SOUTHERN CALIFORNIA
EDISON®
An EDISON INTERNATIONAL® Company

**A. Edward Scherer**
Director
Nuclear Regulatory Affairs

June 10, 2009

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **Docket Nos. 50-361, 50-362, and 72-41**
**Facility and Commitment Change Reports**
**San Onofre Nuclear Generating Station Units 2 and 3**
**and the Independent Spent Fuel Storage Installation**

Dear Sir or Madam:

This letter transmits the Facility Change Reports required by 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2) for San Onofre Nuclear Generating Station Units 2 and 3 for the period from March 1, 2007 through December 18, 2008. Enclosure 1, San Onofre Nuclear Generating Station Units 2 and 3 Facility Change Report, provides a summary of the facility changes and procedure changes, including a summary of the safety evaluations performed for each change. There were no tests or experiments during this period. The report scope is based on a review of plant records and all 50.59 evaluations identified for the time period above. Complete facility change documentation is available onsite. There were no 72.48 evaluations performed by Southern California Edison (SCE) during the time period above.

Enclosure 2 provides a report on commitment changes made per Nuclear Energy Institute (NEI) "Guidelines for Managing NRC Commitment Changes," NEI-99-04, Revision 0, for the March 1, 2007 through December 18, 2008 period.

If you would like any additional information, please contact Ms. Linda T. Conklin at (949) 368-9443.

Sincerely,

Enclosures: As stated

cc: E. E. Collins, Regional Administrator, NRC Region IV
R. Hall, NRC Project Manager, San Onofre Units 2, and 3
G. G. Warnick, NRC Senior Resident Inspector, San Onofre Units 2 & 3
J. C. Shepherd, NRC Project Manager, San Onofre Unit 1

IE47
NMSS01

P.O. Box 128
San Clemente, CA 92674

ENCLOSURE 1

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

FACILITY CHANGE REPORT (FCR)

10CFR50.59 EVALUATION SUMMARIES

FOR THE PERIOD
FROM MARCH 1, 2007 THROUGH DECEMBER 18, 2008

**AR 031101386-43, Abandoning Condensate Storage Tank (CST) 2(3) T121 Vault Space Heaters**

**Description:**

Engineering Change Packages (ECPs) 031101386-24 and -25 allow abandoning the CST, 2(3) T121 space heaters in-place. Calculation M-0050-021 shows the CST water temperature will be maintained above 40° F using Auxiliary Feedwater (AFW) Pump heat. An existing system will continue to monitor the CST water temperature and provide the Control Room with a low temperature alarm. There will be no impact on the instruments located in the CST vault area, since the instruments do not require the heaters to perform their function.

The CST, 2(3) T121 space heaters (E542, E543, E636, & E637) and the vault area temperature control system will be abandoned in-place. After the heaters are abandoned, the tank water temperature can be maintained using the Auxiliary Feedwater (AFW) Pump heat with the mini-flow provisions. Performing maintenance on these heaters is difficult and safety issues exist due to scaffolding and the confined space area. By abandoning these heaters in-place, maintenance workload, and PM frequency for the T121 vault is reduced.

Abandoning the CST T121 Vault Space Heaters in-place is adverse because it substitutes the manual operation of AFW pumps to maintain a minimum temperature in CST T121 in lieu of the automatic function.

**Evaluation Summary:**

The evaluation identified that AFW mini-flow operation provides sufficient heat to keep CST water temperature above 40° F should the vault temperature drop to 35° F. Analysis (Calculation M-0050-021, "Heat Loss from T-121 - Condensate Storage Tank(s) - Units 2&3") determined it would take 86 hours for the water temperature to drop from 46° F to 40° F. This period provides sufficient time for operator response to the control room alarm that occurs due to low CST water temperature. Furthermore, the AFW Pumps and their source of power are safety-related and 1E. Therefore, using the mini-flow discharge of these pumps is a reliable method of maintaining CST water temperature.

The proposed activity does not introduce the possibility of a change in the frequency of an accident, the consequences of an accident or malfunction, the possibility of a new accident, or the likelihood of a malfunction because the method of maintaining the water temperature in CST T121 is not an initiator of any accident and no new failure modes are introduced. The activity has no impact to any fission product barriers nor are there any methods of evaluation associated with this equipment. Therefore, the change may proceed without need for NRC approval.

**AR 040301925-07, Engineering Change Package (ECP) 040301925-02, Dew Point Sensor (2/3AE8075) Replacement**

**Description:**

ECP 040301925-02 replaces the San Onofre Nuclear Generating Station (SONGS) dew point sensor (2/3AE8075) on the Meteorological (Met) Tower with a humidity sensor. The humidity sensor readings will be converted to dew point at the Met Tower and then distributed to Emergency Response Data System (ERDS) and to the control room hallway recorder. The new detector's operation is more reliable in the salt air environment and will require less corrective maintenance while performing the same function. An evaluation was required because the new sensor operates in a different manner than the device described in the Updated Final Safety Analysis Report (UFSAR). The Met Tower dew point sensor and related documentation is changed to reflect the different method for determining the dew point; UFSAR section 2.3.3.2.7 is updated and associated tables to match the new type of dew point sensor.

**Evaluation Summary:**

This change does not require NRC approval prior to its implementation based on the negative responses to 50.59 criteria (i) through (viii). The responses conclude the proposed activity does not cause a change in the frequency or likelihood of an accident or malfunction; there are no changes to accident or malfunction consequences; and there are no changes to system parameters. These conclusions are based on the fact that the sensor is not an initiator of any accident or malfunction and no new failure modes are introduced. Therefore, the change may proceed without need for NRC approval.

**051000729-78, Replacing Chemical & Volume Control System (CVCS) Preset Counters on Control Panels 2(3) CR058 with Equivalent Digital Instruments**

**Description:**

Engineering Change Packages (ECPs) 051000729-57 and 051000729-58 replace four existing Chemical & Volume Control System (CVCS) electromechanical preset counters on control room panels 2(3) CR058 with new, more reliable, digital electronic counters. The electromechanical preset counters have experienced high failure rates. Specific changes include the replacement of primary makeup water (PMW) batch counters 2(3) FQIS0210X and boric acid makeup (BAMU) batch counters 2(3) FQIS0210Y.

The replacement counter model utilizes a microprocessor with firmware to accomplish its design function and requires a new power input to power the device's internal digital circuitry. The instruments support the CVCS function to control Reactor Coolant System (RCS) boric acid concentration and the volume control tank (VCT) for reactor reactivity control. Because of the internal electronics power needs, the replacement counter's response to a loss of count circuit control power differs from the pulsed-voltage input signal used by the existing counter to perform its design function. The required additional power source for the new counters was determined to be adverse. The new Otek microprocessor-based counters have different failure modes than the existing Durant electromechanical relay-based counters, and the Otek counter model's response to a loss of count circuit control power differs from that of the Durant counter model.

**Evaluation Summary:**

The evaluation identified the inadvertent boron dilution accident as the only UFSAR-analyzed accident that could credibly be affected by the proposed activity. UFSAR section 15.4.1.4 currently analyzes this moderate frequency event for plant Modes 1 through 6. The evaluation concluded that the current UFSAR accident analysis for the inadvertent boron dilution accident remains bounding and that the change does not introduce the possibility of any new type of accident. Although differences exist in the failure modes of the proposed digital electronic counter model and the existing electromechanical counter model, the evaluation determined that the results of the failure modes of the replacement counters are bounded by the failure modes of the existing counters. The batch counters are not considered malfunction initiators nor are there any consequences associated with this equipment. The activity has no impact to any fission product barriers nor are there any methods of evaluation associated with this equipment. Therefore, the change may proceed without need for NRC approval.

## 060700747-18, Emergency Core Cooling System (ECCS) Void and Water Hammer Evaluation

**Description:**

The High Pressure Safety Injection (HPSI) and Containment Spray (CS) pumps take suction from the Containment Emergency Sump (CES) on a Recirculation Actuation Signal (RAS). The configuration of the HSPI suction piping from the CES is such that, after normal fill and vent evolutions, there is a short section of horizontal piping and elbow downstream of the inside CES isolation valves, (HV9304 & HV9305) that remains unfilled, (approximately 10 feet). This configuration does not conform to the system design description in the UFSAR.
UFSAR 6.3.2.2.4 Section E reads as follows:

"All ECCS lines will be maintained during normal operation in a filled condition. Suitable vents are provided, and administrative procedures require that all ECCS lines be returned to a filled condition following all events (maintenance, etc.) that require draining of any of the lines. Maintaining the lines in a filled condition will minimize possible water hammer conditions from developing during system startup."

**Evaluation Summary:**

Calculation M-0012-039 (Assignment 0607000747-3), Nuclear Fuel Management evaluation (FS-17) and water hammer evaluation (FS-15) were performed to evaluate the effects of the pocket of air on Engineered Safety Features (ESF) pump performance. The calculations and evaluations concluded that the performance of the HPSI and CS pumps with the entrained air does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety evaluated in the UFSAR, nor does it result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The proposed activity to accept the small void in the CES suction piping and to revise the UFSAR to reflect that condition can be implemented without prior NRC approval.

**AR 060800698-44, Change to Updated Final Safety Analysis Report (UFSAR) Section 8.1 for Performing Electrical Cable Splices in Cable Trays**

**Description:**

This change adds criteria to UFSAR Section 8.1 for performing electrical cable splices in cable trays. Electrical cable spliced in a cable tray results in a negligible effect on the likelihood of a cable malfunction and therefore only a minimal change in the likelihood of occurrence of a malfunction of an SSC important to safety.

The proposed activity revises UFSAR Section 8.1 to clarify how the recommendations of Regulatory Guide 1.75, "Physical Independence of Electric Systems" Revision 1, Section C-9 are followed. This change adds the criteria required for performing cable splices in cable trays.

**Evaluation Summary:**

As stated in UFSAR Appendix 3A, the design for physical independence of electrical systems at SONGS Units 2 and 3 follows recommendations of Reg. Guide 1.75, Rev. 1, except for certain differences indicated in UFSAR 7.1.2.29 and 8.1.4.3. Reg. Guide 1.75 addresses requirements for physical independence of electrical systems to satisfy General Design Criteria 17 (GDC-17). GDC-17 states, in part, that the onsite electric distribution systems have sufficient independence to perform their safety functions assuming a single failure.

This proposed activity changes UFSAR Section 8.1, paragraph 8.1.4.3.14.B to allow cable splicing inside cable trays if the following criteria are applied. These criteria are supported by and consistent with current industry standards and guidelines and are consistent with the intent of Reg. Guide 1.75 to satisfy the commitment to GDC-17

Consequently, it was concluded that the activity may proceed without the need for regulatory approval.

**800072713-0060, Increased Fuel Handling Accident (FHA) Inside the Fuel Handling Building (FHB)**

**Description:**

This evaluation addresses an increase in the analyzed weight of the fuel assembly and dropped components modeled in a postulated fuel handling accident inside the fuel handling building (an FHA-FHB event). As a consequence of the increased drop weight, the number of fuel rods that could fail during the FHA-FHB event increases, resulting in potential increases in the Exclusion Area Boundary (EAB) and Control Room dose consequences reported in UFSAR Table 15.10.7.3.4-3.

**Evaluation Summary:**

The evaluation determined:

(1) The increase in the analyzed weight of the fuel assembly and dropped components is not an initiator of any accident and no new failure modes are introduced. Hence, accident and malfunction frequencies previously evaluated remain unaffected; the activity does not introduce the possibility of a change in the consequences of any malfunction previously evaluated in the UFSAR, nor does the activity create the possibility of an accident of a different type or a malfunction with a different result than any previously evaluated in the UFSAR.

(2) There are no Design Basis Limits for Fission Product Barriers or methods of evaluation described in the UFSAR applicable to the analyzed weight of the fuel assembly and dropped components.

(3) Calculation N-6072-003 Revision 0, CCN# D0000472 determined the radiological consequences of increasing the analyzed weight of the fuel assembly and dropped components and the resultant increase in the number of failed fuel rods modeled in the Fuel Handling Accident Inside the Fuel Handling Building Dose Analysis. The revised doses increased by no more than 10 percent of the difference between the current UFSAR dose values and the 10 CFR 50 Section 50.67 regulatory guideline values, and are within the current NUREG-0800 Standard Review Plan, Section 15.0.1 guideline values for an FHA event. Hence, the increase in the analyzed weight of the fuel assembly and dropped components results in no more than a minimal increase in the radiological consequences of the FHA-FHB event previously evaluated in the UFSAR.

Subsequent to this evaluation a need to revise fuel move procedures and update the Technical Specifications was recognized and the NRC was informed. No fuel movements were allowed until Operations and Maintenance procedures were revised and Corrective Actions were implemented in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety," with appropriate procedure revisions being made to require mitigating ventilation systems to be operable before allowing any fuel movement. A Technical Specification change is being processed to add a condition of Applicability to Technical Specifications that apply during irradiated fuel movement. The new Applicability will include movement of un-irradiated fuel over irradiated fuel.

**800072713-0140, Increased Fuel Handling Accident (FHA) Inside Containment (IC)**

**Description:**

This evaluation addresses an increase in the analyzed weight of the fuel assembly and dropped components modeled in a postulated fuel handling accident inside the containment building (an FHA-IC event). As a consequence of the increased drop weight, an increased number of fuel rods could fail during the FHA-IC event, resulting in potential increases in the EAB and Control Room dose consequences reported in UFSAR Table 15.10.7.3.9-3.

**Evaluation Summary:**

The evaluation determined:

(1) The increase in the analyzed weight of the fuel assembly and dropped components is not an initiator of any accident and no new failure modes are introduced. Hence, accident and malfunction frequencies previously evaluated remain unaffected; the activity does not introduce the possibility of a change in the consequences of any malfunction previously evaluated in the UFSAR, nor does the activity create the possibility of an accident of a different type or a malfunction with a different result than any previously evaluated in the UFSAR.

(2) There are no Design Bases Limits for a Fission Product Barriers or methods of evaluation described in the UFSAR applicable to the analyzed weight of the fuel assembly and dropped components.

(3) Calculation N-6072-002 Revision 0, CCN# D0000934 examined the radiological consequences of increasing the analyzed weight of the fuel assembly and dropped components and the resultant increase in the number of failed fuel rods modeled in the Fuel Handling Accident Inside Containment Dose Analysis. The revised doses increased by no more than 10 percent of the difference between the current UFSAR dose values and the 10 CFR 50 Section 50.67 regulatory guideline values, and are within the current NUREG-0800 Standard Review Plan Section 15.0.1 guideline values for an FHA event. Hence, the increase in the analyzed weight of the fuel assembly and dropped components results in no more than a minimal increase in the radiological consequences of the FHA-IC event previously evaluated in the UFSAR.

Subsequent to this evaluation a need to revise fuel move procedures and update the Technical Specifications was recognized and the NRC was informed. No fuel movements were allowed until Operations and Maintenance procedures were revised and Corrective Actions were implemented in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety," with appropriate procedure revisions being made to require mitigating ventilation systems to be operable before allowing any fuel movement. A Technical Specification change is being processed to add a condition of Applicability to Technical Specifications that apply during irradiated fuel movement. The new Applicability will include movement of un-irradiated fuel over irradiated fuel.

**800187778-0070, Modification to Limit Pressure Increases in the Low Pressure Safety Injection (LPSI) Header**

**Description:**

Check valve S21204MU074 and isolation valve 2HV9328 leak by at a very small flow rate of approximately 0.01 gpm during normal operation, causing a slight inflow into the LPSI discharge header, which results in an increase in internal pressure.

This Engineering Change Package (ECP) adds a quarter inch tubing line between the LPSI discharge header (S21204ML038) and the common Shutdown Cooling (SDC) suction line (S21201ML050) to the two (2) LPSI pumps to allow the small leak rate of 0.01 gpm to be absorbed in the suction side of the system.

**Evaluation Summary:**

Adding the new quarter inch tubing will not increase the frequency of occurrence of an accident previously evaluated in the UFSAR. The new quarter inch line will meet the same tubing design, material, and construction standards that are applicable to the LPSI low pressure discharge header (which bounds the requirements for the shutdown cooling suction header). There is no impact on LPSI system performance because the flow rate through the new quarter inch line will be less than 0.5 gpm. For similar reasons, it will not increase the likelihood of occurrence of a malfunction of any SSC important to safety previously evaluated in the UFSAR.

Adding the quarter inch tubing has no impact on the radiological consequences of any accidents or SSC malfunctions because: (1) there is no reduction in the ability of the LPSI/Shutdown Cooling system to mitigate Chapter 6 and applicable Chapter 15 events because the very small flow rate through the new quarter inch tubing has a negligible impact on system performance, and (2) there is no increase in the bounding leakage rates from the LPSI/Shutdown Cooling system to the Refueling Water Storage tank (RWST) during accident conditions (i.e., no impact to RWST dose calculation N-6060-004).

Adding the quarter inch tubing does not introduce the possibility of an accident of a different type than previously evaluated in the UFSAR because postulated pipe breaks have already been evaluated in the vicinity of the new tubing, and that evaluation remains bounding. As a passive component, the only other possible failure mode is flow blockage, which would result in the current operating condition (prior to installation of the tubing).

There is no impact to any Design Bases Limits for a Fission Product Barriers as described in the UFSAR. It was concluded that this change may be made without prior NRC approval.

ENCLOSURE 2

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

REPORT ON THE COMMITMENT CHANGES MADE PER
NEI "GUIDELINES FOR MANAGING NRC COMMITMENT CHANGES"

FOR THE PERIOD

FROM MARCH 1, 2007 UNTIL DECEMBER 18, 2008

## REPORT ON COMMITMENT CHANGES MADE PER "NEI AGUIDELINES FOR MANAGING NRC COMMITMENTS"

NRC letter to Mr. Ralph Beedle, Senior Vice President and Chief Nuclear Officer, Nuclear Energy Institute (NEI), dated March 31, 2000 and SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," both state that the NEI 99-04 "A Guidelines for Managing NRC Commitments Changes" Revision 0, dated August 2, 1999, was an acceptable guide for licensees to follow for managing and changing their commitments to the NRC. Part of the commitment change process, given in the NEI guidelines, identifies that various commitments can be changed with the notification to the NRC made in a report submitted annually or along with the FSAR updates as required by 10CFR50.71(e). The intent of this report would be to provide a brief summary of the commitments changed since the last report in lieu of filing individual notifications as commitments are revised.

The following summarizes the commitment changes for San Onofre Units 2 and 3 that are to be reported and have occurred from the last facility change report in the period from March 1, 2007 until December 18, 2008.

### 1. REVISING THE METHOLODOGY OF COMPONENT COOLING WATER (CCW) HEAT EXCHANGER TESTING

In the March 29, 1991 "Program Response for Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment," Southern California Edison (SCE) informed the NRC that part of the program to address Service Water System problems affecting Safety-Related equipment would include performance testing of the (CCW) heat exchangers every refueling outage while on shutdown cooling at the start of refueling outages. After three tests, the licensees were allowed to determine the best test frequency to provide assurance that the equipment will perform the intended safety functions during the intervals between tests with the minimum extended frequency is at least once every 5 years. After several inspections, it was determined that the CCW heat exchanger performance tests be conducted under normal operating conditions. This commitment change was transmitted in SCE Facility Change Report dated January 21, 1998. As a result of evaluating the effects of back-flushing, it was determined that the CCW heat exchanger performance tests should be performed on only one CCW heat exchanger during each refueling outage after back-flushing with the heat load at least equal to 20% of the design heat load. The other CCW heat exchanger would be tested under normal operating conditions, on-line, for information only. This commitment change was transmitted in SCE Facility Change Report dated June 11, 2007.

The testing of the heat exchangers has not shown any degrading trends in any of the heat exchangers. Generic Letter 89-13 allows the licensee to determine the best frequency for testing, so long as the minimum testing frequency is at least once every 5 years. Because testing since 1990 has shown that the CCW heat exchangers are not degrading, future testing will consist of one heat exchanger tested at high heat load each refueling outage.

Revising the test interval to test one heat exchanger at high heat load meets the GL 89-13 requirement to "provide assurance that the equipment will perform the intended safety functions during the intervals between tests and meet the requirements of GDC 44, 45, and 46." 10 CFR 50 Appendix A General Design Criteria (GDC) 44, 45, and 45 discuss design requirements of the cooling systems. The design of the CCW and saltwater cooling (SWC) systems is not being altered, so the requirements of GDC 44, 45, and 46 are still met.

This commitment change is being reported to the NRC in the Refueling Interval Summary Report per the NEI Guidelines. This is a change to a commitment made in response to a Generic Letter which has been implemented.

## 2. ADMINISTRATIVE CONTROLS FOR THE DISTRIBUTION ROOM DURING STATION BLACKOUT

In a response dated February 12, 1993 to the NRC Safety Evaluation Report (SER) on Station Blackout which requested additional information on several specific issues, SCE committed to take administrative controls to ensure that the Distribution Rooms temperature remains below 120° F during a Station Blackout event. As a result of the addition of new swing battery chargers, a reanalysis of the temperature response in the Distribution Rooms show none of the room temperature responses are expected to exceed 120° F during a Station Blackout event. The administrative controls committed to in the February 12, 1993 response are no longer needed to ensure that the Distribution Rooms remain below 120° F during a Station Blackout event.

This commitment was explicitly credited as a basis for a safety decision in an NRC SER. Following the NEI Guidelines for Managing NRC Commitment Changes, this commitment can be revised and the NRC notified in the next Refueling interval summary report (Facility Change Report).