ATTACHMENT C

PROPOSED TECHNICAL SPECIFICATION 3/4.7.6

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 2

3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY:

MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.6.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.e or 4.7.6.f, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

d. Functional Tests*

At least once per 18 months during shutdown, a representative sample of at least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.6.e or 4.7.6.f. an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine motor, etc.)

3. Snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.*

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

e. Hydraulic Snubbers Functional Test Acceptance Criteria*

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Mechanical Snubbers Functional Test Acceptance Criteria*

The mechanical snubber functional test shall verify that:

1. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubber are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers were attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

*Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

SURVEILLANCE REQUIREMENTS (Continued)

g. Functional Test Failure Analysis (Continued)

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.6.e. or 4.7.6.f. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

PLANT SYSTEMS

BASES

3/4.7.6 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

For visual inspection snubbers are categorized into two (2) groups, those accessable and those inaccessable during reactor operation. For functional testing, snubbers are categorized into types by design and manufacturer, irrespective of capacity. For example, Pacific Scientific snubbers are divided into four types corresponding to different design features: PSA 1/4 and 1/2 are one type; PSA 1, 3 and 10 are another; PSA 6 is another; and PSA 35 and 100 are a fourth type.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replace, spring replace, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubbers service life review are not intended to affect plant operation.

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

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

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

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient of operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

6.10.2 (Continued)

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRC and the NSG.

l. Records of the service lives of all snubbers within the scope of Technical Specification 3/4.7.6 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

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

ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATION 3/4.7.6

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 3

3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY:

MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.6.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
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The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.e or 4.7.6.f, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

d. Functional Tests*

At least once per 18 months during shutdown, a representative sample of at least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.6.e or 4.7.6.f. an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine motor, etc.)

3. Snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.*

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

e. Hydraulic Snubbers Functional Test Acceptance Criteria*

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Mechanical Snubbers Functional Test Acceptance Criteria*

The mechanical snubber functional test shall verify that:

1. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubber are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers were attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

*Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

SURVEILLANCE REQUIREMENTS (Continued)

g. Functional Test Failure Analysis (Continued)

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.6.e. or 4.7.6.f. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

3/4.7.6 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

For visual inspection snubbers are categorized into two (2) groups, those accessable and those inaccessable during reactor operation. For functional testing, snubbers are categorized into types by design and manufacturer, irrespective of capacity. For example, Pacific Scientific snubbers are divided into four types corresponding to different design features: PSA 1/4 and 1/2 are one type; PSA 1, 3 and 10 are another; PSA 6 is another; and PSA 35 and 100 are a fourth type.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replace, spring replace, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubbers service life review are not intended to affect plant operation.

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

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

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

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient of operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

6.10.2 (Continued)

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRC and the NSG.

l. Records of the service lives of all snubbers within the scope of Technical Specification 3/4.7.6 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

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

DESCRIPTION OF PROPOSED CHANGE NPF 15 163 AND SAFETY ANALYSIS

This is a request to revise Sections 3.1.2.7 Borated Water Source-Shutdown, 3.1.2.8 Borated Water Sources - Operating, and Bases 3/4.1.2 Boration Systems of the Technical Specifications for San Onofre Nuclear Generating Station Unit 3.

Description

The proposed change revises Technical Specifications 3.1.2.7 Borated Water Source - Shutdown, 3.1.2.8 Borated Water Source - Operating and Bases 3/4.1.2 Boration Systems. Technical Specifications 3.1.2.7 and 3.1.2.8 require borated water source operability and specify volume, temperature and boron concentration requirements which ensure that sufficient negative reactivity control is available during each mode of facility operation. Sections 3.1.2.7 and 3.1.2.8 specify the minimum boric acid makeup tank (BAMUT) volume and temperature required as a function of boric acid concentration. The proposed change decreases the BAMUT volume, increases BAMUT concentration and increases the minimum refueling water storage tank (RWST) volume specified by Section 3.1.2.7, while maintaining reactivity control consistent with the revised safety analysis associated with plant refueling and Cycle 2 operation.

The BAMUT and RWST are part of the boron injection system which insures that negative reactivity control is available during each mode of facility operation. This system is required to satisfy 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability." Criterion 26 states that a nuclear power plant should contain two independent reactivity control systems, one of which is capable of holding the reactor core subcritical under shutdown conditions. The BAMUT and RWST are required to provide Reactor Coolant System (RCS) shutdown boration and makeup capability. Procedurally, boric acid from the BAMUT is charged into the RCS before cooldown begins in order to meet shutdown margin requirements at the end of the cooldown. Additional borated water from the BAMUT and RWST is charged into the RCS during the cooldown as required to maintain pressurizer level.

Core performance analyses of the Cycle 2 reactor fuel management design show that the boron concentration required to 1) maintain the required SHUTDOWN MARGIN after xenon decay and cooldown to 200°F and 2) satisfy Criterion 26 has increased due to the differences in core design and core performance characteristics from Cycle 1. As a consequence, the minimum borated water volume in the BAMUT (shown in Figure 3.1-1) and RWST must be revised for Cycle 2 operation in order to meet the Limiting Conditions for Operation on Shutdown Margin. The Modes 1 through 4 BAMUT volume requirement has been decreased in order to facilitate plant operation while providing the required Shutdown Margin. The minimum boric acid concentration in the BAMUT for

Modes 1 through 4 is set by the cold shutdown RCS boron concentration requirement (which is greater for the more reactive Cycle 2 core) and the pressurizer volume available for the pre-cooldown boration with letdown isolated (which is unchanged from Cycle 1). The minimum boric acid volume required in the BAMUT by Technical Specification 3.1.2.8 for Cycle 1, included both the initial boration volume and a portion of the cooldown makeup volume. For Cycle 2 operation, the cooldown makeup volume will be contained in the RWST; and the minimum boric acid volume required in the BAMUT will include the initial boration volume required to establish the Mode 5 Shutdown Margin. The combined effect of the increased boric acid concentration required for Cycle 2 operation and decrease in excessively conservative volume requirement, is a net decrease in the volume specified for operation in Modes 1 through 4, in Figure 3.1-1.

Sections 3.1.2.8 and B3/4.1.2 have been revised to specify the BAMUT volume/concentration and the RWST volume required for negative reactivity control consistent with the requirements of Cycle 2 operation. The minimum RWST volume for boration specified in Bases Section 3/4.1.2, is set by boration and makeup requirements with letdown available and BAMUT's unavailable. This volume (which has increased from 53,500 gallons for Cycle 1 to 81,970 gallons for Cycle 2, in Modes 1 through 4) exceeds that required for RCS cooldown makeup alone and remains bounded by the Safety Injection requirements of Technical Specification 3/4.5.4. The BAMUT volume/concentration required by Technical Specification 3.1.2.7 for operation in Modes 5 and 6, as shown in Figure 3.1-1, has increased for Cycle 2 in order to meet the increased Cycle 2 shutdown margin requirements. Accordingly, the minimum RWST volume for boration in Modes 5 and 6 with letdown available and BAMUT's unavailable has increased from 5465 gallons for Cycle 1 to 9970 gallons for Cycle 2.

The maximum allowed boric acid concentration specified in figure 3.1-1 remains unchanged for Cycle 2. Therefore, the boron mixing demonstrated during the preoperational natural circulation test remains valid.

Existing Technical Specifications

See Attachment A.

Proposed Technical Specifications

See Attachment B.

Safety Analysis

The proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas.

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

The proposed change is designed to maintain the same or greater shutdown margins in the facility, thus avoiding any increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change ensures that sufficient negative reactivity control is available during each mode of facility operation, and maintains the same or greater shutdown margins as in Cycle 1. The proposed change does not result in a condition which could lead to a new or different kind of accident.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The specific purpose of the proposed change is to maintain the same margin of safety with respect to the design criteria during Cycle 2 operation as in Cycle 1.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (iii) relates to a change resulting from a nuclear reactor core reloading where 1) no fuel assemblies are significantly different from those previously found acceptable to the NRC for the subject facility, 2) no significant changes are made to the acceptance criteria for the technical specifications, 3) the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and 4) the NRC has previously found such methods acceptable. The proposed change is representative of Example (iii) in that it results from a nuclear reactor core reloading where, in particular, no significant changes have been made to the boration source technical specification acceptance criteria or the analytical methodology employed in the determination of the proposed criteria.

<u>Safety and Significant Hazards Determination</u>

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

Attachment A

Existing Technical Specifications

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage tanks with:

1. A minimum borated water volume of 5465 gallons above the ECCS suction connection,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water when the outside air temperature is less than 40°F or greater than 100°F.

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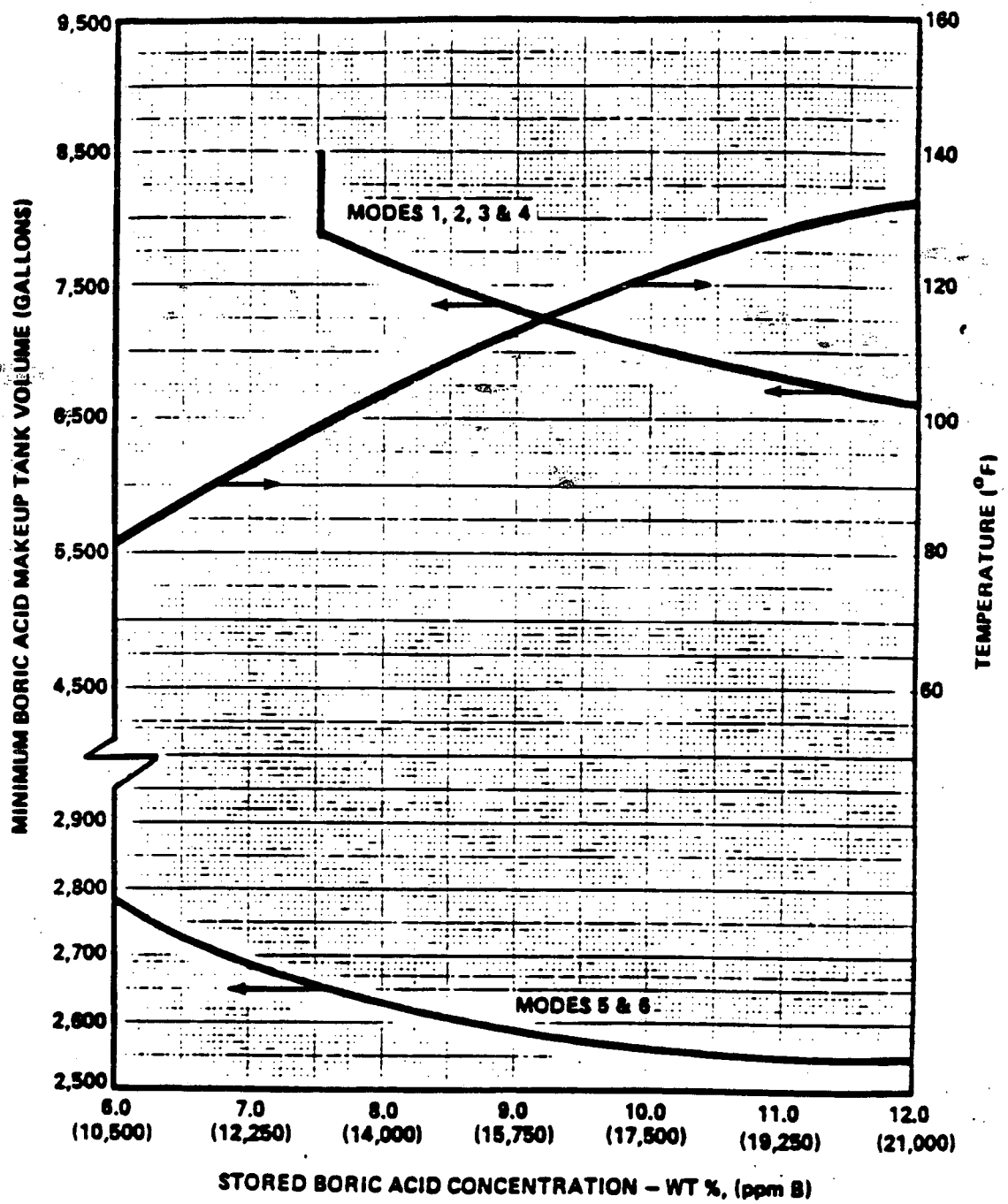

Figure 3.1-1

MINIMUM BORIC ACID STORAGE TANK VOLUME AND TEMPERATURE AS A FUNCTION OF STORED BORIC ACID CONCENTRATION

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank and at least one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,
2. Between 1720 and 2300 ppm of boron, and
3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,
2. Verifying the contained borated water volume of the water source, and
3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 100°F.

NOV 15 1982

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 2.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 53,500 gallons of 1720 ppm borated water from the refueling water tank. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to the more restrictive value of Specification 3.5.4.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5465 gallons of 1720 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

NOV 1 5 1982

Attachment B

Proposed Technical Specifications

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage tanks with:

1. A minimum borated water volume of 9970 gallons above the ECCS suction connection,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water when the outside air temperature is less than 40°F or greater than 100°F.

NOV 1 5 1982

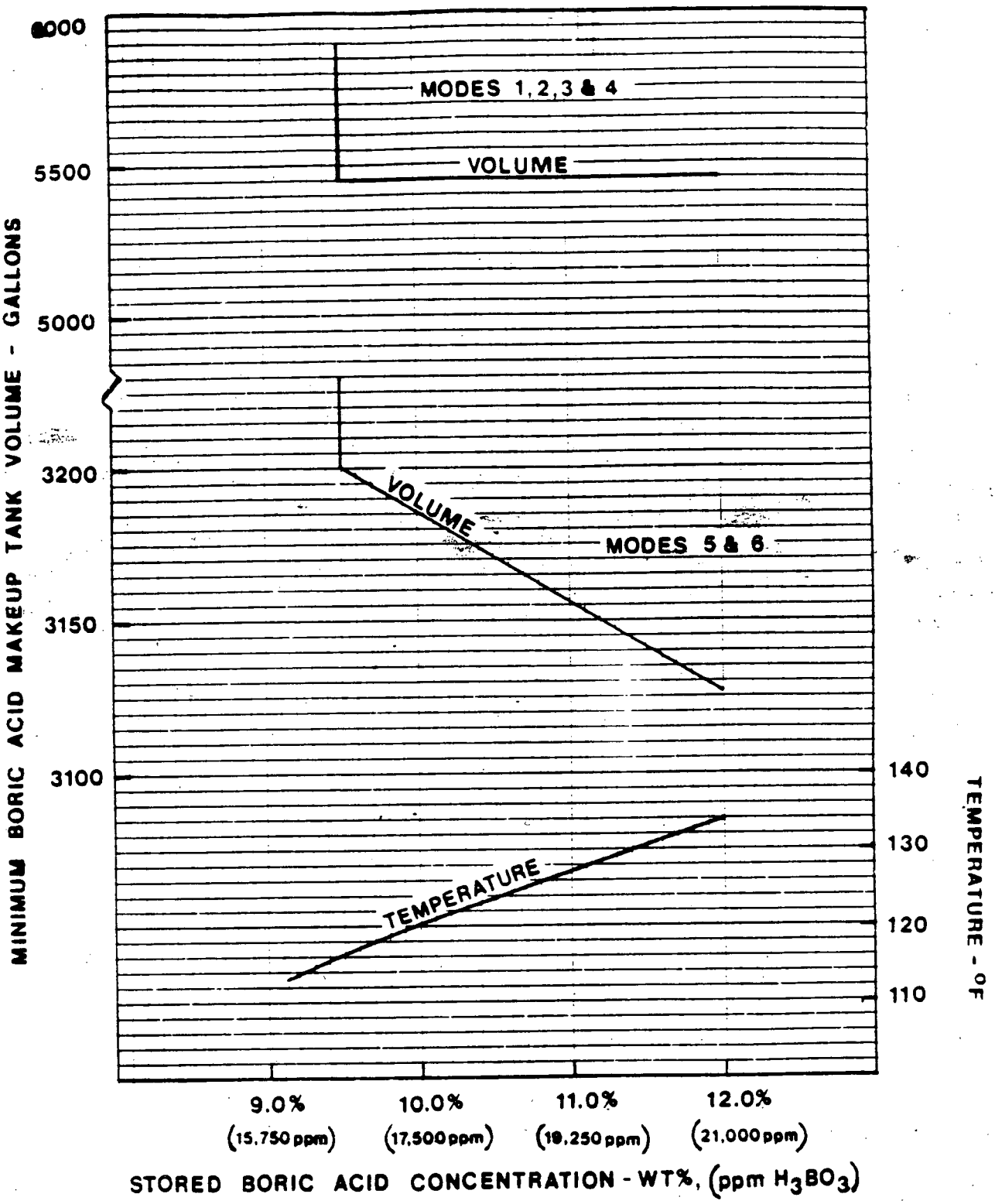

Figure 3.1-1

MINIMUM BORIC ACID STORAGE TANK VOLUME AND MINIMUM TEMPERATURE BEFORE PRECIPITATION AS A FUNCTION OF STORED BORIC ACID H_3BO_3 CONCENTRATION

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank and at least one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,

2. Between 1720 and 2300 ppm of boron, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 100°F.

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3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 2.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 81,970 gallons of 1720 ppm borated water from the refueling water tank. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to the more restrictive value of Specification 3.5.4.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 9970 gallons of 1720 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

NOV 15 1982

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-165
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.4.8.2, Reactor Coolant System - Pressurizer - Heatup/Cooldown, and Technical Specification 5.7, Component Cyclic or Transient Limits, for San Onofre Nuclear Generating Station, Units 2 and 3.

Description

The proposed change revises Surveillance Requirement 4.4.8.2.2 of Technical Specification 3/4.4.8.2, Reactor Coolant System - Pressurizer - Heatup/Cooldown, and Table 5.7-1, Component Cyclic or Transient Limits, of Technical Specification 5.7, Component Cyclic or Transient Limits. The change provides the method to determine the Reactor Coolant System cumulative usage factor for the modified pressurizer spray system. Table 5.7-1 has been revised for calculating the pressurizer spray system cumulative usage factor. This change will allow the evaluation of the modified spray system for temperature transient effects. The new method of calculating the usage factor is based on an analysis in accordance with subarticles NB-3222 and NB-3650 of ASME Section III of the entire spray system, whereas previously only the spray nozzle was considered in the analysis. The change will provide more flexibility in the thermal cycle logging requirements for calculation of the usage factor specified in Technical Specification Section 5.7.1. Nevertheless, the proposed cumulative usage factor limit of 0.65 will provide an increased margin of safety to the ASME Code limit of 1.0 compared to that provided by the existing cumulative usage factor limit of 0.75.

The current Surveillance Requirement 4.4.8.2.2 states that the spray water temperature differential shall be determined for use in Table 5.7-1 at least once per 12 hours during auxiliary spray operation. The proposed change to this section states that the spray water temperature differential shall be determined for use in Table 5.7-1 prior to each cycle of main spray when less than four reactor coolant pumps are operating and for each cycle of auxiliary spray operation.

Calculation of the pressurizer spray line cumulative usage factor has been revised such that a cycle is not recorded until the maximum temperature difference between the pressurizer and pressurizer spray water during the spray cycle exceeds 200°F as opposed to 150°F presently. The number of spray cycles logged prior to calculation of the cumulative usage factor has been revised accordingly. The acceptance criteria to be met by application of Table 5.7-1 is that the cumulative usage factor in both the spray piping and spray nozzle do not exceed a limit of 0.65. This provides an increased safety margin to the ASME Code limit of 1.0.

This change is required due to a modification in the spray system piping to minimize the effects of thermal fatigue loadings in the modified spray piping system.

Because this proposed change of Table 5.7-1 is more restrictive than the present method of spray system usage evaluation and does not affect the operation of the facility, there can be no increase in the probability or consequences of an accident previously evaluated, nor can there be a possibility of a new or different kind of accident from any accident previously evaluated. The addition of this change ensures that the more conservative limit of the cumulative usage factor will not be exceeded, hence, there will be no reduction in the margin of safety.

Present Technical Specifications (Unit 2)

See Attachment A.

Proposed Technical Specifications (Unit 2)

See Attachment B.

Present Technical Specifications (Unit 3)

See Attachment C.

Proposed Technical Specifications (Unit 3)

See Attachment D.

Safety Analysis

The proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas.

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The probability or consequence of an accident is not increased by the proposed change since this change provides a limit for the modified spray system usage which is more restrictive than that provided previously. This change prevents an increase in the probability of an accident previously evaluated by providing a more conservative limitation for the modified spray system usage.

2. Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Since the operation of the facility in accordance with the proposed amendment will not change, there is no new or different kind of accident from any accident previously evaluted that could occur.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   There is no reduction in the margin of safety previously established, since the analysis done for this modification shows that the allowable usage factor limit will not be exceeded. Furthermore, the proposed change provides an increased margin of safety to the ASME code usage factor limit of 1.0.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered least likely to involve significant hazards considerations. Example (ii) from the Federal Register states that a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications would not be likely to involve significant hazards considerations. The proposed change is similar to Example (ii) in that the change provides a more accurate and stringent evaluation of the modified spray system by analyzing the entire spray system, and not only the spray nozzle. Thus, more limiting factors are imposed on the analysis, making it more restrictive.

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT A

EXISTING TECHNICAL SPECIFICATIONS

UNIT 2

PRESSURIZER - HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 at least once per 12 hours during auxiliary spray operation.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	2 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODES 1, 2 or 3.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Method for Calculating Pressurizer Spray Nozzle Cumulative Usage Factor

ΔT	N_A	N	N/N_A
150 - 200	50,000		
201 - 300	7,000		
301 - 400	2,000		
401 - 500	1,000		
501 - 600	800		
		$\Sigma N/N_A$	

Where:

ΔT = Temperature difference between pressurizer water and spray in °F.

N_A = Allowable number of spray cycles.

N = Number of cycles in ΔT range indicated.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Calculational Method:

1. The spray cycle is defined as the opening and closing of a spray valve, either by main spray or auxiliary spray.

2. If the difference between the pressurizer water temperature and the spray water temperature exceeds 150°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor is calculated as follows:

   A. Fill in Column "N" above from plant records.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the calculated usage factor is equal to or less than 0.75, no further action is required.

4. If the calculated usage factor exceeds 0.75, subsequent pressurizer spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 150°F when spray is operated. An engineering evaluation of nozzle fatigue shall be performed and shall determine that that the nozzle remains acceptable for additional service prior to removing this restriction.

ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATIONS

UNIT 2

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any one hour period.

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 for each cycle of main spray when less than 4 reactor coolant pumps are operating and for each cycle of auxiliary spray operation.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	2 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODES 1, 2 or 3.
Pressurizer Spray System	Unlimited number of cycles.	Main spray (4 pumps operating) Main spray (less than 4 pumps operating) with $\Delta T \leq 200^{o}F$. Auxiliary spray with $\Delta T \leq 200^{o}F$.
	Calculate cumulative usage factor.	Main spray (less than 4 pumps operating) with $\Delta T > 200^{o}F$ Auxiliary spray with $\Delta T > 200^{o}F$

Where:

ΔT = Maximum temperature difference between pressurizer and pressurizer spray during the spray cycle.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray System		

Pressurizer Spray System Usage Factor

ΔT	N_A	N	N/N_A
201 - 250	11,000		
251 - 300	4,000		
301 - 350	2,200		
351 - 400	1,300		
401 - 450	900		
451 - 500	500		
501 - 550	300		
551 - 600	200		

$\Sigma N/N_A =$ _____

where:

ΔT = Maximum temperature difference between pressurizer and pressurizer spray during the spray cycle.

N_A = Allowable number of spray cycles

N = Number of cycles in ΔT range indicated

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray System		

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow through the pressurizer spray nozzle.

2. If the maximum temperature difference between the pressurizer and the pressurizer spray during the spray cycle exceeds 200°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray system usage factor is calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$. This total is the cumulative usage factor.

4. A. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

   B. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of spray system fatigue shall be performed within 90 days. The evaluation shall determine that the spray system remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the maximum temperature difference between pressurizer and pressurizer spray during the spray cycle shall be limited to less than or equal to 200°F.

ATTACHMENT C

EXISTING TECHNICAL SPECIFICATIONS

UNIT 3

REACTOR COOLANT SYSTEM

PRESSURIZER - HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any 1 hour period,

b. A maximum cooldown of 200°F in any 1 hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 at least once per 12 hours during auxiliary spray operation.

NOV 15 1982

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	2 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODES 1, 2 or 3.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Method for Calculating Pressurizer Spray Nozzle Cumulative Usage Factor

ΔT	N_A	N	N/N_A
150 - 200	50,000		
201 - 300	7,000		
301 - 400	2,000		
401 - 500	1,000		
501 - 600	800		
		$\Sigma N/N_A$	

Where:

ΔT = Temperature difference between pressurizer water and spray in °F.

N_A = Allowable number of spray cycles.

N = Number of cycles in ΔT range indicated.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Calculational Method:

1. The spray cycle is defined as the opening and closing of a spray valve, either by main spray or auxiliary spray.

2. If the difference between the pressurizer water temperature and the spray water temperature exceeds 150°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor is calculated as follows:

   A. Fill in Column "N" above from plant records.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the calculated usage factor is equal to or less than 0.75, no further action is required.

4. If the calculated usage factor exceeds 0.75, subsequent pressurizer spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 150°F when spray is operated. An engineering evaluation of nozzle fatigue shall be performed and shall determine that the nozzle remains acceptable for additional service prior to removing this restriction.

NOV 15 1982

ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATIONS

UNIT 3

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 for each cycle of main spray when less than 4 reactor coolant pumps are operating and for each cycle of auxiliary spray operation.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	2 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODES 1, 2 or 3.
Pressurizer Spray System	Unlimited number of cycles.	Main spray (4 pumps operating) Main spray (less than 4 pumps operating) with $\Delta T \leq 200^{o}F$. Auxiliary spray with $\Delta T \leq 200^{o}F$.
	Calculate cumulative usage factor.	Main spray (less than 4 pumps operating) with $\Delta T > 200^{o}F$ Auxiliary spray with $\Delta T > 200^{o}F$

Where:

ΔT = Maximum temperature difference between pressurizer and pressurizer spray during the spray cycle.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray System		

Pressurizer Spray System Usage Factor

ΔT	N_A	N	N/N_A
201 - 250	11,000		
251 - 300	4,000		
301 - 350	2,200		
351 - 400	1,300		
401 - 450	900		
451 - 500	500		
501 - 550	300		
551 - 600	200		

$\Sigma N/N_A =$ _____

where:

ΔT = Maximum temperature difference between pressurizer and pressurizer spray during the spray cycle

N_A = Allowable number of spray cycles

N = Number of cycles in ΔT range indicated

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray System		

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow through the pressurizer spray nozzle.

2. If the maximum temperature difference between the pressurizer and the pressurizer spray during the spray cycle exceeds 200°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray system usage factor is calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$. This total is the cumulative usage factor.

4. A. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

   B. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of spray system fatigue shall be performed within 90 days. The evaluation shall determine that the spray system remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the maximum temperature difference between pressurizer and pressurizer spray during the spray cycle shall be limited to less than or equal to 200°F.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-183 AND SAFETY ANALYSIS

This is a request to revise Technical Specification Table 3.3-10, Accident Monitoring Instrumentation, Technical Specification Table 4.3-7, Accident Monitoring Instrumentation Surveillance Requirements, and associated Bases of the Technical Specifications for San Onofre Nuclear Generating Station, Unit 2.

Existing Technical Specifications

See Attachment A.

Proposed Technical Specifications

See Attachment B.

Description

The proposed change revises Technical Specification 3.3.3.6, Accident Monitoring Instrumentation, to add the Heated Junction Thermocouple System - Reactor Vessel Level Monitoring System (RVLMS) to Tables 3.3-10 and 4.3-7, and its Bases. This implements Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling," as requested by NRC Generic Letter No. 83-37, NUREG-0737 Technical Specifications, dated November 1, 1983.

Following the May 1979 accident at Three Mile Island Unit 2, many features were added to nuclear power plants to enhance the ability of the operator to manage accidents and transients. The RVLMS is one of these enhancements and serves the following purposes: 1) to provide corroborative information that the reactor core remains covered with coolant during anticipated operating occurrences and accidents; 2) to provide corroborative means of detecting the existence of a steam bubble in the upper reactor vessel head region during plant transients which result in reactor coolant system (RCS) depressurization; 3) to provide trending information to the plant operators relative to RCS inventory; and 4) to provide a corroborative indication of the approach to, and recovery from an inadequate core cooling (ICC) condition.

The proposed change adds the RVLMS to the technical specifications, to reflect its incorporation into the plant.

Safety Analysis

The proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas.

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The reactor vessel level monitoring system (RVLMS) is neither credited nor required in the mitigation of any previously evaluated accident and is not relied upon for reactor trip or initiation of any plant safety systems. Therefore, the proposed change does not affect the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Although the HJTCS has been utilized in the Emergency Procedures for corroboration of selected indications, no change to normal operating procedures is involved; thus no new path is created which may lead to a new or different kind of accident. The proposed change is intended solely to enhance the ability of the operator to manage accidents and transients by providing the operator with additional corroborative information.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The specific purpose of the proposed amendment is to enhance accident and transient monitoring capability and therefore increase the margin of safety.

   The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideation exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (ii) relates to a change that constitutes an additional limitation, restriction or control not presently included in the technical specifications. The proposed change is representative of Example (ii) in that it is an addition to the accident monitoring instrumentation required by the Nuclear Regulatory Commission's post-TMI-2 Action Plan.

<u>Safety and Significant Hazards Determination</u>

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident

previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and that (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19. Cold Leg HPSI Flow	1/cold leg	N.A.	20
20. Hot Leg HPSI Flow	1/hot leg	N.A.	20

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

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

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

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Cold Leg HPSI Flow	M	R
20. Hot Leg HPSI Flow	M	R

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within two hours following an earthquake ($\geq$ 0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at two and three hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

ATTACHMENT B

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19. Cold Leg HPSI Flow	1/cold leg	N.A.	20
20. Hot Leg HPSI Flow	1/hot leg	N.A.	20
21. Heated Junction Thermocouple System-Reactor Vessel Level Monitoring System*	2	1	22, 23

NOTES:

* A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, one sensor in the upper head and 3 sensors in the lower head, are OPERABLE.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

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

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

ACTION 22 - With the number of OPERABLE Channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 23 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore both channels of the system to OPERABLE status at the next scheduled refueling.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Cold Leg HPSI Flow	M	R
20. Hot Leg HPSI Flow	M	R
21. Heated Junction Thermocouple System-Reactor Vessel Level Monitoring System	M	R

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-5. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.5.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heated Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existance of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as reasonably possible. If both channels are inoperable, both channels shall be restored to OPERABLE status in the nearest refueling outage.

In the event that both HJTC channels are inoperable, existing plant instruments and operator training will be used as an alternate method of monitoring the reactor vessel inventory.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within 2 hours following an earthquake ($\geq$ 0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at 2 and 3 hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistant with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

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DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGE NPF-10/15-184

This is a request to revise Technical Specification 3/4.10.1, "Special Test Exceptions - Shutdown Margin."

Description

The proposed change revises Technical Specification (T.S.) 3/4.10.1, "Special Test Exceptions - Shutdown Margin." T.S. 3.10.1 allows shutdown margin to be reduced to less than the normal operating shutdown margin requirements during the performance of low power physics tests, provided that certain conditions are met. As one of these conditions, Surveillance Requirement 4.10.1.2 requires that all control element assemblies (CEA's) not fully inserted in the core be demonstrated to be capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing shutdown margin to less than the normal operating requirements. The proposed change will allow this surveillance to be performed within the past seven days instead of within the past 24 hours. This will enable low power physics testing to be completed without an additional trip to verify CEA insertability.

Low power physics tests are performed to verify core physics predictions. One of the test sequences measures CEA worths and may involve the reduction of shutdown margin as permitted by T.S. 3.10.1. Prior to initial criticality for performance of the low power physics tests, rod drop testing is performed to demonstrate CEA insertability. The reactor is brought critical and stabilized at the test plateau (approximately 10^{-2}% power). The most logical sequence for lower power physics testing has CEA worth measurements made last. Since approximately five days would have elapsed from when the hot rod drop tests were last performed, the reactor would have to be tripped again to demonstrate CEA insertion capability and satisfy the current 24 hour criteria. The proposed change will eliminate the necessity for an additional trip during low power physics test by requiring CEA insertability to be verified within seven days prior to reducing shutdown margin instead of within 24 hours.

Existing Technical Specifications:

Unit 2: See Attachment "A"
Unit 3: See Attachment "C"

Proposed Technical Specifications:

Unit 2: See Attachment "B"
Unit 3: See Attachment "D"

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of a previously evaluated accident?

   Response: No

   The previously analyzed accidents which potentially could be affected by the proposed change are those which involve overcooling of the reactor coolant system (RCS). Because of the negative moderator temperature coefficient, cooldown results in a reactivity increase. Because of this, a post trip return to power may be experienced during events involving overcooling of the RCS if insufficient negative reactivity is inserted via the CEA's. Calculation of shutdown margin excludes the contribution of the highest reactivity worth CEA which is assumed to remain fully withdrawn from the core on a reactor trip. Since shutdown margin is reduced during low power physics tests, surveillance 4.10.1.2 provides added assurance that the maximum amount of negative reactivity is available for insertion should an overcooling event occur. The proposed change may reduce the amount of assurance provided by this surveillance. To quantify the impact of the proposed change, a probabalistic study was conducted. The study considered the configuration of components utilized in CEA insertion and electronic/electrical failures that could result in a stuck CEA.

   Results of the study of the configuration of the components that are utilized in CEA insertion have shown that there would be no significant increase in the probability of stuck CEA as there is no significant change in the geometry of these components over a 7 day period of time. Components studied include the fuel assembly (including the effects of foreign material that could get trapped in the gap between the control rod and guide tube), CEA, extension shaft, CEDM (and scramability with respect to the proposed extended time period), and upper guide structure.

   Results of the study of electronic/electrical failures that could cause a CEA to be stuck show that failures of the feature that controls the movement of the CEA's is not time related. Power is required to engage or disengage grippers which control the movement of the CEA. Since the CEA's will always insert by gravity upon loss of power, the probability of a stuck CEA cannot be increased as a result of an electrical malfunction.

   Based on the probabilistic analysis, the probability of an RCS overcooling event with a stuck CEA increases from 5.0×10^{-8} to 2.3×10^{-7} when the requirement for trip testing is extended from 24

hours to 7 days. Although this scenario is not a core melt scenario, for comparison purposes it should be noted that the change in probability, 1.8×10^{-7}, is approximately three orders of magnitude below the core melt safety goal proposed by the NRC in NUREG-0880. This respresents a negligible increase to the risk to the public. Therefore, the proposed change does not significantly increase the probability or consequences of previously evaluated accidents.

2. Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   There is no possibility of a new or different kind of accident occurring since the FSAR accident analysis already assumes a hypothetical stuck CEA, and the proposed amendment does not result in any change to the facility.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   Operation of the facility in accordance with this proposed amendment does not involve a significant reduction in a margin of safety since the change does not significantly affect the probability or consequences of any previously evaluated accident.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP): for example a change resulting from the application of a small refinement of a previously used calculational model or design method.

In this case, SRP Section 14.2, "Initial Test Program" and SRP Sections 15.1.1, 15.1.2, 15.1.3, 15.1.4 and 15.1.5 which relate to Reactor Coolant System (RCS) overcooling events provide the pertinent acceptance criteria. SRP Section 14.2 refers to Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Nuclear Power Plants." R.G. 1.68 outlines the elements of an acceptable startup test program including requirements for CEA worth measurements during low power physics testing. The proposed change will facilitate CEA worth measurements and is consistent with R.G. 1.68 and SRP Section 14.2.

The proposed change does not affect the consequences of RCS overcooling events evaluated in accordance with SRP Section 15.1.1 through 15.1.5. Because of a negative moderator temperature coefficient, RCS overcooling results in a reactivity increase. Because of this, a post trip return to power may be experienced in overcooling events if insufficient negative reactivity is inserted via the CEA's. Since shutdown margin is reduced during CEA worth measurements, T.S. 4.10.1.2 provides added assurance that all CEA's are trippable. By increasing the period during which shutdown margin may be reduced following performance of surveillance requirement 4.10.1.2, the proposed change may result in an insignificant reduction in the assurance provided. The resultant increase in the probability of a stuck CEA coincident with an overcooling event has been calculated to be 1.8×10^{-7}. The proposed change has no effect on the consequences of overcooling events since it does not affect the amount by which shutdown margin may be reduced. Because the consequences of these events are not increased, the SRP acceptance criteria continue to be satisfied.

The proposed change satisfies the SRP acceptance criteria and therefore is similar to example (vi).

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:3746F

ATTACHMENT "A"

UNIT 2 EXISTING SPECIFICATION

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

*Operation in MODE 3 shall be limited to 6 consecutive hours.

ATTACHMENT "B"

UNIT 2 PROPOSED SPECIFICATION

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEA's fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

ATTACHMENT "C"

UNIT 3 EXISTING SPECIFICATION

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

ATTACHMENT "D"

UNIT 3 PROPOSED SPECIFICATION

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEA's fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

DESCRIPTION OF PROPOSED CHANGES
NPF-10/15-188 AND SAFETY ANALYSIS

This is a request to revise Section 3/4.3.2, Engineered Safety Feature Actuation System, of the Technical Specifications for San Onofre Nuclear Generating Station, Units 2 and 3.

Description

Technical Specification 3/4.3.2 requires that Engineered Safety Features Actuation System (ESFAS) instrumentation channels be operable and defines a number of functional tests and response time tests that must be conducted periodically in order to verify operability. Specification 3/4.3.2 identifies the instruments required for the Toxic Gas Isolation System (TGIS). The TGIS is actuated when concentrations of toxic gases (e.g. chlorine, butane/propane or ammonia) in the control room supply ducts exceed the concentration set points. Upon receipt of a TGIS signal, the control room heating ventilation and air conditioning (HVAC) system is automatically isolated. The setpoints are selected such that the toxic gas concentration in the control room will not exceed allowable limits during the first two minutes after the detector responds. This provides adequate protection for the control room operators by allowing sufficient time to don protective gear.

Recently, Amendments 29 and 18 to the operating licenses for San Onofre Nuclear Generating Station Unit 2 and 3, respectively, deleted the requirement for TGIS carbon dioxide instrumentation. Analysis demonstrated that the maximum control room concentration of carbon dioxide at any time without control room isolation would be 11,000 ppm. Deletion of carbon dioxide instrumentation was permitted since the protective action limit of 50,000 ppm for carbon dioxide would never be exceeded. The license amendments inadvertently did not delete all references to carbon dioxide instrumentation from the technical specifications. The license amendments only deleted the reference to the carbon dioxide instrumentaton from Table 3.3-4, "ESFAS Instrumentation Trip Values." The TGIS carbon dioxide instrumentation is also included in Table 3.3-3, "ESFAS Instrumentation," and Table 4.3-2, "ESFAS Instrumentation Surveillance Requirements". The proposed change corrects this oversight by deleting the remaining references to the TGIS carbon dioxide instrumentation from these tables.

Existing Technical Specifications

Unit 2 - See Attachment A

Unit 3 - See Attachment B

Proposed Technical Specifications

Unit 2 - See Attachment C

Unit 3 - See Attachment D

Safety Analysis

The proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Deletion of the setpoint for carbon dioxide was evaluated in the Safety Evaluation for Amendment 29 to the Unit 2 Operating License and Amendment 18 to the Unit 3 Operating License. Operation of the facility in accordance with this proposed change will not involve an increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Deletion of the setpoint for carbon dioxide was evaluated in the Safety Evaluation for Amendment 29 to the Unit 2 Operating License and Amendment 18 to the Unit 3 Operating License. Operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   Deletion of the setpoint for carbon dioxide was evaluated in the safety evaluation for Amendment 29 to the Unit 2 Operating License and Amendment 18 to the Unit 3 Operating License. Operation of the facility in accordance with this proposed amendment will not involve a reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to a purely adminstrative change to the technical specifications: for example a change to achieve consistency throughout the technical specifications, correction of an error or a change in nomenclature. The proposed change described above, by deleting the remaining references to TGIS carbon dioxide instrumentation, which should have been removed by Amendments 29 and 18,

corrects this oversight and achieves consistency within the technical specifications. Therefore the proposed change is similar to example (i) and thus it is proposed that this change does not involve significant hazards considerations.

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:3563F

ATTACHMENT A

EXISTING SPECIFICATION - UNIT 2

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. CONTROL ROOM ISOLATION (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13*//
b. Manual SIAS (Trip Buttons)	2 sets of 2/unit	1 set of 2	2 sets of 2/unit	1, 2, 3, 4	8
c. Airborne Radiation					
i. Particulate/Iodine	2	1	1	All	13*//
ii. Gaseous	2	1	1	All	13*//
d. Automatic Actuation Logic	1/train	1	1	All	13*//
10. TOXIC GAS ISOLATION (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*//, 15*//
b. Chlorine - High	2	1	1	All	14*//, 15*//
c. Ammonia - High	2	1	1	All	14*//, 15*//
d. Butane/Propane - High	2	1	1	All	14*//, 15*//
e. Carbon Dioxide - High	2	1	1	All	14*//, 15*//
f. Automatic Actuation Logic	1/train	1	1	All	14*//, 15*//

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to A.C. motor-operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-actuated valves except MSIVs, MFIVs, and CCW Valves 2HV-6211 and 2HV-6216.

4a. CCW noncritical loop isolation Valves 2HV-6212, 2HV-6213, 2HV-6218, and 2HV-6219 close.

4b. Containment emergency cooler CCW isolation Valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	R	R	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Carbon Dioxide - High	S	R	M	All
f. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

ATTACHMENT B

EXISTING SPECIFICATION - UNIT 3

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. CONTROL ROOM ISOLATION (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13*#
b. Manual SIAS (Trip Buttons)	2 sets of 2/unit	1 set of 2	2 sets of 2/unit	1, 2, 3, 4	8
c. Airborne Radiation					
i. Particulate/Iodine	2	1	1	All	13*#
ii. Gaseous	2	1	1	All	13*#
d. Automatic Actuation Logic	1/train	1	1	All	13*#
10. TOXIC GAS ISOLATION (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*#, 15*#
b. Chlorine - High	2	1	1	All	14*#, 15*#
c. Ammonia - High	2	1	1	All	14*#, 15*#
d. Butane/Propane - High	2	1	1	All	14*#, 15*#
e. Carbon Dioxide - High	2	1	1	All	14*#, 15*#
f. Automatic Actuation Logic	1/train	1	1	All	14*#, 15*#

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs and CCW valves 3HV-6211 and 3HV-6216.

4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 seconds) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 seconds) is included.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	R	R	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Carbon Dioxide - High	S	R	M	All
f. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

ATTACHMENT C

PROPOSED SPECIFICATION - UNIT 2

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. CONTROL ROOM ISOLATION (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13*//
b. Manual SIAS (Trip Buttons)	2 sets of 2/unit	1 set of 2	2 sets of 2/unit	1, 2, 3, 4	8
c. Airborne Radiation					
i. Particulate/Iodine	2	1	1	All	13*//
ii. Gaseous	2	1	1	All	13*//
d. Automatic Actuation Logic	1/train	1	1	All	13*//
10. TOXIC GAS ISOLATION (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*//, 15*//
b. Chlorine - High	2	1	1	All	14*//, 15*//
c. Ammonia - High	2	1	1	All	14*//, 15*//
d. Butane/Propane - High	2	1	1	All	14*//, 15*//
e. Automatic Actuation Logic	1/train	1	1	All	14*//, 15*//

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
15. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
16. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to A.C. motor-operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-actuated valves except MSIVs, MFIVs, and CCW Valves 2HV-6211 and 2HV-6216.

4a. CCW noncritical loop isolation Valves 2HV-6212, 2HV-6213, 2HV-6218, and 2HV-6219 close.

4b. Containment emergency cooler CCW isolation Valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	R	R	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

ATTACHMENT D

PROPOSED SPECIFICATION - UNIT 3

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. CONTROL ROOM ISOLATION (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13*#
b. Manual SIAS (Trip Buttons)	2 sets of 2/unit	1 set of 2	2 sets of 2/unit	1, 2, 3, 4	8
c. Airborne Radiation					
i. Particulate/Iodine	2	1	1	All	13*#
ii. Gaseous	2	1	1	All	13*#
d. Automatic Actuation Logic	1/train	1	1	All	13*#
10. TOXIC GAS ISOLATION (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*#, 15*#
b. Chlorine - High	2	1	1	All	14*#, 15*#
c. Ammonia - High	2	1	1	All	14*#, 15*#
d. Butane/Propane - High	2	1	1	All	14*#, 15*#
e. Automatic Actuation Logic	1/train	1	1	All	14*#, 15*#

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
15. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
16. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs and CCW valves 3HV-6211 and 3HV-6216.

4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 seconds) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 seconds) is included.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	R	R	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

Description and Safety Analysis of Proposed Change NPF-10/15-189

This is a request to revise Technical Specification 3/4.5.2, "Emergency Core Cooling Systems" and Technical Specification 3.3.2, Table 3.3-5, "Engineered Safety Features Actuation System Instrumentation Response Times" for San Onofre Nuclear Generating Station Units 2 and 3.

Existing Specifications

Unit 2: See Attachment A
Unit 3: See Attachment C

Proposed Specifications

Unit 2: See Attachment B
Unit 3: See Attachment D

Description

The proposed change revises Technical Specifications (T.S.) 3/4.5.2, "Emergency Core Cooling Systems (ECCS)," and T.S. 3.3.2, Table 3.3-5 "Engineered Safety Features Actuation System (ESFAS) Instrumentation Response Times." T.S. 3/4.5.2 requires that two independent ECCS subsystems be operable and specifies periodic surveillance tests to verify ECCS operability and defines the actions to be taken when the minimum operability requirements are not met. T.S. 3.3.2 Table 3.3-5 specifies maximum acceptable response times for engineered safety features (ESF) which must be demonstrated during surveillance testing. Verification of response times ensures the ESF equipment will be actuated within the times assumed in the accident analyses. The proposed change will reflect the reinstatement of automatic closure of the ECCS miniflow valves on a recirculation actuation signal (RAS).

The ECCS is designed to mitigate the consequences of a loss of coolant accident (LOCA). On detection of a LOCA, the ECCS is automatically actuated by a safety injection actuation signal (SIAS) and maintains core cooling by pumping water into the reactor coolant system, initially from the refueling water storage tank (RWST). Water spilling from the break in the RCS accumulates on the containment floor. On low level in the RWST, a RAS is generated, realigning the ECCS pumps to take suction from the containment sump, establishing recirculation.

In small break LOCA's, RCS pressure may remain higher than the maximum pressure developed by the high pressure safety injection (HPSI) pumps following ECCS actuation. Damage to the HPSI pumps would result after a relatively short period in this condition if a minimum flow is not maintained through the pumps. To prevent HPSI pump damage, minimum flow is guaranteed by the ECCS miniflow lines from the ECCS pump discharge to the RWST. It is desirable to close the ECCS miniflow lines following initiation of recirculation to prevent radioactive water from being pumped from the containment sump to the RWST. The RWST is vented to atmosphere creating a potential release path.

Originally, the ECCS miniflow valves were closed automatically on a RAS generated from low RWST level. Following an event in December, 1982 involving simultaneous SIAS and RAS (i.e., ECCS pumps started and ECCS miniflow valves closed), RAS was removed from the miniflow valves to preclude damage to the ECCS pumps. Currently, closure of the ECCS miniflow valves is manually initiated by the operator. A design change (DCP 6234) is being implemented to restore automatic ECCS miniflow valve closure. With the design change, both low RWST level (RAS) and high containment sump level will be required for automatic closure of the ECCS miniflow valves. Conditioning ECCS miniflow valve closure on RAS and high sump level will preclude an event involving simultaneous SIAS and RAS from damaging the ECCS pumps.

To reflect this design change, the following changes to the technical specifications are proposed:

1) T.S. 3.3.2, Table 3.3-5 specifies response times for ESF equipment. The proposed change will add the ECCS miniflow isolation valves to the equipment included in Table 3.3-5 as actuated by a recirculation actuation signal. A response time of 50.7 seconds is specified which includes an allowance for diesel generator starting and load sequencing. A note is added to indicate that the closure of the ECCS miniflow valves on a RAS is conditioned by high containment sump level.

2) T.S. 3/4.5.2 operability and surveillance testing requirements for the ECCS. One of the surveillance tests (T.S. 4.5.2.e.3) requires verification that ECCS miniflow valves close within a specified period of time (currently 50.7 seconds for Unit 3; 40.7 seconds for Unit 2) upon manual actuation from the control room. The proposed change will require verification that the ECCS miniflow isolation valves close automatically on a RAS test signal coincident with a containment sump level high signal. The required response time is specified in Table 3.3-5.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   The proposed change affects the ECCS which is designed to mitigate the consequences of loss of coolant accidents which have been previously evaluated. Currently the ECCS miniflow valves are closed manually by the operator following a RAS. During a LOCA, radioactively contaminated

water from the containment sump may be pumped back to the RWST via the ECCS miniflow lines following RAS until the operator closes the ECCS miniflow isolation valves. The RWST is vented to atmosphere creating a potential release path. The proposed change will prevent pumping of radioactively contaminated water into the RWST by requiring automatic isolation of the ECCS miniflow lines. Therefore, the proposed change will result in a decrease in the consequences of previously evaluated LOCA's.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The ECCS miniflow isolation valves are required to be closed following a RAS. The addition of automatic closure of ECCS miniflow isolation valves does not alter this requirement, but changes the method of initiation from manual to automatic. The addition of the high containment sump level permissive precludes spurious closure of the ECCS miniflow isolation valves. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

3. Will operation of the facility in accordance with this change involve a significant reduction in a margin of safety.

   Response: No

   The proposed change reflects the addition of automatic closure of the ECCS miniflow isolation. As noted above, this addition reduces the potential for pumping of radioactively contaminated water into the RWST during a LOCA. This results in the reduction of potential consequences of a LOCA. Therefore, no margin of safety is reduced.

The commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazard considerations. Example (ii) relates to a change that constitutes an additional limitation, restriction or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.

The current technical specification requires verification that the ECCS miniflow isolation valves close within a specified time (50.7 seconds for Unit 3, 40.7 seconds for Unit 2) following manual actuation. The proposed change will require that the valves close automatically within 50.7 seconds on a RAS coincident with high containment sump level. The requirement to automatically close within 50.7 of a RAS is more restrictive since the existing specification does not require automatic closure and does not define a closure time relative to the occurrence of a RAS. Therefore, the

proposed change constitutes additional limitations not currently in the technical specifications. The proposed change is similar to example (ii) of 48 FR 14870 and does not involve significant hazards considerations.

Safety and Significant Hazards Determination

Based on the above safety analysis, it is concluded that: 1) the proposed change does not involve significant hazards considerations as defined by 10 CFR 50.92; 2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and, 3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT "A"

UNIT 2 EXISTING SPECIFICATION

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

SEP 21 1984

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to A.C. motor-operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-actuated valves except MSIVs, MFIVs, and CCW Valves 2HV-6211 and 2HV-6216.

4a. CCW noncritical loop isolation Valves 2HV-6212, 2HV-6213, 2HV-6218, and 2HV-6219 close.

4b. Containment emergency cooler CCW isolation Valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

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

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

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

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

a. High-Pressure Safety Injection pump.

b. Low-Pressure Safety Injection pump.

c. Charging pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open; and that, upon manual test initiation from the control room, all the recirculation valves to the refueling water tank close within 40.7 seconds.

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

1. High-Pressure Safety Injection pumps developed head, at an indicated flow rate of 650 gpm, greater than or equal to 2142 feet for P017, 2101 feet for P018 and 2103 for P019.

2. Low-Pressure Safety Injection pump developed head greater than or equal to 406.1 feet.

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

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

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

a. The sum of the injection lines flow rates, excluding the highest flow rate, is greater than or equal to 657 gpm for P017 running, 667 gpm for P018 running and 672 gpm for P019 running, and

b. The total pump flow rate is greater than or equal to 900 gpm for P017 running, 913 gpm for P018 running

ATTACHMENT "C"

UNIT 3 EXISTING SPECIFICATION

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV5054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kV Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

SEP 2 1 1984

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs and CCW valves 3HV-6211 and 3HV-6216.

4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 seconds) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 seconds) is included.

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

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

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

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

a. High-Pressure Safety Injection pump.

b. Low-Pressure Safety Injection pump.

c. Charging pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open; that, upon manual test initiation from the control room, all the recirculation valves to the refueling water tank close within 50.7 seconds.

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

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

2. Low-Pressure Safety Injection pump developed head greater than or equal to 396 feet at miniflow.

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

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

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

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

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

ATTACHMENT "B"

UNIT 2 PROPOSED SPECIFICATION

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Isolation Valves Close	50.7* (Note 7)
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

SEP 21 1984

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to A.C. motor-operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-actuated valves except MSIVs, MFIVs, and CCW Valves 2HV-6211 and 2HV-6216.

4a. CCW noncritical loop isolation Valves 2HV-6212, 2HV-6213, 2HV-6218, and 2HV-6219 close.

4b. Containment emergency cooler CCW isolation Valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.
7. Prior to completion of DCP 6234, valve closure is manually initiated. Following completion of DCP 6234, valves are to close automatically on a RAS coincident with a high-high containment sump signal.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

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

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

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

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

a. High-Pressure Safety Injection pump.

b. Low-Pressure Safety Injection pump.

c. Charging pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open; and that on a RAS test signal coincident with a high-high containment sump test signal, all the recirculation valves to the refueling water tank close.

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

1. High-Pressure Safety Injection pumps developed head, at an indicated flow rate of 650 gpm, greater than or equal to 2142 feet for P017, 2101 feet for P018 and 2103 for P019.

2. Low-Pressure Safety Injection pump developed head greater than or equal to 406.1 feet.

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

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

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

a. The sum of the injection lines flow rates, excluding the highest flow rate, is greater than or equal to 657 gpm for P017 running, 667 gpm for P018 running and 672 gpm for P019 running, and

b. The total pump flow rate is greater than or equal to 900 gpm for P017 running, 913 gpm for P018 running and 918 gpm for P019 [illegible]

ATTACHMENT "D"

UNIT 3 PROPOSED SPECIFICATION

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV5054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Isolation Valves Close	50.7* (Note 7)
7. 4.16 kV Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

SEP 2 1 1984

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs and CCW valves 3HV-6211 and 3HV-6216.

4a. CCW non-critical-loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.
7. Prior to completion of DCP 6234, valve closure is manually initiated. Following completion of DCP 6234 valves are to close automatically on a RAS coincident with a high-high containment sump signal.

* Emergency diesel generator starting delay (10 seconds) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 seconds) is included.

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

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

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

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

a. High-Pressure Safety Injection pump.

b. Low-Pressure Safety Injection pump.

c. Charging pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open; and that on a RAS test signal coincident with a high-high containment sump test signal, all the recirculation valves to the refueling water tank close.

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

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

2. Low-Pressure Safety Injection pump developed head greater than or equal to 396 feet at miniflow.

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

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

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

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

b. The total pump flow rate is greater than or equal to 882 gpm for P017 running, 894 gpm for P018 running,

DESCRIPTION AND SAFETY ANALYSIS OF
PROPOSED CHANGE NPF-10/15-193

This is a request to revise Technical Specification 3/4.9.12, "Fuel Handling Building Post-Accident Cleanup Filter System."

Description

The proposed change would revise Technical Specification (T. S.) 3/4.9.12, "Fuel Handling Building Post-Accident Cleanup Filter System." T. S. 3/4.9.12 requires the operability of the two independent fuel handling building post-accident cleanup filter systems. The purpose of the fuel handling building post-accident cleanup filter system is to ensure that any radioactive material released from an irradiated fuel assembly after a fuel handling accident will be filtered through the HEPA filters and charcoal adsorbers. The action required by T. S. 3/4.9.12 if one of the two filter systems becomes inoperable, is to restore the inoperable system to operable status within 7 days or suspend operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool. The proposed change would revise the action in the event of one of the two filter systems becoming inoperable to require that the remaining filter system be demonstrated operable within 12 hours and at least once every seven days thereafter by starting the system and operating it for a minimum of 15 minutes.

Existing Technical Specifications:

Unit 2: See Attachment "A"
Unit 3: See Attachment "B"

Proposed Technical Specifications:

Unit 2: See Attachment "C"
Unit 3: See Attachment "D"

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of a previously evaluated accident?

Safety Analysis (Continued)

Response: No

The fuel handling building post-accident cleanup filter system has no effect on the probability of an accident. As described in the Final Safety Analysis Report (FSAR) Section 15.7.3.4, the consequences of a postulated fuel handling accident are well within the criteria of 10 CFR 100 even if the fuel handling building post-accident cleanup filter system is not taken into account. The remaining one hundred percent capacity fuel handling building post-accident cleanup filter system will still be available and will be demonstrated to be operable. Thus there is no significant increase in the consequences of a previously evaluated accident.

2. Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The fuel handling building post-accident cleanup filter system serves as an accident mitigation system. This proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The remaining one hundred percent capacity fuel handling building post-accident cleanup filter system will still be available and will be demonstrated to be operable every 7 days. Thus this proposed amendment will not involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of the standards for determiniing whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP).

In this case, SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," provides the pertinent acceptance criteria. SRP Section 15.7.4, part II provides that the plant site and dose mitigating Engineered Safety Features systems are acceptable with respect to the radiological consequences of a postulated fuel handling accident if the calculated whole body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. "Well within" means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 Rem for the thyroid and 6 Rem for the whole body doses.

Safety Analysis (Continued)

Table 15.7-6 of the FSAR indicates that if 236 fuel rods are assumed to be damaged in a fuel handling accident with no filtration assumed in the fuel handling building, the dose rates would be 74.3 Rem thyroid and 0.71 Rem whole body at the exclusion area boundary and 2.1 Rem thyroid and 0.02 Rem whole body at the low population zone boundary.

Thus requiring one fuel handling building post-accident cleanup filter system to remain operable and be demonstrated operable clearly meets the acceptance criteria of the SRP and the proposed change is similar to example (vi).

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environment Statement.

ATTACHMENT A

EXISTING SPECIFICATION - UNIT 2

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling building post-accident cleanup filter system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel handling building post-accident cleanup filter system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling building post-accident cleanup filter systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

ATTACHMENT B

EXISTING SPECIFICATION - UNIT 3

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling building post-accident cleanup filter system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel handling building post-accident cleanup filter system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling building post-accident cleanup filter systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

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ATTACHMENT C

PROPOSED SPECIFICATION - UNIT 2

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling buildling post-accident cleanup filter system is capable of being powered from an OPERABLE emergency power source. Demonstrate the operability of the OPERABLE fuel handling building post-accident cleanup filter system within 12 hours and at least once per 7 days thereafter by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operat for at least 15 minutes.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling building post-accident cleanup filter system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filters or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

ATTACHMENT D

PROPOSED SPECIFICATION - UNIT 3

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling buildling post-accident cleanup filter system is capable of being powered from an OPERABLE emergency power source. Demonstrate the operability of the OPERABLE fuel handling building post-accident cleanup filter system within 12 hours and at least once per 7 days thereafter by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operat for at least 15 minutes.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling building post-accident cleanup filter system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filters or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-194

This is a request to revise License Condition 2.E.

Existing License Conditions:

Unit 2: See Attachment A
Unit 3: See Attachment B

Proposed License Conditions:

Unit 2: See Attachment C
Unit 3: See Attachment D

Description

The proposed change would revise License Condition 2.E. The purpose of License Condition 2.E. is to maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made by the licensee without prior NRC approval pursuant to the authority of 10CFR50.54(p). 10CFR50.54(p) permits licensees to make changes to the Commission-approved security plan which do not involve a reduction in the plan effectiveness without prior NRC approval.

Changes to the security plan which may in some way be construed to reduce the effectiveness of the plan require prior NRC approval via a license amendment request submitted in accordance with 10CFR50.90.

The proposed change would revise license condition 2E to reflect proposed Revision 6 to the August 1983 Physical Security Plan (PSP). The revised portion of the August 1983 Physical Security Plan relates to alarm annunciation requirements. Detail of the proposed revision are safeguards information and are being withheld from public disclosure pursuant to 10CFR73.21.

Safety Analysis

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change affects only the Physical Security Plan and is not related to any accident previously evaluated. The proposed change relates to alarm annunciation requirements as defined by

10CFR73.55 (e)(1). 10CFR73.55 (e)(1) requires that intrusion detection alarms annunciate audibly and visually in a central alarm station and at least one other continuously manned station. The proposed change will resolve an apparent inconsistency with the plan and 10CFR73.55(e)(1) which occurred as a result of judgements and decisions made during the development of the plan responsive to the upgrade requirements of 10CFR73.55 of late 1977. The existing configuration, which was already alarmed to annunciate in another location was not considered to be a security access portal. The configuration is used only during major maintenance, and can only be opened from the interior of a vital area, entry to which is controlled by intrusion alarmed card-reader doors.

Since this configuration provided high assurance against undetected access into a vital area, rewiring to a central alarm station or other continuously manned station was not considered necessary. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   The proposed change to the Physical Security Plan does not alter any safety related design bases of the facility or its operation. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change to the Physical Security Plan does not involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48FR14870) of amendments that are considered not likely to involve significant hazards considerations. These examples are not applicable to the proposed revision to the August 1983 Physical Security Plan.

Based on the above responses to the three standards of 10CFR50.92(c), it can be concluded that the standards are met with a no significant hazards consideration determination.

The proposed change described above, will revise License Condition 2.E to reflect the 10CFR50.90 submittal of Revision 6 to the August 1983 Physical Security Plan.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10CFR50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

TJM:4187F

ATTACHMENT A
(Existing License Condition)

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

E. The Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain 10 CFR 73.21 information are collectively entitled: "San Onofre Nuclear Generating Station Units 1, 2, and 3 Physical Security Plan," dated August 1983 (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983 (transmitted by letter dated December 16, 1983), Revision 1, dated December 1983 (transmitted by letter dated December 16, 1983), and Revision 1A, dated April 1984 (transmitted by letter dated April 2, 1984); the Safeguards Contingency Plan (Chapter 8 of the March 1981 Physical Security Plan*); and the "Guard Training and Qualification Plan, San Onofre Nuclear Generating Station Units 1, 2, and 3," dated August 13, 1979 as revised September 3, 1980, and December 15, 1981.

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

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

ATTACHMENT B
(Existing License Condition)

E. The Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain 10 CFR 73.21 information are collectively entitled: "San Onofre Nuclear Generating Station Units 1, 2 and 3 Physical Security Plan," dated August 1983, (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983, (transmitted by letter dated December 16, 1983), and Revision 1A, dated April 1984 (transmitted by letter dated April 2, 1984); the Safeguards Contingency Plan (Chapter 8 of the March 1981 Physical Security Plan*); and the "Guard Training and Qualification Plan, San Onofre Nuclear Generating Station Units 1, 2 and 3," dated August 13, 1979 as revised September 3, 1980 and December 15, 1981.

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

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

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

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

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

J. This license is effective as of the date of issuance and shall expire at midnight on October 18, 2013.

ATTACHMENT C
(Proposed License Condition)

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

E. The Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain 10 CFR 73.21 information are collectively entitled: "San Onofre Nuclear Generating Station Units 1, 2, and 3 Physical Security Plan," dated August 1983 (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983 (transmitted by letter dated December 16, 1983), and Revision 6, dated June 1985 (transmitted by letter dated June 7, 1985);

the Safeguards Contingency Plan (Chapter 8 of the March 1981 Physical Security Plan*); and the "Guard Training and Qualification Plan, San Onofre Nuclear Generating Station Units 1, 2, and 3," dated August 13, 1979 as revised September 3, 1980, and December 15, 1981.

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

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

ATTACHMENT D
(Proposed License Condition)

E. The Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain 10 CFR 73.21 information are collectively entitled: "San Onofre Nuclear Generating Station Units 1, 2 and 3 Physical Security Plan," dated August 1983, (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983, (transmitted by letter dated December 16, 1983), and Revision 6, dated June 1985 (transmitted by letter dated June 7, 1985); the Safeguards Contingency Plan (Chapter 8 of the March 1981 Physical Security Plan*); and the "Guard Training and Qualification Plan, San Onofre Nuclear Generating Station Units 1, 2 and 3," dated August 13, 1979 as revised September 3, 1980 and December 15, 1981.

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

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

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

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

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

J. This license is effective as of the date of issuance and shall expire at midnight on October 18, 2013.