for the concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank and fed into the gaseous radwaste vent system is less than the amount that would result in a whole body exposure of greater than or equal to 0.5 rem to any individual in the unrestricted area, in the event of an uncontrolled release of the tanks contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.2.8 5.5.2.8 Primary Coolant Sources Outside Containment Program

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), and the liquid radwaste

(continued)

5.5.2.8 5.5.2.8 Primary Coolant Sources Outside Containment Program (continued)

system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path). The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months ~~refueling cycle intervals or less~~. The provisions of SR 3.0.2 are applicable.

A08

A09

5.5.2.9 5.5.2.9 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containment, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. ~~Program itself is relocated to the LCS.~~ INSERT 1

M01

5.5.2.10 5.5.2.10 Inservice ~~Inspection and~~ Testing Program

This ~~program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components and Code Class CC and MC components including applicable supports. The~~ program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. ~~The program itself is located in the LCS.~~ INSERT 2

LA02

M02

5.5.2.11 5.5.2.11 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

(continued)

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INSERT 1

The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

M02

INSERT 2

The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

to

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified in the Inservice Testing Program for performing inservice testing activities;

as 2 years or less

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

A01

5.5.2.11

5.5.2.11 Steam Generator (SG) Program (continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.5.2.11 5.5.2.11 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.

In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals (continued)

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5.5.2.12

5.5.2.12 Ventilation Filter Testing Program (VFTP)

This Program establishes the required testing of the Engineered Safety Feature filter ventilation system "Control Room Emergency Air Cleanup System." The frequency of testing shall be in accordance with Regulatory Guide 1.52, Revision 2. As a minimum the VFTP program shall include the following:

5.5.2.12.a

a. Inplace testing of the high efficiency particulate air (HEPA) filters to demonstrate ~~acceptable~~ penetration and system bypass < 0.05 % when tested at the ~~appropriate~~ system flowrate of 35,705 cfm ± 10% in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and M07

5.5.2.12.b

b. Inplace testing of the charcoal adsorber to demonstrate ~~acceptable~~ penetration and system bypass < 0.05 % when tested at the ~~appropriate~~ system flowrate of 35,705 cfm ± 10% in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and M07

5.5.2.12.c

c. Laboratory testing of charcoal adsorber samples obtained in accordance with Regulatory Guide 1.52, Revision 2 and tested per the methodology of ASTM D3803-1989 at 30°C and 70% relative humidity to show ~~acceptable~~ < 1% methyl iodide penetration; and M07

5.5.2.12.d

d. Testing to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers, when tested at the ~~appropriate~~ system flowrate of 35,705 cfm ±10% and a Delta P of 8.3 inches of water. M07

5.5.2.12.a, 5.2.2.12.b

Note 1: Sample and injection points shall be qualified per ANSI N510-1975 unless manifolds have been qualified per ASME N510-1989. ~~HEPA testing will be conducted with DOP aerosol or suitable alternate.~~

(continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals (continued)

5.5.2.12 5.5.2.12 Ventilation Filter Testing Program (VFTP) (continued)

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

5.5.2.13 5.5.2.13 Diesel Fuel Oil Testing Program

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

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

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

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

3. a water and sediment content within limits.

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following ~~sampling and~~ addition to the storage tanks, ~~with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3~~; and, A04

c. Total particulate concentration of fuel oil is ≤ 10 mg/l when tested every 92 days ~~in accordance with ASTM D-2276, Method A~~. LA03

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The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals (continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals (continued)

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5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.15 5.5.2.15 Containment Leakage Rate Testing Program

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

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~~NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the September 10, 1995 Type A Test shall be performed prior to startup from Unit 3 Cycle 16 refueling outage, which is scheduled to commence in the fall of 2010 and to end in the first quarter of 2011. SONGS Unit 3 shall not operate past September 9, 2011 until the Type A Test is satisfactorily completed.~~

A02

5.5.2.15.b The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 48.0 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The containment design pressure is 60 psig.

M06

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

5.5.2.15.d Leakage rate acceptance criteria are:

5.5.2.15.d.1 a. The Containment overall leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for the Type A tests;

5.5.2.15.d.2 b. Air lock testing acceptance criteria are:

5.5.2.15.d.2.a) 1) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$.

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

(continued)

M03

INSERT 3

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

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5.5.2.15 5.5.2.15 Containment Leakage Rate Testing Program (Continued)

A06

~~The provisions of Surveillance Requirement 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However,~~ test frequencies specified in this Program may be extended consistent with the guidance provided in NEI 94-01, "Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has these provisions for test frequencies extension:

5.5.2.15.f

1. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

2. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for the recommended surveillance frequency for Type B and Type C testing may be extended by up to 25 percent of the test interval, not to exceed 15 months.

5.5.2.15.e The provisions of Surveillance Requirement 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.2.16 5.5.2.16 Control Room Envelope Habitability Program

A Control Room Envelope Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACUS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

(continued)

5.5.2.16 5.5.2.16 Control Room Envelope Habitability Program (Continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following is an exception to Sections C.1 and C.2 of regulatory Guide 1.197, Revision 0.

Appropriate application of ASTM E-741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACUS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

(continued)

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A01

5.5 ~~Procedures,~~ Programs~~,~~ and Manuals ~~(continued)~~

5.5.2.17

5.5.2.17 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to verify that the remaining cells are above 2.07 V when a battery cell or cells have been found less than 2.13 V, and

c. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates. INSERT 4 M04

M04

INSERT 4

5.5.2.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI-04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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5.5 Programs and Manuals

A01

5.5 Programs and Manuals ~~SFDP~~

5.~~6~~ 5

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.6~~ 5.5.2.14 Safety Function Determination Program (SFDP)

5.5.2.14 5.6.1 This program ensures loss of safety function is detected and appropriate actions taken. Upon ~~failure to meet two or more LCOs at the same time~~ entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. A05

5.5.2.14.a 5.6.2 The SFDP shall contain the following:

5.5.2.14.a.1 a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected.

5.5.2.14.a.2 b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists.

5.5.2.14.a.3 c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities.

5.5.2.14.a.4 d. Other appropriate limitations and remedial or compensatory actions.

5.5.2.14.b 5.6.3 A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and: A05

5.5.2.14.b.1 a. A required system redundant to system(s) supported by the inoperable support system is also inoperable ~~(Case A)~~; or LA04

5.5.2.14.b.2 b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable ~~(Case B)~~; or LA04

5.5.2.14.b.3 c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable ~~(Case C)~~. LA04

(continued)

ITS

5.5 Programs and Manuals

~~SFDP~~ 5.~~6~~ 5

~~5.6 Safety Function Determination Program (SFDP)~~

5.6.3 (continued)

~~Generic Example:~~

~~Train A~~		~~Train B~~	
~~System i~~		~~System i~~	~~← Case C~~
~~+~~		~~+~~	
~~System ii~~	~~← (Support System) Inoperable~~	~~System ii~~	
~~+~~		~~+~~	
~~System iii~~		~~System iii~~	~~← Case A~~
~~+~~		~~+~~	
~~System iv~~		~~System iv~~	~~← Case B~~

LA04

5.5.2.14.c 5.6.4 The Safety Function Determination Program identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

A05

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DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 5.5.2.15 contains the following requirements for Unit 2, "NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the March 31, 1995 Type A Test shall be performed no later than March 30, 2010," and for Unit 3, "NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the September 10, 1995 Type A Test shall be performed prior to startup from the Unit 3 Cycle 16 refueling outage, which is scheduled to commence in the fall of 2010 and to end in the first quarter of 2011. SONGS Unit 3 shall not operate past September 9, 2011 until the Type A Test is satisfactorily completed." ITS 5.5.2.15 will not contain these requirements. This changes the CTS by deleting requirements that have been completed and are therefore not necessary.

The purpose of these CTS exceptions is to provide the latest date the Type A test can be performed. The proposed change will delete requirements from the CTS that are no longer required. The Unit 2 requirement requires the first Type A test performed after the March 31, 1995 Type A test to be performed no later than March 30, 2010. The Unit 3 requirement requires the first Type A Test performed after the September 10, 1995 Type A Test to be performed prior to startup from the Unit 3 Cycle 16 refueling outage, which commenced in the fall of 2010 and to end the first quarter of 2011. It further required that SONGS Unit 3 shall not operate past September 9, 2011 until the Type A Test is satisfactorily completed. This change is acceptable because both these requirements have been met and are therefore not needed anymore. This change is designated as administrative because requirements that are no longer applicable are being deleted.

A03 CTS 5.5.2.3 specifies the requirements for the Radioactive Effluent Controls Program, however there is no statement as to whether or not the provisions of SR 3.0.2 and SR 3.0.3 are applicable. ITS 5.5.2.3 (ISTS 5.5.4) states that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies. This changes the CTS by adding the allowances of ITS SR 3.0.2 and SR 3.0.3 to the Radioactive Effluent Controls Program.

This statement is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.2 and SR 3.0.3 are not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered

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administrative in nature. Furthermore, these requirements were in the CTS LCO and SR sections of the CTS prior to them being moved to a program. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 5.5.2.13.b requires verification that other properties for fuel oil are within limits within 31 days following sampling and addition to the storage tanks, with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3. ITS 5.5.2.13.b states that within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in 5.5.2.13.a, are within limits for ASTM 2D fuel oil. This changes the CTS by removing the word "sampling" and deleting the phrase "with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3."

The purpose of the diesel fuel oil program is to ensure that the quality of the stored diesel fuel oil is acceptable so that the emergency diesel generators can perform their safety function. These changes are acceptable since there is no need to state that the verification is required 31 days after sampling and addition. Since the fuel oil cannot be added until after the sample is taken, there is no need to reiterate that it is performed after sampling.

A05 CTS 5.6.1 states, in part, that when there is a failure to meet two or more LCOs at the same time that an evaluation must be made to determine if loss of safety function exists. ITS 5.5.2.14 requires that whenever LCO 3.0.6 is entered, that an evaluation shall be made to determine if a loss of safety function exists. CTS 5.6.3 states, in part, that a loss of safety function exists when a safety function assumed in the accident analysis cannot be performed, assuming no concurrent single failure. ITS 5.5.2.14.b states, in part, that a loss of safety function exists when a safety function assumed in the accident analysis cannot be performed, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s). ITS 5.5.2.14 contains a statement that when a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system. CTS 5.6 does not contain this statement. This changes the CTS by clarifying that the SFDP does not have to consider a loss of power for determining a loss of function.

This change is acceptable because the requirements in the SFDP are being clarified to state that consideration does not have to be made for a loss of power in determining loss of function. The NUREGs were developed such that the Actions for a single support system inoperability would be addressed by that support system's Actions without cascading to the supported system, even if both trains of the support system were inoperable resulting from a loss of function. This intent is clarified in the LCO 3.0.6 Bases. Without this clarification, supported systems with a single support system (such as both Containment Spray and ECCS trains supported by the Refueling Water Tank) would be declared inoperable when the support system is inoperable under the provisions of LCO 3.0.6 even though the support system Actions were designed to provide the appropriate response. This change is considered administrative because it does not result in technical changes to the CTS.

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A06 CTS 5.5.2.15 states, in part, that the provisions of Surveillance Requirement 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. ITS 5.5.2.15 does not contain this statement. This changes the CTS by not including the statement that SR 3.0.2 does not apply to the Containment Leakage Rate Testing Program.

The purpose of the Containment Leakage Rate Testing Program is to implement the leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J. The statement that SR 3.0.2 does not apply is not required to be included in the Chapter 5 Programs. None of the 3.0 Surveillance Requirements apply to Chapter 5 Programs unless they are specifically stated in the program. Therefore, the absence of the statement means that SR 3.0.2 does not apply. This change is considered administrative because it does not result in technical changes to the CTS.

A07 ISTS 5.5.2.5 contains a general statement for the Reactor Coolant Flywheel Program which says "This program shall provide for the inspection of each reactor coolant pump flywheel." CTS 5.5.2.5 does not contain this statement. This changes the CTS by stating what the program will provide.

The statement being added is a general statement used in the ITS format. Therefore, this change is acceptable and designated as administrative since there is no technical change to the CTS.

A08 CTS 5.5.2.8.b requires that the Primary Coolant Sources Outside Containment Program shall include integrated leak test requirements for each system at refueling cycle intervals or less. ITS 5.5.2.8.b contains the same requirements, but specifies the time frame as "at least once per 24 months." This changes the CTS by specifying the refueling cycle interval as 24 months.

This change is acceptable because SONGS is in the process of going to a 24 month refueling cycle. Additionally, SONGS has evaluated the applicable SRs in the Technical Specification for extension to 24 months. Therefore, it is acceptable to state "at least once per 24 months" in place of the words "at refueling cycle intervals." Since this change is a clarification of the time period in which the refueling cycle is performed, it is considered administrative in nature. This change is considered administrative because it does not result in technical changes to the CTS.

A09 CTS 5.5.2.8 specifies the requirements for the Primary Coolant Sources Outside Containment Program, however there is no statement as to whether or not the provisions of SR 3.0.2 are applicable. ITS 5.5.2.8 states that the provisions of SR 3.0.2 are applicable to the Primary Coolant Sources Outside Containment Program test frequencies. This changes the CTS by adding the allowances of ITS SR 3.0.2 to the Primary Coolant Sources Outside Containment Program.

This statement is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.2 is not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in

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nature. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 5.5.2.9 states that the Pre-Stressed Concrete Containment Tendon Surveillance Program is relocated to the License Controlled Specifications (LCS). ITS 5.5.2.9 requires the Tendon Surveillance Program, inspection frequencies, and acceptance criteria to be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC. Additionally, ITS 5.5.2.9 states that the provisions of SR 3.0.3 are applicable to the Pre-Stressed Concrete Containment Tendon Surveillance Program test frequencies. This changes the CTS by including the program, inspection frequencies, and acceptance criteria in the Technical Specifications.

The purpose of the Pre-Stressed Concrete Containment Tendon Surveillance Program is to monitor for tendon degradation to ensure the containment structure's integrity. The containment consists of a pre-stressed, reinforced concrete, cylindrical structure with a hemispherical dome. The post-tensioning system used for the shell and dome of the containment employs tendons. Each tendon consists of high strength steel wires and anchoring components. The prestressing load is transferred, by cold formed button heads on the ends of the individual wires through stressing washers, to steel bearing plates embedded in the structure. The unbonded tendons are installed in tendon ducts and tensioned in a predetermined sequence. The addition of the inspection testing frequencies and acceptance criteria will verify, in the Technical Specification, that this degradation does not take place. Additionally, it allows control of the program to be in one location. Furthermore, this makes the Pre-Stressed Concrete Containment Tendon Surveillance Program consistent with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This change is designated as more restrictive since requirements for the Pre-Stressed Concrete Containment Tendon Surveillance Program are being added to the Technical Specifications.

M02 CTS 5.5.2.10 states that the Inservice Testing Program is located in the LCS. ITS 5.5.2.10 requires that the Inservice Testing Program be tested at the frequencies specified in the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code). Additionally, ITS 5.5.2.10 states that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Inservice Testing Program and that the ASME OM Code cannot supersede the requirements of any Technical Specification. This changes the CTS by adding the specific requirements for the Inservice Testing Program to the Technical Specifications.

The purpose of the Inservice Testing Program is to provide controls for testing of ASME Code Class 1, 2, and 3 components. The addition of the testing frequencies will verify, in the Technical Specification, that ASME Code Class 1, 2, and 3 components remain OPERABLE. Additionally, it allows control of the program to be in one location. This change is designated as more restrictive

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since requirements for the Inservice Testing Program are being added to the Technical Specifications.

M03 ITS 5.5.2.15 contains two exceptions to the Regulatory Guide 1.163, "Performance – Based Containment Leak – Test Program." The first exception is that the visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC. The second exception is that the visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC. CTS 5.5.2.15, which provides the Containment Leakage Rate Testing Program requirements, does not contain these exceptions. This changes the CTS by adding two exceptions to Regulatory Guide 1.163 to the ITS.

The Technical Specification requirements for the Containment Leakage Rate Testing Program specify that the program shall be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Position C.3 of the regulatory guide states that "Section 9.2.1, Pretest Inspection and Test Methodology, of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." There are no specific requirements in NEI 94-01 for the visual examination except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.

In addition to the requirements of Regulatory Guide 1.163 and NEI 94-01, the concrete surfaces of the containment must be visually examined in accordance with the ASME Section XI Code, Subsection IWL, and the liner plate inside containment must be visually examined in accordance with Subsection IWE. The frequency of visual examination of the concrete surfaces per Subsection IWL is once every five years, and the frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year period. The visual examinations performed pursuant to Subsection IWL may be performed at any time during power operation or during shutdown, and the visual examinations performed pursuant to Subsection IWE are performed during refueling outages since this is the only time that the liner plate is fully accessible.

The visual examinations performed pursuant to Subsections IWL and IWE are more rigorous than those performed pursuant to Regulatory Guide 1.163 and NEI 94-01. This change is designated as more restrictive since exceptions to Regulatory Guide 1.163 are being added to the Technical Specifications.

M04 The CTS does not have a Surveillance Frequency Control Program. ITS 5.5.2.18 requires a program to satisfy the relocation of the Surveillance

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Frequency from the individual specifications. This changes the CTS by incorporating the requirements of ITS 5.5.2.18.

The NRC has been reviewing and granting improvements to the Improved Standard Technical Specifications (ISTS) based, at least in part, on probabilistic risk analysis insights. Typically, the proposed improvements involved a relaxation of one or more Completion Times or Surveillance Frequencies in the TS.

In August 1995, the NRC adopted a final policy statement on the use of probabilistic risk assessment (PRA) methods, which included the following regarding the expanded use of PRA.

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, licensee commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on need for proposing and backfitting new generic requirements on nuclear power plant licensees.

In its approval of the policy statement, the Commission articulated its expectation that implementation of the policy statement will improve the regulatory process in three areas: foremost, through safety decision-making enhanced by the use of PRA insights; through more efficient use of agency resources; and through a reduction in unnecessary burdens on licensees. This change is consistent with TSTF-425-A. TSTF- 425-A required that licensees who adopted this TSTF confirm that the plant PRA is consistent with Section 4.2 of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of probabilistic Risk Assessment results for Risk-Informed Activities." SCE has performed an assessment on the SONGS Units 2 and 3 PRA, and confirmed that it is consistent with the guidance in Section 4.2 of Regulatory Guide 1.200. Future models updates (internal model or external model) will be evaluated to

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determine any impact on the conclusions of the assessment that was performed in support of adopting this change. For each individual Surveillance Frequency relocation, see each of the associated Technical Specifications for the Discussion of Changes (DOC) justifying the individual relocations. This change is considered more restrictive since a new program is being added to the Technical Specifications.

M05 CTS 5.5.2.6.c requires identification of process sampling points for the Secondary Water Chemistry Program. ITS 5.5.2.6.c includes the same identification of sampling points requirement, but contains an additional statement that one of the sampling points will be the discharge of the condensate pumps to monitor for evidence of condenser inleakage. This changes the CTS by including the monitoring of the discharge of the condensate pumps for evidence of condenser inleakage.

The purpose of CTS and ITS 5.5.2.6 is to provide specific detail of what should be included in a program to monitor secondary water chemistry. The added statement is to identify a specific process sampling point to monitor for evidence of condenser inleakage. This change is acceptable because the additional detail provides assistance in development to ensure the intent of the program is met and will contribute to an effective means of monitoring the chemistry of the water. Furthermore, the current SONGS program already includes this sampling point. This change is designated as more restrictive because additional details are being added.

M06 CTS 5.5.2.15 states, in part, the calculated peak containment internal pressure related to the design basis loss of coolant accident, P_a is 48.0 psig. ITS 5.5.2.15.b contains a specific value for the calculated peak containment internal pressure related to the design basis loss of coolant accident and the containment design pressure. The containment design pressure is 60 psig. This changes the CTS by adding the containment design pressure.

This change is acceptable because the peak calculated containment internal pressure is used for the design basis loss of coolant accident. This change is designated as more restrictive because it imposes a new value that was not included in the CTS.

M07 CTS 5.5.2.12 requires testing of the Engineered Safety Feature of the Control Room Emergency Air Cleanup System ventilation filters. CTS 5.5.2.12.a, 5.5.2.12.b, and 5.5.2.12.d require the testing at the appropriate system flowrate. CTS 5.5.2.12.c requires laboratory testing of charcoal adsorber samples to show acceptable methyl iodide penetration. ITS 5.5.2.12 requires similar testing but specifies the actual penetration and system bypass, the actual flowrate, with a tolerance, and the actual methyl iodide penetration. Additionally, ITS 5.5.2.12.d specifies the Delta P to be used for testing the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers. This changes the CTS by specifying the actual values to be used in the testing process.

The purpose of the Engineered Safety Feature Control Room Emergency Air Cleanup System Ventilation Filter Testing Program is to verify the OPERABILITY of the ventilation filter trains. This change is acceptable because the values are

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required to perform the testing in accordance with the appropriate Regulatory Guide, ANSI standards, and ASTM standard. This change is designated as more restrictive because it imposes new values that were not included in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.5.2.1.1.a.3 requires licensee initiated changes to the ODCM to have documentation that the change has been reviewed and is acceptable. CTS 5.5.2.1.1.b requires licensee initiated changes to the ODCM to be reviewed and approved by the corporate officer with direct responsibility for the plant. ITS 5.5.2.1.1 does not specify that the review and acceptability of the changes be documented. ITS 5.5.2.1.1.b states that licensee initiated changes to the ODCM shall become affective after the approval of the corporate officer with direct responsibility for the plant. This changes the CTS by moving the record retention requirement references and the corporate officer's review to the Quality Assurance Program (QAP).

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.2.1.1 still retains the requirements for changes to the ODCM to be documented and retained. Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAP. Any changes to the QAP are made under 10 CFR 50.54(a), which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specifications requirements are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS requirement to the LCS, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 5.5.2.10, in part, provides requirements for the Inservice Inspection Program. The ITS does not include Inservice Inspection Program requirements. This changes the CTS by moving these requirements from the Technical Specifications to the Inservice Inspection (ISI) Program.

The removal of these requirements is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications still retain requirements for the affected components to be OPERABLE. Also, this change is acceptable because these requirements will be adequately controlled by the ISI Program, which is required by 10 CFR 50.55a. Compliance with 10 CFR 50.55a is required by SONGS Operating License. This

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change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.5.2.13.c requires total particulate testing of fuel oil every 92 days in accordance with ASTM D-2276, Method A. ITS 5.5.2.13.c requires the same particulate testing every 92 days but does not specify the ASTM requirement. This changes the CTS by moving the reference to the ASTM standard to the ITS 3.8.3 Bases.

One purpose of the diesel fuel oil program is to establish the total particulate concentration of fuel oil is ≤ 10 mg/l in accordance with ASTM D 2276 Method A-2 or A-3. This change is acceptable because the proposed change will provide the flexibility to maintain the capability to implement the required testing of both new fuel oil and stored fuel oil, including sampling and testing requirements, in accordance with applicable ASTM Standards whenever there are changes in Environmental Protection Agency (EPA) regulations for fuel oil or newer editions of the ASTM Standards. Furthermore, the actual ASTM Standard in ITS 3.8.3 Bases for the particulate test has been changed to ASTM D6217, as recommended by EPRI Guideline 1015061, Revision 3. Additionally, removing these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications will still retain the requirement for performing particulate testing of fuel every 92 days. Also, this change is acceptable because the removed information will be adequately controlled in the 3.8.3 Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This change is considered a less restrictive removal of detail change because a requirement is being removed from Technical Specifications.

LA04 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.6.3 contains a generic example of how to determine when a loss of safety function exists. ITS 5.5.2.14 does not contain this example. This changes the CTS by relocating the loss of safety function example to the ITS 3.0.6 Bases.

The purpose of the example is to help the user in determining when a loss of safety function exists. The ITS 3.0.6 Bases contains the example and provides more detail on how to perform cross system checks. Removing these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications will still retain the requirement for the Safety Function Determination Program (SFDP). Also, this change is acceptable because the removed information will be adequately controlled in the 3.0.6 Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This change is considered a less restrictive removal of detail change because a requirement is being removed from Technical Specifications.

LA05 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 5.5.2.1.1.b states Licensee initiated changes to

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the ODCM will become effective upon approval by the corporate officer with direct responsibility for the plant or his designee. ITS 5.5.2.1.1 provides the same requirements except Licensee initiated changes to the ODCM will become effective upon approval by the plant manager. This changes the CTS by moving the specific organizational title of the person whose signature is required to make changes to the ODCM effective, to the UFSAR and replacing it with a generic title.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairman, dated November 10, 1994. Furthermore, both CTS and ITS 5.2.1.a require the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications to be documented in the UFSAR. Also, this type of change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e). This change is designated as a less restrictive removal of detail change because information relating to meeting Technical Specification requirements is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 5.5.2.13 contains the requirements for the Diesel Fuel Oil Testing Program. ITS 5.5.2.13 provides the requirements for the Diesel Fuel Oil Testing Program and contains a statement that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program. This changes the CTS allowing the provisions of SR 3.0.2 and SR 3.0.3 to be applicable to the Diesel Fuel Oil Testing Program.

The purpose of the diesel fuel oil program is to ensure that the quality of the stored diesel fuel oil is acceptable so that the emergency diesel generators can perform their safety function. The addition of SR 3.0.2 is necessary to facilitate Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance. The 25% extension does not significantly degrade the reliability of Diesel Fuel Oil Testing Program. Additionally, since the requirements in the Diesel Fuel Oil Testing Program were moved from Technical Specification to the Program in a previous conversion from Pre-ITS Technical Specifications to the ITS, these requirements were allowed the 25% extension time. Therefore, it is acceptable to allow the 25% extension for the Diesel Fuel Oil Testing Program.

The addition of SR 3.0.3 is acceptable because it is based on Generic Letter (GL) 87-09. GL 87-09 addressed three specific issues with the application of Technical Specifications. One of those issues was missed Surveillances. It specifically addressed a problem with unnecessary shutdowns caused by when

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surveillance intervals are inadvertently exceeded. The solution was to clarify the applicability of the Action Requirements, to specify a specific acceptable time limit for completing a missed surveillance and to clarify when a missed surveillance constitutes a violation of the Operability Requirements of an LCO. It is overly conservative to assume that systems or components are inoperable when a surveillance requirement has not been performed because the vast majority of surveillances do in fact demonstrate that systems or components are OPERABLE. When a surveillance is missed, it is primarily a question of operability that has not been verified by the performance of a Surveillance Requirement. Because the allowable outage time limits of some Action Requirements do not provide an appropriate time for performing a missed surveillance before Shutdown Requirements apply, the TS should include a time limit that allows a delay of required actions to permit the performance of the missed surveillance based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and, of course, the safety significance of the delay in completing the surveillance. The NRC concluded in GL 87-09 that 24 hours is an acceptable time limit for completing a missed surveillance when the allowable outage times of the Action Requirements are less than this limit, or when time is needed to obtain a temporary waiver of the Surveillance Requirement. Therefore, allowing the provisions of SR 3.0.3 for diesel fuel oil testing is acceptable. This delay period will allow for the performance of the Surveillance.

This change is designed as less restrictive because an extension of the Surveillance Requirement Completion Time is allowed and additional time is allowed for a missed Surveillance Requirement.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals 5.5.1 Not Used 2

5.5.2 The following programs shall be established, implemented, and maintained. 2

5.5.2.1 5.5.~~1~~ 2. Offsite Dose Calculation Manual (ODCM) 2

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program ~~and~~ ; 3

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[5.6.1]~~ 5.7.1.2 and Specification ~~[5.6.2]~~ 5.7.1.3. 3 4

5.5.2.1.1 5.5.2.1.1 Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) ~~and~~ ; 3

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.~~1302~~ 106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations ; 18 3

b. Shall become effective after the approval of the plant manager ~~and~~ ; 3

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

5.5.2.8 5.5.~~2~~ 2.8 **Primary Coolant Sources Outside Containment** Program (2) (1)

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include INSERT 1 ~~[Low Pressure Injection, Reactor Building Spray, Makeup and Purification, and Hydrogen Recombiner]~~. The program shall include the following: (4)

a. Preventive maintenance and periodic visual inspection requirements ~~and~~ ; (3)

b. Integrated leak test requirements for each system at least once per 24 ~~[18]~~ months. (4)

The provisions of SR 3.0.2 are applicable to the Primary Coolant Sources Outside Containment Program test frequencies. (16)

~~5.5.3 [Post Accident Sampling~~

~~---REVIEWER'S NOTE---~~
~~This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000.~~
~~---~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:~~ (6)

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

5.5.2.3 5.5.~~4~~ 2.3 **Radioactive Effluent Controls Program** (2)

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

3 **INSERT 1**

high pressure safety injection recirculation, the Shutdown Cooling System, the Reactor Coolant Sampling System (post-accident sampling piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), the Containment Spray System, the Radioactive Waste Gas System (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), and the Liquid Radwaste System (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path)

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5.5 Programs and Manuals

5.5.2.3 5.5.~~4~~ 2.3 Radioactive Effluent Controls Program (continued) 2

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM~~,~~ ; 3

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~ten times~~ the concentration values in 10 CFR 20, Appendix B, Table ~~2~~ II, Column 2 ~~to 10 CFR 20.1001-20.2402~~~~,~~ ; 18 18 3

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.~~1302~~ 106 and with the methodology and parameters in the ODCM~~,~~ ; 18 3

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I~~,~~ ; 3

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days~~,~~ ; 3

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I~~,~~ ; 3

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with ~~the following:~~ 10 CFR 20, Appendix B, Table II, Column 1; 18 3

~~1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and~~

~~2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ,~~ 2

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I~~,~~ ; 3

5.5 Programs and Manuals

5.5.2.3 5.5.~~4~~ 2.3 Radioactive Effluent Controls Program (continued) 2

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, ~~and~~ ; 3

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.2.4 5.5.~~5~~ 2.4 Component Cyclic or Transient Limit Program 2 1

This program provides controls to track the U FSAR~~, Section [ ],~~ Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits. 4

5.5.2.9 5.5.~~6~~ 2.9 ~~[~~ Pre-Stressed Concrete Containment Tendon Surveillance Program 2

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Pre-Stressed Concrete Containment Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC. 17

The provisions of SR 3.0.3 are applicable to the Pre-Stressed Concrete Containment Tendon Surveillance Program inspection frequencies. ~~]~~ 17 4

5.5.2.5 5.5.~~7~~ 2.5 Reactor Coolant Pump Flywheel Inspection Program 2

This program shall provide for the inspection of each reactor coolant pump flywheel ~~per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975~~ INSERT 2. 7

5.5.2.10 5.5.~~8~~ 2.10 Inservice Testing Program 2

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

7 **INSERT 2**

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the flywheels each 10 years.

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5.5 Programs and Manuals

5.5.2.10 5.5.~~8~~ 2.10 Inservice Testing Program (continued) 2

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

as 2 years or less

to

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified in the Inservice Testing Program for performing inservice testing activities~~.~~ ; TSTF-497-A 3

c. The provisions of SR 3.0.3 are applicable to inservice testing activities~~,~~ ; and 3

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.2.11 5.5.~~9~~ 2.11 Steam Generator (SG) Program 2

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, ~~[or repaired]~~ to confirm that the performance criteria are being met. 4 4

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

5.5 Programs and Manuals

5.5.2.11 5.5.~~9~~ 2.11 Steam Generator (SG) Program (continued) 2

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm ~~[1 gpm]~~ per SG ~~[, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program]~~ and 1 gpm through both SGs. 4

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

1.

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding ~~[40%]~~ 35%. of the nominal tube wall thickness shall be plugged ~~[or repaired]~~. 4

~~-------------------------------------REVIEWER'S NOTE---~~
~~Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.~~
~~---~~ 8

5.5 Programs and Manuals

5.5.2.11 5.5.~~9~~ 2.11 Steam Generator (SG) Program (continued) 2

[The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria: 4

1. . . .]

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

------------------------------------REVIEWER'S NOTE---
Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing. 8
----

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

[2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.]

[2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.] 4

5.5 Programs and Manuals

5.5.2.11 5.5.~~9~~ 2.11 Steam Generator (SG) Program (continued) 2

~~[~~2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~]~~ 4

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. 8

e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~[f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~-------------------------------------REVIEWER'S NOTE--~~ 4

~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~ 8

~~--~~

~~1. . . .]~~

5.5.2.6 5.5.~~10~~ 2.6 Secondary Water Chemistry Program 2

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables~~.~~ ; 3

5.5 Programs and Manuals

5.5.2.6 5.5.~~10~~ 2.6 Secondary Water Chemistry Program (continued) 2

b. Identification of the procedures used to measure the values of the critical variables~~,~~ ; 3

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser ~~in leakage~~ inleakage~~,~~ ;

d. Procedures for the recording and management of data~~,~~ ; 3

e. Procedures defining corrective actions for all off - control point chemistry conditions~~,~~ ; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.2.12 5.5.~~11~~ 2.12 Ventilation Filter Testing Program (VFTP) 2

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in ~~[Regulatory Guide],~~ and in accordance with ~~[~~Regulatory Guide 1.52, Revision 2~~, ASME N510-1989, and AG-1]~~ at the system flowrate specified below ~~[~~± 10%~~]~~. 4

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < ~~[~~0.05~~]~~% when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-~~1989]~~ 1975 unless manifolds have been qualified per ASME N510-1989 at the system flowrate specified below ~~[~~± 10%~~]~~. 4

ESF Ventilation System	Flowrate
~~[ ]~~ Control Room Emergency Air Cleanup System	~~[ ]~~ 35,705 cfm

4

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < ~~[~~0.05~~]~~% when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-~~1989]~~ 1975 unless manifolds have been qualified per ASME N510-1989 at the system flowrate specified below ~~[~~± 10%~~]~~. 4

ESF Ventilation System	Flowrate
~~[ ]~~ Control Room Emergency Air Cleanup System	~~[ ]~~ 35,705 cfm

4

5.5 Programs and Manuals

5.5.2.12 5.5.~~11~~ 2.12 Ventilation Filter Testing Program (continued) 2

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ~~[~~Regulatory Guide 1.52, Revision 2~~]~~, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below. 4

ESF Ventilation System	Penetration	RH	~~Face Velocity~~
~~[ ]~~ Control Room Emergency Air Cleanup System	~~[See Reviewer's Note]~~ < 1%	~~[See Reviewer's Note]~~ 70%	~~[See Reviewer's Note]~~

15

~~-----REVIEWER'S NOTE-----~~
~~The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.~~

~~ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]~~

~~When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:~~ 15

~~Safety factor ≥ 2 for systems with or without humidity control.~~

~~Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst case design basis conditions.~~

~~If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.~~

~~*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.~~
~~-----~~

5.5 Programs and Manuals

5.5.2.12 5.5.~~11~~ 2.12 Ventilation Filter Testing Program (continued) 2

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2~~, and ASME N510-1989]~~ at the system flowrate specified below ~~[~~± 10%~~]~~. 4

ESF Ventilation System	Delta P	Flowrate
~~[ ]~~ Control Room Emergency Air Cleanup System	~~[ ]~~ 8.3 inches of water	~~[ ]~~ 35,705 cfm

4

~~[e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].~~ 4

~~ESF Ventilation System~~	~~Wattage]~~
~~[ ]~~	~~[ ]~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.2.7 5.5.~~12~~ 2.7 Explosive Gas and Storage Tank Radioactivity Monitoring Program 2

This program provides controls for potentially explosive gas mixtures contained in the ~~[Waste Gas Holdup~~ Gaseous Radwaste System~~]~~, ~~[~~the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system,~~ and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks~~]~~. The gaseous radioactivity quantities shall be determined following the methodology in ~~[~~Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"~~]~~. The liquid radwaste quantities shall be determined in accordance with ~~[~~Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"~~]~~. 4 4

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup~~ Gaseous Radwaste System~~]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion)~~.~~ ; 4 3

5.5 Programs and Manuals

5.5.2.7 5.5.~~12~~ 2.7 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued) 2

b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~each ~~gas storage~~ waste gas decay tank and fed into the ~~offgas treatment~~ gaseous radwaste vent system~~]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~an uncontrolled release of the tanks' contents~~]~~~~,~~ ; and 4 4 4 3

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the ~~[~~Liquid ~~Radwaste Treatment~~ Waste Management System~~]~~ is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 4

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.2.13 5.5.~~1~~ 2.13 Diesel Fuel Oil Testing Program 2

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

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

1. An API gravity or an absolute specific gravity within limits~~,~~ ; 3

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil~~,~~ ; and 3

3. A ~~clear and bright appearance with proper color or a~~ water and sediment content within limits~~,~~ ; 9 3

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil~~,~~ ; and 3

5.5 Programs and Manuals

5.5.2.13 5.5.~~13~~ 2.13 Diesel Fuel Oil Testing Program (continued) 2

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every ~~31~~ 92 days. 1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.4 5.5.~~14~~ 2.2 Technical Specifications (TS) Bases Control Program 2

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license ; or 3

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 1

d. Proposed changes that meet the criteria of ~~5.5.14b~~ 5.5.2.2.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC ~~on a frequency consistent with 10 CFR 50.71(e)~~ INSERT 3. 2 5

5.6 5.5.~~15~~ 2.14 Safety Function Determination Program (SFDP) 2

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. a. The SFDP shall contain the following: 10

~~a~~ 1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected~~.~~ ; 10 3

5 **INSERT 3**

within 6 months following every Unit 3 refueling, not to exceed 24 months. This schedule is consistent with SCE's submittal of UFSAR updates as allowed by the NRC approved exemption from 10 CFR 50.71(e) dated April 27, 1999.

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5.5 Programs and Manuals

5.6 5.5.~~15~~ 2.14 Safety Function Determining Program (continued) 2

~~b~~ 2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists ; 10 3

~~c~~ 3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities ; and 10 3

~~d~~ 4. Other appropriate limitations and remedial or compensatory actions. 10

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable ; and 10 3

~~a~~ 1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable ; or 10 3

~~b~~ 2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable ; or 10 3

~~c~~ 3. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable. 10

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system. 10

5.5.2.15 5.5.~~16~~ 2.15 Containment Leakage Rate Testing Program 2

~~[OPTION A]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~

~~b. The maximum allowable containment leakage rate, L_a at P_a, shall be [ ]% of containment air weight per day.~~ 11

~~c. Leakage rate acceptance criteria are:~~

5.5 Programs and Manuals

5.5.2.15 5.5.~~16~~ 2.15 Containment Leakage Rate Testing Program (continued) 2

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and < 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.

   b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig]. 11

d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

~~[OPTION B]~~ 11

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

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

~~[3. . . .]~~

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is ~~[45~~ 48 psig~~]~~ INSERT 4. The containment design pressure is ~~[50~~ 60 psig~~]~~. 4 12

12 **INSERT 4**

(P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification)

5.5 Programs and Manuals

5.5.2.15 5.5.~~16~~ 2.15 Containment Leakage Rate Testing Program (continued) 2

c. The maximum allowable containment leakage rate, L_a at P_a, shall be ~~[ ]~~ 0.10 % of containment air weight per day. 4

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ~~<~~ $\leq$ 0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests. 1

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq$ ~~[~~0.05 L_a~~]~~ when tested at $\geq$ P_a.

b) For each door, leakage rate is $\leq$ ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~$\geq$ ~~10~~ 9.0 psig~~]~~. 4

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. INSERT 5 ~~Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~ 13

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C] [Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~ 11

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

13 **INSERT 5**

Test frequencies specified in this Program may be extended consistent with the guidance provided in NEI 94-01, "Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has these provisions for test frequencies extension:

1. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

2. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for the recommended surveillance frequency for Type B and Type C testing may be extended by up to 25 percent of the test interval, not to exceed 15 months.

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5.5 Programs and Manuals

5.5.2.15 5.5.~~16~~ 2.15 Containment Leakage Rate Testing Program (continued) 2

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is [45 psig]. The containment design pressure is [50 psig].

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

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and [$< 0.75\ L_a$ for Option A Type A tests] [$\leq 0.75\ L_a$ for Option B Type A tests]. 11

2. Air lock testing acceptance criteria are:

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

b) For each door, leakage rate is $\leq$ [$0.01\ L_a$] when pressurized to [$\geq$ 10 psig].

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

INSERT 6 14

14 **INSERT 6**

5.5.2.16 5.5.2.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACUS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

   The following is exception to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

      Appropriate application of ASTM E-741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACUS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

U2/U3 CTS

14 **INSERT 6 (Continued)**

5.5.2.16 f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

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5.5 Programs and Manuals

5.5.2.17 5.5.~~17~~ 2.17 Battery Monitoring and Maintenance Program 2

This Program provides for battery restoration and maintenance, ~~based on [the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer] including~~ which includes the following : 1 3

a. Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V~~, and~~ ; 4 3

INSERT 7

INSERT 8

~~b~~ c. Actions to equalize and test battery cells that had been discovered with electrolyte level below the ~~minimum established design limit~~ top of the plates. 1 2 TSTF-425-A

5.5

U2/U3 CTS

1

INSERT 7

b. Actions to verify that the remaining cells are above 2.07 V when a battery cell or cells have been found less than 2.13V; and

TSTF-425-A

INSERT 8

DOC M04

5.5.18 Surveillance Frequency Control Program

2.

2

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI-04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Specification number has been changed to be consistent with the Specification number in the SONGS CTS. SCE has decided not to renumber the CTS to be consistent with the ISTS because by doing so would result in the unnecessary administrative burden of changing TS numbers in plant procedures.

3. Changes are made to use correct punctuation, correct typographical errors or to make corrections consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01.

4. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

5. ISTS 5.5.14 (ITS 5.5.2.2) requires that changes to the Bases which are implemented without prior NRC approval, shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). CTS 5.4.4 requires these same types of changes be provided to the NRC within 6 months following every Unit 3 refueling outage. Therefore, ITS 5.5.2.2 has been changed to match the CTS schedule for providing these changes. This change is acceptable because the NRC approved an exemption from 10 CFR 50.71(e) for SONGS on April 27, 1999.

6. The Post Accident Sampling Program has been deleted from the SONGS Units 2 and 3 CTS as documented in Amendments 178 and 169, respectively, dated 3/26/2001 (ADAMS Accession No. ML010870439).

7. ISTS 5.5.7 (ITS 5.5.2.5) requires the Reactor Coolant Pump Flywheel Inspection Program to provide inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975. SONGS currently has a license commitment to Regulatory Guide 1.14 but does not specifically state in the Reactor Coolant Pump Flywheel Inspection Program that it is committed to the Regulatory Guide. On 9/8/2000 San Onofre was issued a license amendment for Unit 2 and Unit 3 (Amendment 170 and 168, respectively) to change the Reactor Coolant Pump Flywheel Inspection Program volumetric frequency of the upper flywheel on each of the primary reactor coolant pump motors from a 3 year to a 10 year cycle (Adams Accession Number ML003748732). In the safety evaluation for these amendments, the NRC concluded that SONGS meets the intent of Regulatory Guide 1.14. Therefore, SONGS will maintain their current licensing bases for the Reactor Coolant Pump Flywheel Inspection Program.

8. The Reviewers Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

9. The clear and bright appearance test with proper color has been deleted from ISTS 5.5.13 (ITS 5.5.2.13) consistent with current practice. The clear and bright test

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

is only applicable to fuels that meet the ASTM D4176 color rating requirements and is considered a qualitative test. ISTS TSTF-374-A revised the testing requirements of ISTS 5.5.13 to include an additional method of testing for water and sediment testing. The water and sediment test is performed using acceptable ASTM standards for performing this qualitative test. Currently SONGS only performs the water and sediment testing. Therefore, SONGS will maintain their current licensing bases and not include clear and bright testing.

10. This Specification has been renumbered to be consistent with the ITS Format and for clarity.

11. SONGS complies with Option B of 10 CFR 50, Appendix J. Therefore, the ISTS 5.5.16 Option A and combined Option A and B provisions have been deleted.

12. ISTS 5.5.16.b (ITS 5.5.2.15.b) contains a statement on the design pressure limit that states that P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification. ITS 5.5.2.15.b will retain this current licensing requirement.

13. ISTS 5.5.16.f (ITS 5.5.2.15.f) states that nothing in Technical Specification shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J. CTS 5.5.2.15 states, in part, that the test frequencies in Containment Leakage Rate Testing Program may be extended consistent with the guidance provided in NEI 94-01 as endorsed by Regulatory Guide 1.163. It then gives an extension for Type A testing and an extension for Type B and C testing. ITS 5.5.2.15.f will maintain the current license extensions allowed by CTS 5.5.2.15.

14. ITS 5.5.2.16, "Control Room Envelope Habitability Program" has been added consistent with current Technical Specification 5.5.2.16 and as described in TSTF-448.

15. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the Ventilation Filter Testing Program specific information/value is revised to reflect the current Technical Specifications. Additionally, the face velocity has not been included in SONGS ITS 5.5.2.12. The Reviewer's Note for ISTS 5.5.11 (ITS 5.5.2.12) only requires the face velocity to be specified if any standard other than ASTM D3803-1989 is used and the face velocity is greater than 110 percent of 0.203 m/s (40 ft/min). Since SONGS uses ASTM D3803-1989 and has a face velocity of 40 ft/min, this face velocity value is not included in the SONGS ITS.

16. The last sentence of ISTS 5.5.2 (ITS 5.5.2.8) has been modified to include "to the Primary Coolant Sources Outside Containment Program test frequencies," consistent with all the other program statements that allow the provisions of SR 3.0.2 to be applicable.

17. The complete Program title has been provided, consistent with the actual program title in ISTS 5.5.6.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

18. ISTS 5.5.1.a.2, ISTS 5.5.4.b, and ISTS 5.5.4.c use the new 10 CFR Part 20 subsections. In addition, ISTS 5.5.4.g also use the new 10 CFR Part 20 values. ITS 5.5.2.1.1.a.2, ITS 5.5.2.3.b, and ITS 5.5.2.3.c use the previous (pre-1994) 10 CFR 20 subsections and ITS 5.5.2.3 g uses the pre-1994 values for gaseous effluents released from the site. These values and 10 CFR Part 20 subsection numbers are consistent with the CTS. This is allowed by 10 CFR 20.1008, which states that the pre-1994 values and requirements can be used if they are more restrictive than the current 10 CFR 20 values.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

There are no specific NSHC discussions for this Specification.

ATTACHMENT 6

ITS 5.7, REPORTING REQUIREMENTS

**Current Technical Specification (CTS) Markup
and Discussion of Changes (DOCs)**

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 5.7.1 ~~Routine Reports~~

~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations,~~ the following reports shall be submitted in accordance with 10 CFR 50.4. ~~The reports shall be addressed to the U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.~~ A02

Not Used

5.7.1.1 5.7.1.1 ~~Annual Reports~~ L01

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~
~~----------~~

~~Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by March 31 of each year.~~

~~Reports required on an annual basis include:~~

~~a. (Deleted)~~

(continued)

~~5.7.1.1 Annual Reports (continued)~~

L01

~~b. Reactor Coolant System Specific Activity Report~~

~~Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.16. The following information shall be included in these reports:~~

~~1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~2. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; and~~

~~3. Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and~~

~~5. The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

5.7.1.2 5.7.1.2 Annual Radiological Environmental Operating Report

-------------------------------NOTE-------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the

(continued)

5.7.1.2 5.7.1.2 Annual Radiological Environmental Operating Report (continued)

objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. ~~The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. LA01

5.7.1.3 5.7.1.3 Radioactive Effluent Release Report

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

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the unit. The report shall also include a summary of the quantities of solid radioactive waste shipped from the unit ~~directly to the disposal site and quantities of solid radioactive waste shipped from the unit's intermediary processor to the disposal site~~. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1. A06 LA02

(continued)

ITS A01

5.7 Reporting Requirements (continued)

5.7.1.4 5.7.1.4 (Deleted)

5.7.1.5 5.7.1.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. LCO ~~Specification~~ 3.1.1, "SHUTDOWN MARGIN (SDM) ~~– T_{avg} >200°F;~~" A03 A04

~~2. Specification 3.1.2, "SHUTDOWN MARGIN (SDM) – T_{avg} ≤200°F;"~~

3. LCO ~~Specification~~ 3.1.4, "Moderator Temperature Coefficient (MTC);"

4. LCO ~~Specification~~ 3.1.5, "Control Element Assembly (CEA) Alignment;"

5. LCO ~~Specification~~ 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits;"

6. LCO ~~Specification~~ 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits;"

7. LCO ~~Specification~~ 3.2.1, "Linear Heat Rate (LHR);"

8. LCO ~~Specification~~ 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR);"

9. LCO ~~Specification~~ 3.2.5, "Axial Shape Index (ASI);"

10. LCO ~~Specification~~ 3.4.1, "RCS ~~DNB (~~Pressure, Temperature, and Flow ~~)~~ Departure from Nucleate Boiling (DNB) Limits;

11. LCO ~~Specification~~ 3.9.1, "Boron Concentration."

A03

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

(continued)

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5.7.1.5 5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

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

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

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

8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"

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

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

5.7.1.6 5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

5.7.1.6.a a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

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A01

ITS

5.7.1.6 5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

LCO ~~Technical Specification~~ 3.4.3 RCS Pressure and Temperature (P/T) Limits,

A03

LCO ~~Technical Specification~~ 3.4.6 RCS Loops - MODE 4,

LCO ~~Technical Specification~~ 3.4.7 RCS Loops - MODE 5, Loops Filled,

LCO ~~Technical Specification~~ 3.4.12~~.1~~ Low Temperature Overpressure Protection (LTOP) System ~~RCS Temperature ≤ PTLR Limit~~, and

A04

3.4.10, Pressurizer Safety Valves

LCO ~~Technical Specification 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System RCS Temperature > PTLR Limit~~.

5.7.1.6.b b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

CE NPSD-683-A, The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications.

5.7.1.6.c c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

~~5.7.1.7 Hazardous Cargo Traffic Report~~

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

L02

(continued)

5.7.2 5.7.2 Special Reports

~~Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~ A01

The following reports ~~Special Reports~~ shall be submitted ~~to the U. S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC,~~ in accordance with 10 CFR 50.4 ~~within the time period specified for each report~~. A02

~~The following Special Reports shall be submitted:~~

5.7.2.a a. Post Accident Monitoring Report

When ~~a pre-planned alternate method of monitoring post-accident instrumentation functions~~ a report is required by Condition B or Condition ~~G~~ F of LCO 3.3.11, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within ~~30~~ the following 14 days ~~from the time the action is required~~. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to OPERABLE status. A05 M01

5.7.2.b b. Tendon Surveillance Report

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

5.7.2.c c. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.2.11, Steam Generator (SG) Program. The report shall include:

(continued)

~~5.7.2 Special Reports (continued)~~

5.7.2.c

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5. Number of tubes plugged during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged to date,

7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

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5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 5.7.1 ~~Routine Reports~~

~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations,~~ the following reports shall be submitted in accordance with 10 CFR 50.4. ~~The reports shall be addressed to the U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.~~ A02

Not Used

5.7.1.1 5.7.1.1 ~~Annual Reports~~ L01

~~NOTE~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

~~Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by March 31 of each year.~~

~~Reports required on an annual basis include:~~

~~a. (Deleted)~~

(continued)

~~5.7.1.1 Annual Reports (continued)~~

L01

~~b. Reactor Coolant System Specific Activity Report~~

~~Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.16. The following information shall be included in these reports:~~

~~1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~2. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; and~~

~~3. Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and~~

~~5. The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

5.7.1.2 5.7.1.2 Annual Radiological Environmental Operating Report

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the

(continued)

5.7.1.2 5.7.1.2 Annual Radiological Environmental Operating Report (continued)

objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. ~~The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. LA01

5.7.1.3 5.7.1.3 Radioactive Effluent Release Report

-------------------------------NOTE-------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.
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in accordance with 10 CFR 50.36a

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the unit. The report shall also include a summary of the quantities of solid radioactive waste shipped from the unit ~~directly to the disposal site and quantities of solid radioactive waste shipped from the unit's intermediary processor to the disposal site~~. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1. A06 LA02

(continued)

5.7 Reporting Requirements (continued)

5.7.1.4 5.7.1.4 (Deleted)

5.7.1.5 5.7.1.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. LCO ~~Specification~~ 3.1.1, "SHUTDOWN MARGIN (SDM) ~~- T_{avg} >200°F;~~" A03 A04

~~2. Specification 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤200°F;"~~

3. LCO ~~Specification~~ 3.1.4, "Moderator Temperature Coefficient (MTC);"

4. LCO ~~Specification~~ 3.1.5, "Control Element Assembly (CEA) Alignment;"

5. LCO ~~Specification~~ 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits;"

6. LCO ~~Specification~~ 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits;"

7. LCO ~~Specification~~ 3.2.1, "Linear Heat Rate (LHR);"

8. LCO ~~Specification~~ 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR);"

9. LCO ~~Specification~~ 3.2.5, "Axial Shape Index (ASI);"

10. LCO ~~Specification~~ 3.4.1, "RCS ~~DNB (~~Pressure, Temperature, and Flow ~~)~~ Departure from Nucleate Boiling (DNB) Limits;

11. LCO ~~Specification~~ 3.9.1, "Boron Concentration."

A03

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

(continued)

5.7.1.5 5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

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

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

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

8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"

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

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

5.7.1.6 5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

5.7.1.6.a a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

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5.7.1.6 5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

LCO ~~Technical Specification~~ 3.4.3 RCS Pressure and Temperature (P/T) Limits,

A03

LCO ~~Technical Specification~~ 3.4.6 RCS Loops - MODE 4,

LCO ~~Technical Specification~~ 3.4.7 RCS Loops - MODE 5, Loops Filled,

LCO ~~Technical Specification~~ 3.4.12~~.1~~ Low Temperature Overpressure Protection (LTOP) System ~~RCS Temperature ≤ PTLR Limit,~~ and

A04

LCO ~~Technical Specification 3.4.12.2 Low Temperature Overpressure~~

3.4.10, Pressurizer Safety Valves ~~Protection (LTOP) System RCS Temperature > PTLR Limit~~.

5.7.1.6.b b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

CE NPSD-683-A, The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications.

5.7.1.6.c c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

~~5.7.1.7 Hazardous Cargo Traffic Report~~

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

L02

(continued)

5.7.2 5.7.2 Special Reports

~~Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~ A01

The following reports

~~Special Reports~~ shall be submitted ~~to the U. S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC,~~ in accordance with 10 CFR 50.4 ~~within the time period specified for each report~~. A02

~~The following Special Reports shall be submitted:~~

Post Accident Monitoring Report

a report

5.7.2.a a. When ~~a pre-planned alternate method of monitoring post-accident instrumentation functions~~ is required by Condition B or Condition ~~G~~ of LCO 3.3.11, a report shall be submitted within ~~30~~ days ~~from the time the action is required~~. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to OPERABLE status. A05 M01

"Post Accident Monitoring (PAM) Instrumentation,"

F

the following 14

Tendon Surveillance Report

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

Steam Generator Tube Inspection Report

5.7.2.c c. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.2.11, Steam Generator (SG) Program. The report shall include:

(continued)

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5.7 Reporting Requirements (continued)

~~5.7.2 Special Reports (continued)~~

5.7.2.c

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5. Number of tubes plugged during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged to date,

7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

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DISCUSSION OF CHANGES
ITS 5.7, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 5.7.1 and 5.7.2 require, in addition to the requirements of 10 CFR, that the Routine and Special Reports be submitted to the US Nuclear Regulatory Commission document control desk with a copy be sent to Regional Office. ITS 5.7.1 and 5.7.2 require that the reports be submitted in accordance with 10 CFR 50.4. This changes the CTS by removing the explicit requirements to send the reports to Document Control Desk and the Regional Office.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.4. SONGS Units 2 and 3 already are required to meet the requirements of 10 CFR 50.4, since it is a regulation. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 5.7.1.5.a identifies the Technical Specifications that are required to be documented in the COLR. In the identification, the word "Specification" is used. ITS 5.7.1.5.a requires similar Technical Specifications to be identified in the COLR, but uses the term "LCO." CTS 5.7.1.6.a identifies the Technical Specifications that are required to be documented in the PTLR. In the identification, the words "Technical Specifications" are used. ITS 5.7.1.6.a requires similar Technical Specifications to be identified in the PTLR, but uses the term "LCO." Furthermore, the titles used in the CTS have been modified to meet the titles used in the ITS. This changes the CTS by using the term "LCO" in place of "Specification" or "Technical Specification" and changes the titles of the LCOs to match the ITS title.

This change is acceptable because the term specification and LCO can be interchangeable. The words were changed to match the ITS format. Additionally, title changes are minor editorial changes. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 5.7.1.5.a states, in part, that the core operating limits shall be documented in the COLR for Specification 3.1.1, "Shutdown Margin (SDM) – T_{avg} > 200°F and Specification 3.1.2, "Shutdown Margin (SDM) – T_{avg} ≤ 200°F. ITS 5.7.1.5.a requires the same documentation but only lists LCO 3.1.1, "SHUTDOWN MARGIN." CTS 5.7.1.6.a states, in part, that the pressure temperature limits shall be documented in the PTLR for Technical Specifications 3.4.12.1, "Low Temperature Overpressure Protection (LTOP) System RCS Temperature ≤

DISCUSSION OF CHANGES
ITS 5.7, REPORTING REQUIREMENTS

PTLR Limit," and 3.4.12.2, Low Temperature Overpressure Protection (LTOP) System RCS Temperature > PTLR Limit." ITS 5.7.1.6.a requries the same documentation but lists LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," and LCO 3.4.10, "Pressurizer Safety Valves." This changes the CTS by not specifying CTS 3.1.2, changing the title of LCO 3.4.12.1, and changing the LCO number and title of LCO 3.4.12.2.

This change is acceptable because a) CTS 3.1.1 has been combined with CTS 3.1.2 in ITS 3.1.1, "SHUTDOWN MARGIN"; b) CTS 3.4.12.1 has been renumbered as ITS 3.4.12 and the title changed; and c) CTS 3.4.12.2 has been incorporated into ITS 3.4.10. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 5.7.2.a states, in part, that when a preplanned alternative method of monitoring post-accident instrumentation functions is required by Condition B or Condition G of LCO 3.3.11, a report shall be submitted. ITS 5.7.2.a states, in part, that when a report is required by Condition B or Condition F of LCO 3.3.11, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted. This changes the CTS by editorially re-writing the Post Accident Monitoring Report requirement and changing the Condition G of LCO 3.3.11.

This change is acceptable because it is an editorial re-write. Additionally, the CTS Condition G has been changed to Condition F to match changes made to ITS LCO 3.3.11. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 CTS 5.7.1.3 requires a Radioactive Effluent Release Report to be submitted to the NRC each year. The CTS also, in part, states that the material provided in the report is in conformance with 10 CFR 50.36a. ITS 5.7.1.3 requires the same report, but includes an additional statement that it be submitted in accordance with 10 CFR 50.36a. This changes the CTS by clearly stating in the first sentence of the program that not only is the material provided in the report in conformance with 10 CFR 50.36a, but it is also submitted in accordance with 10 CFR 50.36a.

This change is editorial and acceptable since it is only clarifying that not only is the report's material in conformance with 10 CFR 50.36a, but it is also submitted consistent with the 10 CFR 50.36a. This change is also acceptable since the reporting requirement is located in 10 CFR 50.36a, and SCE is required to meet 10 CFR 50.36a. This change is designated as administrative because no technical changes are being made to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 5.7.2.a requires, in part, that when a pre-planned alternate method of monitoring post accident instrumentation functions is required by Condition B or G of LCO 3.3.11, then a report shall be submitted with 30 days from the time the action is required. ITS 5.7.2.a requires that when a report is required by Condition B or F (changed due to change to ITS 3.3.11), the post accident monitoring report shall be submitted within the following 14 days. This changes

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DISCUSSION OF CHANGES
ITS 5.7, REPORTING REQUIREMENTS

the CTS by requiring the report, required by Condition B or F of ITS 3.3.11, to be submitted within 14 days instead of within 30 days.

The purpose of the report is to outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrument channels of the function to OPERABLE status. This change is acceptable since it decreases the amount of time allowed to prepare the report and submit it to the NRC. This change is more restrictive because less time is given to prepare the Post Accident Monitoring Report.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the LCS, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 5.7.1.2 requires, in part, that the Annual Radiological Environmental Operating Report identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. ITS 5.7.1.2 does not contain this information. This changes the CTS by moving to the ODCM the requirement to identify the thermoluminescent dosimeter (TLD) results.

The removal of the requirement to identify the thermoluminescent dosimeter (TLD) results is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still maintains the requirement for Annual Radiological Environmental Operating Report to be consistent with the ODCM and 10 CFR Appendix I, Sections IV.B.2, IV.B.3, and IV.C. Therefore, this change is acceptable because the removed requirement to identify the thermoluminescent dosimeter (TLD) results will be adequately controlled in the ODCM. Changes to the ODCM are controlled by the ODCM change control process in ITS 5.5.2.1, which ensures changes are properly evaluated. This change is designated as less restrictive removal of detail change because a Technical Specification Requirement is being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the LCS, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 5.7.1.3 requires, in part, that the Radioactive Effluent Release Report include the summary of the quantities of solid radioactive waste shipped from the unit directly to the disposal site and quantities of solid radioactive waste shipped from the unit's intermediary processor to the disposal site. ITS 5.7.1.3 only states, in part, that the report shall include a summary of the quantity of solid radioactive waste released from the unit. This changes the CTS by moving the clarifying information of the two types of solid

DISCUSSION OF CHANGES
ITS 5.7, REPORTING REQUIREMENTS

radioactive waste shipments (i.e., directly to the disposal site and to the disposal site via an intermediary processor) to the Licensee Controlled Specifications (LCS).

The removal of the requirement to identify the two types of shipment is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still maintains the requirement for the Radioactive Effluent Release Report include the summary of the quantity of solid radioactive waste released from the unit (which includes both what is shipped directly to the disposal site and what is shipped to the intermediary processor and then on to the disposal site) and that the material be consistent with the objectives outlined in the ODCM and Process Control Program. The Process Control Program has previously been removed from the Technical Specifications and placed in the SONGS LCS. Therefore, this change is acceptable because the removed requirement will be adequately controlled in the LCS. The LCS is currently incorporated by reference into the UFSAR, thus any changes to the LCS are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as less restrictive removal of detail change because a Technical Specification Requirement is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 8 – Deletion of Reporting Requirements)* CTS 5.7.1.1.b requires annual reporting of information regarding any instances of when the specific activity limit for the primary coolant is exceeded. ITS 5.7 does not contain any requirements for such a report. This changes the CTS by not including the requirements for the annual reporting of instances when the Technical Specification specific activity limit for the primary coolant is exceeded.
The purpose of CTS 5.7.1.1.b is to specify the requirement for submitting information regarding any instances when the Technical Specification specific activity limit for the primary coolant is exceeded in an annual report. This change is acceptable because the regulations provide adequate details of reporting requirements, and the reporting of exceeding the specific activity limit does not affect continued plant operation. Operations or conditions prohibited by the plant's Technical Specifications are required to be reported in accordance with 10 CFR 50.73. Subsequent reports would be provided if necessary, with requiring a specific annual report. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L02 *(Category 8 – Deletion of Reporting Requirements)* CTS 5.7.1.7 requires a Hazardous Cargo Traffic Report to be submitted to the NRC Regional Administrator every three years. ITS 5.7.1 does not contain this report. This changes the CTS by not including the requirement for a Hazardous Cargo Traffic Report.

The purpose of CTS 5.7.1.7 is to monitor the hazardous cargo traffic on Interstate 5 and the AT&SF railway and to notify the NRC every three years of

DISCUSSION OF CHANGES
ITS 5.7, REPORTING REQUIREMENTS

the results. This report is based on the requirements of Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release." This change is acceptable because Technical Specifications do not require this type of report. Furthermore, this report will continue to be issued to the NRC outside of the Technical Specifications. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.~~6~~ 7 Reporting Requirements

5.7.1 5.7.1 The following reports shall be submitted in accordance with 10 CFR 50.4.

5.7.1.1 Not used.

5.7.1.2 5.~~6~~.1 .2 7 Annual Radiological Environmental Operating Report

-----NOTE-----
[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.] 2

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 2 2

5.7.1.3 5.~~6.2~~ 7.1.3 Radiological Effluent Release Report

-----NOTE-----
[A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.] 2 2

The Radioactive Effluent Release Report covering the operation of the unit in the previous calendar year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1. 7

5.7.1.4 Not Used

5.~~6~~ 7 Reporting Requirements

5.7.1.5 5.~~6.3~~ 7.1.5 CORE OPERATING LIMITS REPORT (COLR)

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

INSERT 1 → ~~[The individual specifications that address core operating limits must be referenced here.]~~ 2

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

INSERT 2 → ~~[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).]~~ 2

5.7.1.5.c c. The core operating limits shall be determined assuming operation at RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. 6

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

5.7.1.6 5.~~6.4~~ 7.1.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

5.7.1.6.a a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

INSERT 3 → ~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~ 2

5.7.1.6.b b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document~~s~~: 2

INSERT 4 → ~~[Identify the Topical Report(s) by number and title or identify the NRC Safety Evaluation for a plant specific methodology by NRC letter and date. The PTLR will contain the complete identification for each of the TS referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements).]~~ 2

2 INSERT 1

1. LCO 3.1.1, "SHUTDOWN MARGIN" (SDM);"
2. LCO 3.1.4, "Moderator Temperature Coefficient (MTC);"
3. LCO 3.1.5, "Control Element Assembly (CEA) Alignment;"
4. LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits;"
5. LCO 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits;"
6. LCO 3.2.1, "Linear Heat Rate (LHR);"
7. LCO 3.2.4, Departure from Nucleate Boiling Ratio (DNBR);"
8. LCO 3.2.5, "AXIAL SHAPE INDEX (ASI);"
9. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits;" and
10. LCO 3.9.1, "Boron Concentration."

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2 INSERT 2

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

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

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

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

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

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

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

8. CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs"; and

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes."

2 INSERT 3

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits;"

2. LCO 3.4.6, "RCS Loops - MODE 4;"

3. LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled;"

4. LCO 3.4.10, "Pressurizer Safety Valves;" and

5. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

2 **INSERT 4**

CE NPSD-683-A, "The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications."

5.~~6~~ 7 Reporting Requirements

5.7.1.6 5.~~6.4~~ 7.1.6 RCS Pressure and Temperature Limits Report (continued)

5.7.1.6.c c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

---REVIEWER'S NOTE---

~~The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~

1. ~~The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~
2. ~~The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~
3. ~~Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.~~
4. ~~The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~ 3
5. ~~The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.~~
6. ~~The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~
7. ~~Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~

U2/U3 CTS

All changes are 1 unless otherwise noted

Reporting Requirements
5.~~6~~ 7

5.7.2 Special Reports

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.~~6~~ 7 Reporting Requirements

5.7.2.a 5.~~6.5~~ 7.2.a **Post Accident Monitoring Report**

When a report is required by Condition B or F of LCO 3.3.[11], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. 2

5.7.2.b 5.~~6.6~~ 7.2.b **Tendon Surveillance Report**

~~[~~ Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken. ~~]~~ 2

5.7.2.c 5.~~6.7~~ 7.2.c **Steam Generator Tube Inspection Report**

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.~~9~~ 2.11, "Steam Generator (SG) Program." The report shall include:

1. ~~a.~~ The scope of inspections performed on each SG~~,~~ ; 4
2. ~~b.~~ Active degradation mechanisms found~~,~~ ; 4
3. ~~c.~~ Nondestructive examination techniques utilized for each degradation mechanism~~,~~ ; 4
4. ~~d.~~ Location, orientation (if linear), and measured sizes (if available) of service induced indications~~,~~ ; 4
5. ~~e.~~ Number of tubes plugged ~~[or repaired]~~ during the inspection outage for each active degradation mechanism~~,~~ ; 4
6. ~~f.~~ Total number and percentage of tubes plugged ~~[or repaired]~~ to date~~,~~ ; and 4
7. ~~g.~~ The results of condition monitoring, including the results of tube pulls and in-situ testing~~,~~ . 4

~~[h. The effective plugging percentage for all plugging [and tube repairs] in each SG, and]~~

~~[i. Repair method utilized and the number of tubes repaired by each repair method.]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 5.7, REPORTING REQUIREMENTS

1. The Specification number has been changed to be consistent with the Specification number in the SONGS CTS. SCE has decided not to renumber the CTS to be consistent with the ISTS because by doing so would result in the unnecessary administrative burden of changing TS numbers in plant procedures. Also, since some reports are numbered under ITS 5.7.1 and some are numbered under ITS 5.7.2, the statement in ISTS 5.7.1 that the reports are to be submitted in accordance with 10 CFR 50.4 has also been included in ITS 5.7.2.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewers Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

4. Correct punctuation is used and is consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01.

5. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

6. ISTS 5.6.3.c (ITS 5.7.1.5.c) is being revised to add in the statement "assuming operation at RATED THERMAL POWER." This change is consistent with approved TSTF-487, which has already been adopted into the SONGS Units 2 and 3 CTS, as documented in the NRC Safety Evaluation for Amendments 219 and 212, respectively, dated 02/03/2009 (ADAMS Accession No. ML083470091).

7. ISTS 5.6.2 (ITS 5.7.1.3) is being revised from requiring the Radioactive Effluent Release Report to cover operation of the unit in the previous year, to requiring the report to cover operation of the unit in the previous "calendar" year. This change defines the specific year period that the report covers to the previous calendar year versus any year period. This change is consistent with the SONGS CTS.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.7, REPORTING REQUIREMENTS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 7

ITS 5.8, HIGH RADIATION AREA

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A01

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

M01

INSERT 1

INSERT 2

L01

5.8.1.a, 5.8.1.b, 5.8.2.a, 5.8.2.b

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

Add proposed ITS 5.8.1.c and 5.8.2.c

L06

5.8.1.d, 5.8.2.d

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

5.8.1.d.1

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

5.8.1.d.2, 5.8.2.d.1

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device

L02

5.8.1.e, 5.8.2.e

may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them.

INSERT 3

INSERT 4

M02

5.8.1.d.4, 5.8.2.d.3

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

L04

Add proposed ITS 5.8.1.d.4 and 5.8.2.d.3

Add proposed ITS 5.8.1.d.3

L03

(continued)

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A01

INSERT 1

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.8.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

INSERT 2

L01 M01

or equivalent that includes specification or radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measurers.

L02

INSERT 3

Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals,

M02

INSERT 4

These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

ITS

A01

5.8. High Radiation Area (continued)

5.8.2

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP or equivalent that specifies the dose rates in the immediate work areas ~~and the maximum allowable stay time for individuals in that area~~ INSERT 5. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.2.a

5.8.2.b

5.8.2.d.3(ii)

L01

M01

L04

Add proposed ITS 5.8.2.d.2

L03

Add proposed ITS 5.8.2.d.4

L05

5.8.2.f

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

5.8

M01 **INSERT 5**

and other appropriate radiation protection equipment and measures.

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5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

INSERT 1

M01

L01

5.8.1.a, 5.8.1.b, 5.8.2.a, 5.8.2.b 5.8.1 Each high radiation area as defined in 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).

INSERT 2

Add proposed ITS 5.8.1.c and 5.8.2.c

L06

5.8.1.d, 5.8.2.d Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

5.8.1.d.1 a. A radiation monitoring device that continuously indicates the radiation dose rate in the area,

5.8.1.d.2, 5.8.2.d.1 b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received.

L02

5.8.1.e, 5.8.2.e Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them.

INSERT 3

INSERT 4

M02

5.8.1.d.4, 5.8.2.d.3 c. ~~An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.~~

L04

Add proposed ITS 5.8.1.d.4 and 5.8.2.d.3

Add proposed ITS 5.8.1.d.3

L03

(continued)

5.8

A01

INSERT 1

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.8.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

INSERT 2

L01 M01

or equivalent that includes specification or radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measurers.

L02

INSERT 3

Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals,

M02

INSERT 4

These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.8. High Radiation Area (continued)

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

5.8.2

L01

or equivalent

5.8.2.a

5.8.2.b

INSERT 5

M01

5.8.2.d.3(ii)

L04

Add proposed ITS 5.8.2.d.2

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

L03

5.8.2.f

Add proposed ITS 5.8.2.d.4

L05

5.8

M01 **INSERT 5**

and other appropriate radiation protection equipment and measures.

DISCUSSION OF CHANGES
ITS 5.8, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A01 In the conversion of the San Onofre Nuclear Generating Station (SONGS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 3.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 5.8.1, in reference to entrance to a high radiation area states "…entrance thereto shall be controlled by requiring the issuance of a Radiation Exposure Permit (REP)." CTS 5.8.2 states "… under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area." ITS 5.8.1.b and ITS 5.8.2.b state, "Access to, and activities in, each such area shall be controlled by means of a Radiation Exposure Permit or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures." This changes the CTS by specifying certain information is required to be in the REP or equivalent. The addition of the option to a means equivalent to the REP is addressed in DOC L01.

The purpose of the REP requirement in CTS 5.8.1 and 5.8.2 is to ensure personnel entering a high radiation area have the information necessary to work safely in those areas from a radiation standpoint. This change is acceptable because it states specific information to be included in the REP to accomplish the same goal, and requiring issuance of the REP with the required information makes the information available. These changes are designated as more restrictive because additional information to be included in the REP is required.

M02 CTS 5.8.1.b states that one of the optional criteria that allows entry into a high radiation is a radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. It further states that entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them. ITS 5.8.1.e and ITS 5.8.2.e include a similar requirement, but also require that the continuously escorted personnel will receive a pre-job briefing prior to entry into such areas, and that the dose rate determination, knowledge, and pre-job briefing do not require documentation prior to entry. This changes the CTS by expanding the requirement to apply to all the options for conditions allowing entry into a high radiation area, and adding the criteria that the continuously escorted personnel will receive a pre-job briefing prior to entry into such areas and that the dose rate determination, knowledge, and pre-job briefing do not require documentation prior to entry.

DISCUSSION OF CHANGES
ITS 5.8, HIGH RADIATION AREA

The purpose of the second sentence in CTS 5.8.1.b is to ensure personnel entering high radiation areas are aware of dose rates in the area. The proposed addition further ensures that a pre-job brief is conducted. This change is acceptable because it provides additional guidance to ensure personnel are aware of the relevant dose rates. This change is designated as more restrictive because additional criteria are added to the requirements for entering a high radiation area.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 5.8.1 states, for high radiation areas, "… entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP). CTS 5.8.2 also references use of an REP when personnel access certain high radiation areas. ITS 5.8.1 and ITS 5.8.2 state, for high radiation area, "Access to, and activities in, each such area shall be controlled by means of Radiation Exposure Permit (REP) *or equivalent* (emphasis added) that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures." This changes the CTS by allowing an equivalent document to be used for access control. The addition of details required in the REP is addressed in DOC M01.

The purpose of the specified phrase in CTS 5.8.1 and 5.8.2 is to designate the document through which access is controlled to the specified high radiation areas. This change is acceptable because a proper document is still required, but it may serve the same purpose as an REP without having to be specifically called an REP. This change is designated as less restrictive because an alternate document may be used for access control in lieu of an REP.

L02 *(Category 1 – Relaxation of LCO Requirements)* CTS 5.8.1.b states that entry into a high radiation area when using a radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received can only be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them. ITS 5.8.1.e and ITS 5.8.2.e allows entry without this specific requirement provided the individuals are continuously escorted by an individual qualified in radiation protection procedures. This changes the CTS by allowing individuals qualified in radiation protection procedures or by individuals that are continuously escorted by an individual qualified in radiation protection procedures to enter a

DISCUSSION OF CHANGES
ITS 5.8, HIGH RADIATION AREA

high radiation area using a radiation monitoring device that continuously integrates the dose and alarms at a preset dose prior to knowing the dose rate in the area.

The purpose of CTS 5.8.1.b is to provide adequate protection such that individuals entering high radiation areas will not exceed predetermined dose limits. However, allowing entry prior to knowing the actual dose rates can reduce the overall dose received by personnel, since it reduces the total number of personnel that have to enter the area. The current requirements require an individual to first enter the area and determine the dose rates prior to individual(s) that actually need to enter the area to perform work. Thus, an additional person receives a dose above what is needed to actually perform the work. Allowing an individual qualified in radiation protection procedures to enter the area prior to knowing the dose would reduce the overall dose received by site personnel, while ensuring that personnel are adequately protected (since at least one individual making the initial entry must be trained in radiation protection procedures). Therefore, this change is considered acceptable. This change is designated as less restrictive because the ITS will allow entry into a high radiation area under certain conditions without knowing the specific dose rate in the area.

L03 *(Category 1 – Relaxation of LCO Requirements)* ITS 5.8.1.d.3 and ITS 5.8.2.d.2 state that one of the options for devices that an individual or group shall possess for radiation monitoring when entering a high radiation area is "A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area." ITS 5.8.2.d.2 also requires a means to communicate with and control every individual in the area. CTS 5.8.1, 5.8.2, and 5.8.3 do not contain these options for an individual or group. This changes the CTS by providing an additional device that an individual entering these high radiation areas must possess for radiation monitoring.

The purpose of ITS 5.8.1.d.3 and ITS 5.8.2.d.2 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide reliable means of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose in high radiation areas has been provided.

L04 *(Category 1 – Relaxation of LCO Requirements)* CTS 5.8.1.c states that one of the optional criteria that allow entry into a high radiation area is "An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP." CTS 5.8.2 allows that in lieu of the stay time specification on the REP, direct or remote continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. ITS 5.8.1.d.4 states "A self reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

DISCUSSION OF CHANGES
ITS 5.8, HIGH RADIATION AREA

(i) be under the surveillance, as specified in the REP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area; or (ii) be under the surveillance, as specified in the REP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance." ITS 5.8.2.d.3 reads the same as ITS 5.8.1.d.4 except the last phrase, "communicate with individuals in the area who are covered by such surveillance," is replaced with the phrase, "communicate with and control every individual in the area." This changes the CTS by deleting the discussion of positive controls over activities and performing radiation surveillance with a requirement for the monitoring device to have continuous dose rate displays and the responsibility to control dose rates in the area and adds an option to perform the monitoring of personnel remotely using the specified equipment and processes for a high radiation area < 1000 mrem/hr. It further changes the CTS by specifying the individual has a self-reading pocket dosimeter when using this option.

The purpose of CTS 5.8.1.c is to provide the option of monitoring the exposure of individuals in high radiation areas by a separate individual qualified in radiation procedures. This change is acceptable because it provides adequate means of monitoring the personnel in the high radiation areas, but provides added flexibility for how to do it. This change is designated as less restrictive because additional methods for monitoring personnel exposure are provided.

L05 *(Category 1 – Relaxation of LCO Requirements)* ITS 5.8.2.d.4 permits the use of a radiation monitoring device that continuously displays radiation dose rates in the area when ITS 5.8.2.d.2 and ITS 5.7.2.d.3 are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle. CTS 5.8.1, 5.8.2, and 5.8.3 do not contain this option. This changes the CTS by providing an additional option for devices an individual entering these high radiation areas must use to control radiation dose.

The purpose of ITS 5.8.2.d.4 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide a reliable means of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose of personnel in high radiation areas has been provided.

L06 *(Category 1 – Relaxation of LCO Requirements)* CTS 5.8.1 requires entry into any high radiation area to be controlled by the issuance of a Radiation Exposure Permit (REP). ITS 5.8.1.c and 5.8.2.c relaxes this requirement such that individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the REP requirement (or equivalent) while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry into, exit from, and work in such areas. This changes the CTS by allowing entry into a high radiation area for plant personnel under certain conditions.

DISCUSSION OF CHANGES
ITS 5.8, HIGH RADIATION AREA

The purpose of a REP is to ensure personnel are cognizant of radiation doses in the work areas and to ensure the dose limits for personnel exposure in 10 CFR Part 20 are not exceeded. This change is acceptable because the exemption is only for personnel qualified in radiation protection procedures and for personnel continuously escorted by said personnel. Personnel qualified in radiation protection procedures are adequately trained to ensure dose is maintained as low as reasonably achievable. This change is designated as less restrictive since a REP will not be required under all plant conditions during entry into a high radiation area.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

~~[~~ 5.~~7~~ 8 High Radiation Area ~~]~~ 2

DOC A01 As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

DOC A01 5.~~7~~ 8.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

5.8.1 a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

5.8.1 b. Access to, and activities in, each such area shall be controlled by means of Radiation Exposure ~~Work~~ Permit (REP ~~RWP~~) or equivalent that includes specification of ~~or~~ radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measure~~r~~s. 4 5 5

DOC L06 c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an REP ~~RWP~~ or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas. 4

5.8.1 d. Each individual or group entering such an area shall possess:

5.8.1.a 1. A radiation monitoring device that continuously displays radiation dose rates in the area; ~~,~~ or 3

5.8.1.b 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; ~~,~~ or 3

DOC L03 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; ~~,~~ or 3

DOC L04 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter); and ~~,~~ 3

8

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

8

REP

DOC L04 (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or ; 4 3

REP

DOC L04 (ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance. 4

5.8.1.b e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

8

5.8.2 a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

5.8.2 1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

5.8.2 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

5.8.2 b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures. 4

REP

8
5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u> (continued)
8

DOC L06 c. Individuals qualified in radiation protection procedures ~~my~~ may be exempted from the requirement for an ~~RWP~~ REP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas. 5 4

5.8.1 d. Each individual or group entering such an area shall possess:

5.8.1.b 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint~~,~~ ; or 3

DOC L03 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area~~,~~ ; or 3

DOC L04 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) ; and 3

DOC L04 (i) Be under the surveillance, as specified in the ~~RWP~~ REP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area~~,~~ ; or 4 3

DOC L04 (ii) Be under the surveillance, as specified in the ~~RWP~~ REP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area. 4

DOC L05 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

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5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

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5.8.1.b e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.8.3 f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

JUSTIFICATION FOR DEVIATIONS
ITS 5.8, HIGH RADIATION AREA

1. The Specification number has been changed to be consistent with the Specification number in the SONGS CTS. SCE has decided not to renumber the CTS to be consistent with the ISTS because by doing so would result in the unnecessary administrative burden of changing TS numbers in plant procedures.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Correct punctuation is used and is consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01.

4. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.8, HIGH RADIATION AREA

There are no specific NSHC discussions for this Specification.