UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

February 13, 2008

EA-08-051

Richard M. Rosenblum
Senior Vice President and
Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000361/2007005; 05000362/2007005 AND NOTICE OF VIOLATION

Dear Mr. Rosenblum:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your San Onofre Nuclear Generating Station, Units 2 and 3 facility. The enclosed integrated report documents the inspection findings, which were discussed on December 21, 2007, and February 13, 2008, with Mr. R. Ridenoure and other members of your staff.

The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

One violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding this violation are described in detail in the enclosed report. The violation involved your failure to implement effective corrective actions to ensure thermal overloads associated with safety-related equipment would not fail prematurely (EA-08-051). Although determined to be of very low safety significance (Green), this violation is being cited because not all the criteria specified in Section VI.A.1 of the NRC Enforcement Policy for a noncited violation (NCV) were satisfied. Specifically, Southern California Edison failed to restore compliance within a reasonable time after the violation was first identified in Inspection Report 05000361;05000362/2006005. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

This report also documents three NRC identified and self-revealing findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as NCVs consistent with Section VI.A of the NRC Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at San Onofre Nuclear Generating Station, Units 2 and 3, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jeffrey A. Clark, Chief
Project Branch E
Division of Reactor Projects

Dockets: 50-361
50-362

Licenses: NPF-10
NPF-15

Enclosures:
Notice of Violation
NRC Inspection Report 05000361/2007005; 05000362/2007005
w/Attachment: Supplemental Information

cc w/enclosure:
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SUNSI Review Completed: _GBM__ ADAMS: ✖Yes ☐ No Initials: __GBM_
✖ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ✖ Non-Sensitive
R:_REACTORS_SO23\2007\SO2007-05RP-CCO.wpd ADAMS ML080440436

RIV:RI:DRP/E	SRI:DRP/E	SPE:DRP/E	C:DRS/PSB	C:DRS/OB
GMiller	CCOsterholtz	GReplogle	MPShannon	RELantz
/RA/	***/RA teleph./***	***/RA electronic/***	***/RA/***	***/RA/***
02/13/08	02/13/08	02/13/08	02/12/08	02/12/08

C:DRS/EB	C:DRS/PEB	SES/ACES	C:DRP/E
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02/13/08	02/11/08	2/12/08	02/13/08

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NOTICE OF VIOLATION

Southern California Edison Co.
San Onofre Nuclear Generating Station

Docket No. 50-361;362
License No. NPF-10;15
EA 08-051

During an NRC inspection conducted on September 27 through December 31, 2007, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that "measures shall be established to ensure that for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition."
>
> Contrary to this, from February 6 through August 8, 2007, the licensee failed to take corrective actions to preclude repetition of the premature tripping of thermal overloads for safety-related equipment, a significant condition adverse to quality.

This violation is associated with a Green SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Southern California Edison Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-08-051" and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you <u>must</u>

specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 13th day of February, 2008

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-361, 50-362
Licenses:	NPF-10, NPF-15
Report No.:	05000361/2007005 and 5000362/2007005
Licensee:	Southern California Edison Co. (SCE)
Facility:	San Onofre Nuclear Generating Station, Units 2 and 3
Location:	5000 S. Pacific Coast Hwy. San Clemente, California
Dates:	September 27, 2007 through December 31, 2007
Inspectors:	C. C. Osterholtz, Senior Resident Inspector, Project Branch E, DRP M. O. Miller, Senior Resident Inspector, Project Branch E, DRP M. R. Young, Resident Inspector, Project Branch E, DRP G. Warnick, Senior Resident Inspector, Project Branch D, DRP R. A. Kopriva, Senior Reactor Inspector, Engineering Branch 1, DRS J. H. Nadel, Reactor Inspector, Engineering Branch 1, DRS G. A. George, Reactor Inspector, Engineering Branch 1, DRS B. D. Baca, Health Physics Inspector, Plant Support Branch, DRS L. T. Ricketson, Senior Health Physics Inspector, Plant Support Branch, DRS S. T. Makor, Reactor Inspector, Engineering Branch 1, DRS J. P. Adams, Reactor Inspector, Engineering Branch 1, DRS L. E. Ellershaw, Senior Reactor Inspector, Engineering Branch 1, DRS M. T. Baquera, Reactor Inspector, Engineering Branch 1, DRS K. Clayton, Senior Operations Engineer, Operations Branch, DRS
Approved By:	Jeffrey A. Clark, Chief Project Branch E Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR05000361/2007005, 05000362/2007005; 09/27/07 - 12/31/07; San Onofre Nuclear Generating Station, Units 2 & 3; Integrated Resident and Regional Report; Emergent Work, Operability Evaluations, Occupational Radiation Safety, Problem Identification and Resolution.

This report covered a 3-month period of inspection by resident inspectors and Regional office inspectors. The inspection identified four Green findings consisting of one cited violation and three noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management's review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green noncited violation of 10 CFR 50.65(a)(2) associated with the failure to include Units 2 and 3 emergency diesel generator (EDG) automatic voltage regulator (AVR) deficiencies as functional failures in the maintenance rule program. The inspectors noted that the voltage regulator deficiencies should have placed the emergency diesel generators into Maintenance Rule 10 CFR 50.65(a)(1) status approximately 6 months after the failures occurred. This caused a lapse in the determination of appropriate system monitoring and goal setting to maintain system reliability. This issue was entered into the licensee's corrective action program as Action Request 070300161.

  This finding was associated with the mitigating systems cornerstone. This issue was similar to non-minor Example 7.b of Manual Chapter 0612, Appendix E, in that the finding was more than minor since violations of 10 CFR 50.65(a)(2) necessarily involve degraded system performance. This finding is not suitable for evaluation using the Significance Determination Process because the performance deficiency did not cause the degraded equipment performance. This is a Category II finding per Inspection Procedure 71111.12, so it was determined to have very low safety significance (Green) by management judgement per Manual Chapter 0609, Appendix M. The cause of the finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program (P.1©) because the licensee failed to thoroughly evaluate the cause and extent of condition of the failed emergency diesel generator automatic voltage regulator (Section 1R12).

- Green. The inspectors identified a Green noncited violation of Technical Specification 5.5.1.1 associated with the failure to implement procedural guidance to ensure the proper application of a submersible pump to prevent wetting of the steam supply to the Unit 2 turbine-driven auxiliary feedwater pump.

If the water level were to wet the steam line insulation, it would cause condensation in the steam line and render the auxiliary feedwater pump inoperable due to possible water hammer or turbine overspeed on a pump start. This issue was entered into the licensee's corrective action program as Action Request 071000309.

The finding was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance (Green) because it did not result in a loss of safety function and did not affect the risk of external initiators. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program (P.1©) in that the licensee did not thoroughly evaluate the problem such that the resolutions address causes and extent of conditions (Section 1R15).

- <u>Green</u>. A self-revealing Green violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to prevent recurrence of premature tripping of Square D thermal overloads used for equipment protection on safety-related equipment. The licensee failed to scope the thermal overloads associated with the Unit 3 saltwater cooling pump room because they had previously determined that it had sufficient margin such that it would not be susceptible to failure. This resulted in the premature tripping of thermal overloads for the Unit 3 saltwater cooling pump room intake structure fan on August 8, 2007. This issue was entered into the licensee's corrective action program as Action Request 070800454.

  The finding was determined to be more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and it affected the cornerstone objective by challenging the availability and capability of safety-related components. The inspectors also noted that this a repetitive problem in implementing corrective actions. Based on the results of the Significance Determination Process Phase 1 evaluation, the finding was determined to have very low safety significance because it did not result in an actual loss of a system safety function, a loss of a single train of safety equipment for greater than its Technical Specification allowed outage time, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding also had crosscutting aspects in the area of problem identification and resolution associated with the corrective action program (P.1©) because the licensee failed to thoroughly evaluate the extent of condition of insufficient solder material on safety-related thermal overloads (Section 4OA2).

Cornerstone: Occupational Radiation Safety

- Green. The inspector reviewed a self-revealing noncited violation of Technical Specification 5.5.1.1 when a worker failed to follow radiation work permit instructions. On July 14, 2007, after completing a pre-job site review, a worker proceeded to verify work authorization boundaries in Unit 3, Room 209, without contacting radiation protection for current radiological conditions and discussing the work scope and locations as required by the radiation work permit. The worker approached Valve S31902MU012 and received a dose rate alarm. The maximum dose rate levels in the area were 30 millirem per hour on contact with the piping system and 12 millirem per hour at 30 centimeters. The licensee's corrective actions were to coach the worker and to develop and implement a mechanism to communicate associated boundary walk downs in maintenance orders.

  The failure to follow a radiation work permit instruction is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the Occupational Radiation Safety cornerstone objective, in that workers not following their radiation work permit does not ensure adequate protection of the worker health and safety from additional personnel exposure. The finding was determined to be of very low safety significance because it did not involve: (1) as low as is reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Further, this finding had a human performance crosscutting aspect in the work practices component because the workers did not use human error prevention techniques, such as self checking, to ensure the full work scope, locations, and radiological conditions were discussed with radiation protection personnel as required by the radiation work permit [H4a] (Section 2OS1).

B. Licensee-Identified Violations

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and their corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at 99 percent power. On October 20, 2007, Unit 2 was shutdown to Mode 3 to perform an extent of condition review as a result of Unit 3 main steam isolation valve, main feedwater isolation valve, and main feedwater block valve solenoid failures. The surveillance tests for Unit 2 valves that contained the specific solenoids in question were performed when Unit 2 was in Mode 3. All surveillance tests were completed satisfactory. Unit 2 was to restart on October 21, 2007, but did not begin restart until October 25, 2007, due to complications with the Southern California brush fires. Unit 2 returned to power operation on October 26, 2007.

On November 26, 2007, Unit 2 was shutdown and cooled down for a planned refueling outage. Unit 2 entered Mode 6 and began core alterations on December 7, 2007. Unit 2 was still in the refueling outage at the end of the inspection period.

Unit 3 began the inspection period at 99.9 percent. On October 9, 2007, the licensee performed a shutdown of Unit 3 for a planned mid-cycle outage. Unit 3 was returned to power operation on November 9, 2007, and ended the inspection period at approximately 99.9 percent reactor power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of changes to the facility structures, systems, and components (SSC); risk-significant normal and emergency operating procedures; test programs; and the Updated Final Safety Analysis Report (UFSA) in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The inspectors utilized Inspection Procedure 71111.02, "Evaluation of Changes, Tests, or Experiments," for this inspection.

The inspectors reviewed eight safety evaluations performed by the licensee since the last NRC inspection of this area at San Onofre Nuclear Generating Station. The evaluations were reviewed to verify that licensee personnel had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. The inspectors reviewed 33 screenings, in which licensee personnel determined that evaluations were not required, to ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59. Evaluations and screenings reviewed are listed in the attachment to this report.

The inspectors reviewed and evaluated a sample of recent licensee action requests to determine whether the licensee had identified problems related to 10 CFR Part 50.59

evaluations, entered them into the corrective action program (CAP), and resolved technical concerns and regulatory requirements. The reviewed action requests are identified in the Attachment.

The inspection procedure specifies that the inspectors review a minimum sample of six licensee safety evaluations and 12 applicability determinations and screenings (combined). The inspectors completed a review of eight licensee safety evaluations and 33 screenings.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the three listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walk down to the licensee's UFSAR and CAP to ensure problems were being identified and corrected.

- October 18, 2007, Unit 3, Shutdown Cooling Train B prior to mid-loop operations
- October 29, 2007, Unit 3, Train B containment spray pump (P013) used as backup to shutdown cooling
- December 18, 2007, Unit 2, electrical alignment to safety Bus 2A06 while 2A04 is out of service

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the UFSAR, Technical Specifications (TS), and vendor manuals to determine the correct alignment of the Unit 2 auxiliary feedwater system; (2) reviewed outstanding design issues, operator workarounds, and UFSAR documents to determine if open issues affected the functionality of the Unit 2 auxiliary feedwater system; and (3) verified that the licensee was identifying and resolving equipment alignment problems. Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Quarterly Inspection

The inspectors walked down the six listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the UFSAR to determine if the licensee identified and corrected fire protection problems.

- October 2, 2007, Unit 2, emergency diesel Generator (EDG) 2G002 room
- October 2, 2007, Unit 2, EDG 2G003 room
- October 2, 2007, Unit 3, EDG 3G002 room
- October 2, 2007, Unit 3, EDG 3G003 room
- November 14, 2007, Unit 2, emergency core cooling system pump Room 002
- December 5, 2007, Unit 2, containment

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards and reviewed critical operating parameters and maintenance records for the Unit 3 Train B component cooling water heat Exchanger S31203ME002. The inspectors verified that: (1) performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; (2) the licensee utilized the periodic maintenance method outlined in Electric Power Research Institute (EPRI) NP- 7552, "Heat Exchanger Performance Monitoring Guidelines;" (3) the licensee properly utilized biofouling controls; (4) the licensee’s heat exchanger inspections adequately assessed the state of cleanliness of their tubes, and (5) the heat exchanger was correctly categorized under the Maintenance Rule. Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination (NDE) activities and, if performed, one to three welds on the reactor coolant system (RCS) pressure boundary.

The inspectors directly observed the following nondestructive examinations:

System	Component/Weld ID	Exam Type
RCS	Surge Nozzle to Safe End Weld, 02-005-031	PT/UT
RCS	Shutdown Cooling Piping 10" SCH 140 Pipe-Valve, 02-059-008	PT/UT
RCS	Shutdown Cooling Piping 16" SCH 160 Pipe-Elbow, 02-059-002	PT/UT

RCS	Shutdown Cooling piping 16" SCH 160 Pipe-Valve, 02-059-001	PT/UT
RCS	Snubber, 02-052-110	VT3

The inspectors reviewed the following NDEs through record review:

System	Component/Weld ID	Exam Type
RCS	Y-Stop Valve, 02-021-068	VT3
RCS	Y-Stop Valve, 02-021-081	VT3
RCS	Guide & Y-Stop Valve, 02-039-058	VT3
Feedwater	Guide & Y-Stop Valve, 02-045-037	VT3
RCS	10" SCH 140 Reducer Tee-Pipe, 02-021-038	UT

The inspectors observed the initial Ultrasonic Examination System calibration for the Panametrics Epoch 4 instrument, S/N 040229207, which was recorded on Ultrasonic Instrument Calibration Data Record and Certification. The inspectors reviewed Table 1 in Electric Power Research Institute's PDI Protocol PDI-UT-2, Revision 20, dated 25 APR 07, to verify that the transducers to be used for ultrasonic examinations on stainless steel piping were appropriately qualified.

The inspectors reviewed the NDE personnel qualification records for those contractor personnel (Lambert MacGill Thomas, Inc. or LMT) performing ASME Code Section XI inservice inspections. The LMT personnel had been appropriately certified using LMT's procedure QA-46, "Qualification and Certification of NDE and Visual Examination Personnel per ASME Section XI," Revision 0. The inspectors verified that the requirements in QA-46 were consistent with ASNT CP-189-1995, “ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel,” 1995 Edition.

The inspection procedure further required verification of one to three welds on Class 1 or 2 pressure boundary piping to ensure that the welding process and welding examinations were performed in accordance with the ASME code. The inspectors observed portions of the preemptive structural weld overlay on the ASME code Class 1 pressurizer surge line nozzle-to-safe end dissimilar weld and pipe-to-safe end stainless steel weld identified as follows:

System	Component/Weld Identification
Pressurizer Surge Line Nozzle-to-Safe End-to-Pipe	Weld DMW 02-0005-031and Weld 02-016-001 Gas Tungsten Arc Welding (machine)

Welding procedures and NDE of the welding repair conformed to ASME code requirements and licensee commitments.

Welder qualification documentation packages and welder maintenance logs were reviewed for all contract welders (Welding Services, Inc.) performing welding activities on the pressurizer surge nozzle. The documentation packages and logs were in accordance with Article III, QW-300 "Welding Performance Qualification" in Section IX of the ASME code.

Welding Procedure Specifications WPS 08-08-T-001-Butter SS, Revision 0, and WPS 03-08-T-804-Bottom, Revision 0, were the welding procedures observed being used during the weld overlay process on the pressurizer surge nozzle. The inspectors reviewed the welding procedure specifications and their corresponding procedure qualification records (identified in the Attachment) to verify that ASME Code required essential variables for the gas tungsten arc welding process had been identified, recorded in the procedure qualification record, and formed the basis for qualification of the welding procedure specifications.

Additionally, the inspectors reviewed manual gas tungsten arc welding and shielded metal arc welding performed on an ASME Code Class 3 component cooling water by-pass line around the letdown heat exchanger. This welding consisted of carbon steel pipe-to-pipe and pipe-to-fitting (4" and 8") welding using ER70S-6 and E7018 welding filler material. The reviewed welds are identified as Weld Records WR2-07-212, WR2-07-213, and WR2-07-210.

The inspectors verified, by review, that the Welding Procedure Specification (1-GT-SM) had been properly qualified in accordance with the requirements of Section IX of the ASME code. The inspectors verified that the essential variables for both the shielded metal arc welding and the gas tungsten arc welding processes had been identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specification.

The inspectors also observed the liquid penetrant examinations performed on the buffer (stainless steel) layer and the transition bead (between the buffer layer and the dilution layer). The buffer layer represents the initial stainless steel layer of the weld overlay that started at a point on the stainless steel pipe and covered the pipe, pipe-to-safe end weld, safe end, and ending as close as practical to the dissimilar metal weld fusion line, without contacting the dissimilar metal weld. These examinations were recorded on Liquid Penetrant Nondestructive Examination Report 104532-PT-001. The examination personnel qualification records for the examiner performing the examination were reviewed to verify that the individual was properly certified. Further, the inspectors reviewed the liquid penetrant procedure (WSI QAP 9.21, Revision 1) to verify that it was properly qualified in accordance with ASME code Section V requirements. Additionally, the inspectors reviewed the Ultrasonic Examination Report of the ultrasonic examination performed on December 10, 2007, of the weld overlay which was at a nominal thickness of 0.30 inches at the examination time.

The inspectors also verified by observation that welding filler materials were properly stored and controlled in accordance with Procedure SO 123-I-11.1. Welding Filler Material Control Records, used to document issuance and return of welding filler materials, were reviewed for those materials issued on December 13, 2007, to verify that specified administrative controls regarding welders, materials (quantity and time limits), and use of portable ovens or caddys were being implemented.

The inspection procedure required inspection of any augmented or industry initiation examinations. The inspectors determined that the licensee had not performed such examinations. Consequently, the inspectors did not perform any activities in this area.

b. Findings

No findings of significance were identified.

.2 Vessel Upper Head Penetration (VUHP) Inspection Activities

a. Inspection Scope

The licensee performed NDEs of 100 percent of reactor VUHP. The inspector directly observed a sample of the examinations performed on the control element drive mechanism element (CEDM) and incore instrumentation (ICI) as listed below:

System	Component/Weld Identification	Examination Method
RCS	CEDM 87	UT/ET
RCS	CEDM 88	UT/ET
RCS	CEDM 79	UT/ET
RCS	CEDM 68	UT/ET
RCS	CEDM 60	UT/ET
RCS	CEDM 28	UT/ET
RCS	CEDM 78	UT/ET
RCS	CEDM 86	UT/ET
RCS	ICI 96	UT/ET
RCS	ICI 95	UT/ET
RCS	ICI 94	UT/ET
RCS	ICI 93	UT/ET
RCS	RVUH vent line	UT/ET

The NDEs were performed in accordance with the requirements of NRC Order EA-03-009.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection (BACC) Activities

a. Inspection Scope

Resident inspectors observed a sample of BACC activities and verified that visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components.

The inspector reviewed five instances where boric acid deposits were found on reactor coolant system piping components during the walkdown. The inspectors reviewed licensee procedures governing the boric acid corrosion control program and inspector qualifications, reviewed the extent of boric acid residue on the various components, verified that the licensee inspectors who performed the walkdown were qualified, and determined whether components that exhibited leakage during the current outage had experienced leakage in the past. The following table lists the specific components reviewed by the inspector, including the component numbers, brief component descriptions, and the resulting Action Requests.

Component Number	Description	Action Request
2HV0512	Pressurizer surge line sample isolation valve	070500261
2HV9203	Charging line insolation valve	071101172
2HV9201	Charging auxiliary spray isolation valve	071101173
2HV9339	Shutdown cooling isolation valve	070500262
2HV9326	Shutdown injection tank drain valve	070500265

No boric acid leakage evaluations were performed for any of the instances where leaks were identified during walkdowns.

The condition of the components was appropriately entered into the licensee's CAP and corrective actions taken were consistent with ASME code requirements. No engineering evaluations were required for any of the instances where leaks were identified during walkdowns.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in-situ screening criteria to assure consistency between assumed NDE flaw sizing accuracy and data from the EPRI examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results.

At the time of this inspection, no conditions had been identified that warranted in situ pressure testing. The inspectors did, however, review the licensee's report for Units 2 and 3, “Steam Generator Degradation Assessment for the Cycle 15 Refueling Outages in 2007 and 2008,” dated November 29, 2007, and compared the in situ test screening parameters to the guidelines contained in the EPRI document “In Situ Pressure Test Guidelines”, Revision 2, and the Combustion Engineering Owners Group screening criteria. This review determined that the remaining screening parameters were consistent with the EPRI and Combustion Engineering Owners Group guidelines.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition and analysis technique sheets used during this refueling outage and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspector reviewed acquisition technique and analysis technique sheets are identified in the attachment.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. The inspectors compared the previous outage operational assessment predictions contained in Report R-3671-00-1, “Tube Degradation Predictions for the San Onofre Nuclear Generating Station Unit 2 Steam Generators - 2006 Update,” with the flaws identified thus far during the current steam generator tube inspection effort. Compared to the projected damage mechanisms identified by the licensee, the number of identified indications fell within the range of prediction and were quite consistent with predictions. No new damage mechanisms had been identified during this inspection.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet TS requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy current test scope established by TS requirements and the licensee’s degradation assessment report. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had accounted for all known flaws and had, as a minimum, established a test scope that met TS requirements, EPRI guidelines, and commitments made to the NRC. The scope of the licensee's eddy current examinations of tubes in both steam generators included:

- Bobbin examination full length of tubing (tube end hot-tube end cold) from both hot and cold legs, in non-sleeved tubes, rows 4-147

- Bobbin examination of the unsleeved portion of tubing (sleeve top hot-tube end cold) from the cold leg, in sleeved tubes, rows 4-147

- Bobbin examination of the straight length section of tubing from both hot and cold legs, rows 1-3

- Rotating plug point coil examination of hot leg Tubsheet TSH +4", -13", 100 percent of all tubes

- Rotating plug point coil examination of cold leg tubesheet, TSC +2", -13", 100 percent of all tubes. Exception: Steam Generator 89 tubes R141-C63, R140-C64, R139-C63, and surrounding tubes in 2-tube bounding pattern, examination extent is TSC +4", -13".

- Rotating plug point coil examination of the sleeves (sleeve bottom hot-sleeve top hot), 100 percent of sleeved tubes

- Rotating plug point coil examination of SBF 0.00", -1.25" in Steam Generator 88, Tube R28-C60 only

- Rotating plug point coil examination of U-bend section of tubing (07H-07C) with mid/high frequency coil probe, 100 percent of tubes in rows 1-3

- Rotating plug point coil examination of U-bend section of tubing (07H-07C) with mid-frequency coil probe, 20 percent sample of tubes in rows 4-10 (rows 5-10 sample drawn from tubes not examined with MRPC probe in the 2006 inspection)

- Rotating plug point coil examination of the following bobbin indications: ADR, DNI, DEI,DSI, DTI, LPI, PLP, NQI, TWD (0-100 percent), DNT >= 2.0 volts, DNG >= 4.0 volts, TSD, TSM, PDP, and CUD

- Rotating plug point coil examination of PLP indications (with LAR confirmation) in a 2-tube bounding pattern, location +/- 1-inch of PLP edges

- Rotating plug point coil examination of all sections of tubing which cannot be examined with the 600UL bobbin probe due to restriction

The inspection procedure specified, if new degradation mechanisms were identified, verify that the licensee fully enveloped the problem in its analysis of extended conditions including operating concerns and had taken appropriate corrective actions before plant startup. To date, the eddy current test results had not identified any new degradation mechanisms.

The inspection procedure requires confirmation that the licensee inspected all areas of potential degradation, especially areas that were known to represent potential eddy current test challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation were included in the scope of inspection and were being inspected.

The inspection procedure further requires verification that repair processes being used were approved in the TSs. The total number of tubes plugged was 133 tubes in Steam Generator 88 and 125 tubes in Steam Generator 89. The inspectors verified that the mechanical expansion plugging process to be used was an NRC-approved repair process.

The inspection procedure also requires confirmation of adherence to the TS plugging limit, unless alternate repair criteria have been approved. The inspection procedure further requires determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors determined that the TS plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than three gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure requires verification that the licensee had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure requires confirmation that the eddy current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of eddy current tests performed on the tubes in Steam Generators 88 and 89. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered, and (4) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of site-specific qualifications of the techniques being used. These are identified in the attachment.

If loose parts or foreign material on the secondary side were identified, the inspection procedure specified confirmation that the licensee had taken or planned appropriate repairs of affected steam generator tubes and that they inspected the secondary side to either remove the accessible foreign objects or perform an evaluation of the potential effects of inaccessible object migration and tube fretting damage. At this time of the inspection, no foreign material had been identified.

Finally, the inspection procedure specified review of one to five samples of eddy current test data if questions arose regarding the adequacy of eddy current test data analyses. The inspectors did not identify any results where eddy current test data analyses adequacy was questionable.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspection procedure requires review of a sample of problems associated with inservice inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions.

The inspector reviewed corrective action reports which dealt with inservice inspection activities and found the corrective actions were appropriate. Action requests reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee has an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified. The inspectors completed one sample by completing all required inspection activities.

1R11 Licensed Operator Requalification (71111.11)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario on October 22, 2007, involved just-in-time training for Unit 2 startup. Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Annual Inspection

a. Inspection Scope

The inspectors reviewed the annual operating examination test results for 2007. Since this was the first half of the biennial requalification cycle, the licensee was not required to administer a written examination. These results were assessed to determine if they were consistent with NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," guidance and Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process," requirements. This review included the test results for a total of 15 crews composed of 87 licensed operators, which included: shift-standing senior operators, staff senior operators, shift-standing reactor operators, and staff reactor operators. There were no crew failures and no individual failures on the simulator scenario portion of the test. There was one individual failure on the job performance measure portion of the test. This individual was successfully remediated prior to returning to shift.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the listed maintenance activity to: (1) verify the appropriate handling of SSC performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50 Appendix B, and the TSs.

- October 1, 2007, Units 2 and 3, upgraded EDG automatic voltage regulators

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50.65(a)(2) for the failure to include Units 2 and 3 EDG automatic voltage regulator (AVR) deficiencies as functional failures in the maintenance rule program. The inspectors noted that the voltage regulator deficiencies should have placed the EDGs into maintenance rule 10 CFR 50.65(a)(1) status approximately six months after the failures occurred. This caused a lapse in the determination of appropriate system monitoring and goal setting to maintain system reliability.

Description. On March 3, 2007, the licensee identified that an AVR for the Unit 3 EDG was oscillating excessively during a load test. The cause of the oscillation was poor contact of the R3 potentiometer because of the open type housing of the potentiometers which made them susceptible to dirt intrusion.

The licensee's analysis of the failed AVR concluded that the R3 potentiometer poor contact caused the AVR to oscillate the EDG output voltage setting between zero and 3.8 megavolt ampere reactive (MVAR). Operations personnel subsequently declared the EDG inoperable. All of the susceptible potentiometers on all eight EDGs were subsequently upgraded to sealed multiturn gold plated potentiometers. The upgraded installations were completed on August 26, 2007.

The inspectors discovered that the licensee had not evaluated the AVR deficiency in their maintenance rule program for monitoring or goal setting. The inspectors determined that the AVR failure impacted the reliability of the EDGs in accordance with NUMARC 93-01, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants," Revision 2. The inspectors concluded that the AVR failure if correctly counted as a MPFF, would have caused the EDG to exceed the performance criteria and should have been tracked for monitoring and goal setting in the licensee's maintenance rule program. In response to this finding, the licensee subsequently placed the EDGs in 10 CFR 50.65(a)(1), and established an EDG performance goal such that both Unit 2 and 3 EDG AVRs be successfully surveillance tested four times each, with normal voltage and MVAR control, by the end of the fourth quarter of 2007. Each EDG contains an AVRs A and B, therefore four diesels each containing two AVRs would need to be surveillance tested four times to successfully complete the goal.

Analysis. The failure to recognize the applicability of the maintenance rule for a failure of the EDG AVR was a performance deficiency. This finding was associated with the mitigating systems cornerstone. This issue was similar to non-minor Example 7.b of Manual Chapter 0612, Appendix E, in that the finding was more than minor since violations of 10 CFR 50.65(a)(2) necessarily involve degraded system performance. This finding is not suitable for evaluation using the Significance Determination Process because the performance deficiency did not cause the degraded equipment performance. This is a Category II finding per Inspection Procedure 71111.12, so it was determined to have very low safety significance (Green) by management judgement per Manual Chapter 0609, Appendix M. The cause of the finding has a crosscutting aspect in the area of problem identification and resolution associated with the CAP (P.1(c)) because the licensee failed to thoroughly evaluate the cause and extent of condition of the failed EDG AVR.

Enforcement. 10 CFR Part 50.65(a)(1) requires, in part, that holders of an operating license shall monitor the performance or condition of SSCs within the scope of the rule against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended safety functions. 10 CFR 50.65(a)(2) requires, in part, that monitoring specified in paragraph (a)(1) is not required where it has been demonstrated the performance or condition of an SSC is being effectively controlled through appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, from March through September, 2007, the licensee failed to demonstrate the performance of the EDGs was being effectively controlled through appropriate preventive maintenance and did not establish goals to provide a reasonable assurance that the Units 2 and 3 EDGs were capable of fulfilling their intended function. Because the finding is of very low safety significance and has been entered into the licensee's CAP as AR 070300161, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000361; 05000362/2007005-01, "Failure to Properly Implement Maintenance Rule Requirements for Emergency Diesel Generators."

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Risk Assessment and Management of Risk

a. Inspection Scope

The inspectors reviewed the four below listed assessment activities to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- October 4, 2007, Unit 3, risk assessment and management during an unplanned emergency core cooling system TS 3.0.3 entry

- October 25, 2007, Unit 2, risk assessment and management during a startup after unplanned shutdown and southern California fires

- October 12, 2007, Unit 3, risk assessment and management during a main steam isolation valve dual indication

- November 30, 2007, Unit 2, risk assessment and management during the Devers offsite power out of service - delayed midloop operations

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the UFSAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- October 3, 2007, Units 2 and 3, incorrect calibration probe used for saltwater cooling flow indicators
- October 4, 2007, Unit 2 turbine-driven auxiliary feedwater pump failed trench eductor
- October 9, 2007, Unit 3, grounded pressurizer heater
- October 25, 2007, Unit 2 and 3, main feedwater isolation Valve 2HV4048 and main steam isolation Valve 2HV8204 solenoid failed in-service testing

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

Introduction. The inspectors identified a Green NCV of TS 5.5.1.1 associated with the failure to implement procedural guidance to ensure the proper application of a submersible pump to prevent wetting of the steam supply to the Unit 2 turbine-driven auxiliary feedwater pump. If the water level were to wet the steam line insulation, it would cause condensation in the steam line and render the auxiliary feedwater pump inoperable due to possible water hammer or turbine overspeed on a pump start.

Description. On October 4, 2007, during a plant walk-down, the inspectors noted that a submersible pump was in use in a pipe trench in the Unit 2 auxiliary feedwater (AFW) pump building while steam was discharging into the bottom of the pipe trench. The pump was a temporary modification installed due to a failure of a permanently installed eductor. The purpose of the eductor was to ensure water did not accumulate in the trench such that it could contact the steam piping. If the water level were to wet the steam line insulation, it would cause condensation in the steam line and render the turbine-driven AFW pump inoperable due to the possibility of water hammer or overspeed on turbine start.

The inspectors noted that the atmosphere in the top of the pipe trench felt very hot to the touch. The inspectors then reviewed the vendor manual for the submersible pump and hose and found that both had a maximum temperature rating of 140°F. The inspectors concluded that water in the pipe trench could easily exceed the maximum temperature rating for the submersible pump and hose rated of 140°F. Since this temperature would exceed the rating of the pump and hose, the submersible pump modification could not be relied upon to drain the trench. This could potentially render the turbine driven AFW pump inoperable.

The inspectors interviewed the licensee's staff and found that the submersible pump and discharge hose had been installed per Procedure S023-2-16, "Use of Temporary Sump Pumps," Revision 20. The inspectors noted this procedure did not direct consideration of the environment in which the pump would be used or the potential consequences of failure of the pump, as would have been required by Procedure S0123-XV-5.1, "Temporary Modifications Control," Revision 8. Since the failure of the submersible pump had the potential consequence of rendering safety-related equipment inoperable, the inspectors concluded the procedure used to install the modification was inadequate.

Corrective actions taken by the licensee included revising the "Use of Temporary Sump" procedure to reflect the guidance found in the "Temporary Modifications Control" procedure for consideration of the environmental effects on the submersible pump. Additionally, the licensee revised Procedure OSM-5, "Operator Rounds," Revision 7, and replaced the submersible pump with one that was adequately temperature rated for the environment in the AFW trench.

Analysis. The failure to have an adequate procedure resulting in an inadequate modification with the potential to affect safety-related equipment was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance (Green) because it did not result in a loss of safety function and did not affect the risk of external initiators. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the CAP (P.1(c)) in that the licensee did not thoroughly evaluate the problem such that such that the resolutions address causes and extent of conditions.

Enforcement. TS 5.5.1.1 requires that written procedures be established, implemented, and maintained for activities specified in Appendix A, "Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors," of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations), dated February 1978. Regulatory Guide 1.33, Appendix A, Section 9.e recommends general procedures for the control of maintenance and modification work. Contrary to this requirement, on May 11, 2007, the licensee failed to implement appropriate procedures to control modification work in the Unit 2 auxiliary feedwater steam supply trench to ensure the trench would not fill up with water and render the Unit 2 turbine driven auxiliary feedwater pump inoperable. Because this violation is of very low safety significance and has been entered into the licensee's CAP as AR 071000309, it is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000362/2007005-02, "Failure to Implement Procedural Requirements for Modifications in the Auxiliary Feedwater Steam Supply Trench."

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope

The inspectors reviewed seven permanent plant modification packages and associated documentation, such as implementation reviews, safety evaluation applicability determinations, and screenings, to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed the procedures governing plant modifications to evaluate the effectiveness of the program for implementing modifications to risk-significant SSCs, such that these changes did not adversely affect the design and licensing basis of the facility.

Procedures and permanent plant modifications reviewed are listed in the attachment to this report. Further, the inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages and process.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications by reviewing a sample of related condition reports. The reviewed condition reports are identified in the attachment.

The inspection procedure specifies inspectors review a required minimum sample of six permanent plant modifications. The inspectors completed review of seven permanent plant modifications.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the six listed postmaintenance test activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the UFSAR to determine if the licensee identified and corrected problems related to post maintenance testing.

- October 25, 2007, Unit 2, main steam isolation Valve 2HV8204, Train A & B, fail safe closure postmaintenance test

- October 25, 2007, Unit 2, Main Feedwater Isolation Valve, 2HV-4048, stroke and fail safe closure postmaintenance test

- October 29, 2007, Unit 3, Pressurizer Surge Line Nozzle Field Weld OVL-031, post weld overlay liquid penetrant postmaintenance test

- October 31, 2007, Unit 3, reactor coolant gas vent system postmaintenance test

- November 3, 2007, Unit 3 reactor coolant gas vent system postmaintenance test following corrective maintenance

- November 8, 2007, Unit 3, saltwater cooling Pump 3P112 postmaintenance test

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, "Loss of Decay Heat Removal:" (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory or midloop conditions; (11) refueling activities; (12) heatup and coldown activities; (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris; and observation of supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging. Documents reviewed by the inspectors are listed in the attachment. The inspectors reviewed outage activities for Unit 3 from October 9, 2007 to November 9, 2007. The inspectors also reviewed outage activities for Unit 2 from November 26, 2007, until the end of the inspection period.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and TSs to ensure that the four listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- August 1, 2007, Unit 2, 2HV-9900 normal chilled water to containment isolation Valve 2HV-9900 stroke test

- October 4, 2007, Unit 3, Train A saltwater cooling outlet Valve 3HV6497 partial manual stroke test

- October 18, 2007, Unit 2, high pressure safety injection Pump 2MP018 response time testing

- October 18, 2007, Unit 2, component cooling water Pump 2MP024 inservice test

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the UFSAR, plant drawings, procedure requirements, and TSs to ensure that the below listed temporary modification was properly implemented. The inspectors: (1) verified that the modifications did not have an affect on system operability/availability; (2) verified that the installation was consistent with modification documents; (3) ensured that the post-installation test results were satisfactory and that the impact of the temporary modifications on permanently installed SSCs were supported by the test; and (4) verified that appropriate safety evaluations were completed. The inspectors verified that licensee identified and implemented any needed corrective actions associated with temporary modifications.

- October 9, 2007, Unit 3, swap grounded pressurizer Heater ME616 with Heater E614

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance was identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the listed drill and simulator-based training evolutions contributing to Drill/Exercise Performance and Emergency Response Organization Performance Indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and Protective Action Recommendation development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the NEI 99-02, "Voluntary Submission of Performance Indicator Data," acceptance criteria.

- October 3, 2007, Units 2 and 3 simulator, control room, technical support center, operations support center, and emergency operations facility, Unit 3 diesel Generator 3G003 fuel oil day tank fire, Unit 2 steam generator tube leak and subsequent tube rupture with potential unfiltered radioactive release pathway through the steam driven auxiliary feed Pump P-140 turbine exhaust

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas in the Auxiliary, Radwaste, Reactor, and Containment Buildings
- Radiation exposure permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in two potential airborne radioactivity areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools.
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Radiation exposure permit briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspector completed 21 of the required 21 samples.

b. Findings

Introduction. The inspector reviewed a self-revealing NCV of TS 5.5.1.1 when a worker failed to follow radiation work permit instructions.

Description. On July 14, 2007, a worker notified health physics of a pre-job site review prior to starting work on Valve 3HV7261 in the Post Accident Sampling System Lab. The worker was informed of the radiological conditions for the work area. However, after completing the pre-job site review, the worker proceeded to verify the work authorization boundaries in Unit 3, Room 209. The worker approached Valve S31902MU012 and received a dose rate alarm. The worker exited the radiologically controlled area and informed health physics of the alarm. The peak dose rate received by the worker was 11.1 millirem per hour and area around valve S31902MU012 had a maximum dose rate level of 30 millirem per hour on contact with the piping system and 12 millirem per hour at 30 centimeters. During the licensee's investigation of the dose rate alarm, the licensee determined that the worker did not inform health physics of all areas needing access to complete the work scope and did not receive a radiological briefing for Unit 3, Room 209. The licensee's corrective actions were to coach the worker and to develop and implement a mechanism for communicating associated boundary walk downs in maintenance orders.

Analysis. The failure to follow a radiation work permit instruction is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the Occupational Radiation Safety cornerstone objective, in that workers not following their radiation work permit does not ensure adequate protection of the worker health and safety from additional personnel exposure. This occurrence involved a worker's unplanned, unintended dose, or potential for such a dose that could have been significantly greater as a result of a single minor, reasonable alteration of the circumstances, higher dose rate levels. This finding was determined to be of very low safety significance because it did not involve: (1) as low as is reasonably achievable (ALARA) planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Further, this finding has a work practices human performance cross cutting aspect in human error prevention techniques because the worker failed to self check the work scope and work locations when briefing with health physics prior to entering the radiological controlled area [H4a].

Enforcement. Technical Specification 5.5.1.1.a requires applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 7(e), of the Appendix, requires procedures for access control and a radiation work permit system. Procedure SO 123-VII-20, "Health Physics Program," Revision 12, Section 6.10.6.5 states, in part, that individuals entering a radiological controlled area sign on an appropriate radiation exposure permit acknowledging that they agree to comply with the radiological controls specified on the radiation exposure permit. Radiation Exposure Permit 07070562000/200159, states, in part, that workers, prior to entering the radiologically controlled area, are to inform the Health Physics Control Point of the job scope and work locations. Contrary to the Radiation Exposure Permit requirement, on July 14, 2007, the worker did not inform the health physicist at the control point of the full work scope and work locations prior to entering the radiological controlled area which resulted in the worker knowing the current radiological conditions of Room 209. Because this finding is of very low safety significance and was entered into the licensee's corrective action program (Action Request 070700545), this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000362/2007005-03, Failure to follow a radiation exposure permit requirement.

2OS2 Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Site-specific ALARA procedures
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Dose rate reduction activities in work planning
- Exposure tracking system

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers' use of the low dose waiting areas

- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques

- Corrective action documents related to the ALARA program and follow-up activities, such as initial problem identification, characterization, and tracking

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspector completed 5 of the required 15 samples and 8 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors sampled licensee data for the Mitigating System Performance Index (MSPI) performance indicators (PI) listed below for Units 2 and 3 for the period from September 26, 2007 through December 31, 2007. The definitions and guidance of Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, were used to verify the licensee's basis for reporting unavailability and unreliability in order to verify the accuracy of PI data. The inspectors reviewed operating logs, Limiting Conditions for Operation logs, ARs, and the maintenance rule database to verify that the licensee properly accounted for planned and unplanned unavailability as part of the assessment. The inspectors sampled data to verify that the licensee: (1) accurately documented the actual unavailability hours for the MSPI systems; and (2) accurately documented the actual unreliability information for each MSPI monitored component. In addition, the inspectors interviewed licensee personnel associated with PI data collection and evaluation.

- Units 2 and 3, safety system functional failures

The inspectors completed two samples.

<u>Cornerstone: Barrier Integrity</u>

The inspectors sampled licensee submittals for the four performance indicators listed below for the period September 26, 2007 through December 31, 2007, for Units 2 and 3. The definitions and guidance of Nuclear Energy Institute 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 4, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period. The inspectors: (1) reviewed RCS chemistry sample analyses for dose equivalent Iodine-131 and compared the results to the TS limit; (2) observed a chemistry technician obtain and analyze a RCS sample; (3) reviewed operating logs and surveillance results for measurements of RCS identified leakage; and (4) observed a surveillance test that determined RCS identified leakage. Licensee performance indicator data were also reviewed for the following:

- Units 2 and 3, reactor coolant system specific activity
- Units 2 and 3, reactor coolant system leakage

The inspectors completed four samples.

<u>Cornerstone : Occupational Radiation Safety</u>

Occupational Exposure Control Effectiveness

The inspector reviewed licensee documents from January 1 through September 30, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee’s technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Indicator Guideline,” Revision 5). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

<u>Cornerstone: Public Radiation Safety</u>

Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspector reviewed licensee documents from January 1 through September 30, 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Radiological Controls Review

a. Inspection Scope

The inspector evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

.2 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's corrective action program. This assessment was accomplished by reviewing maintenance orders, action requests, the management focus list, and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the corrective action program; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the two below listed issues for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- August 7, 2007, Unit 3, saltwater cooling pump room thermal overload trip
- December 18, 2007, Units 2 and 3, comprehensive review of operator workarounds

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

Introduction. A self revealing Green violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to prevent recurrence of premature tripping of Square D thermal overloads used for equipment protection on safety-related equipment. The licensee failed to scope the thermal overloads associated with the Unit 3 saltwater cooling pump room because it had erroneously determined that it had sufficient margin such that it would not be susceptible to failure. This resulted in the premature tripping of thermal overloads for the Unit 3 saltwater cooling pump room intake structure fan on August 8, 2007.

Description. The licensee previously had problems with spurious thermal overload trips and received a noncited violation for untimely corrective actions to resolve the problem (see NRC Inspection Report 05000361;362/2006-005). On October 17, 2006, the Unit 2 fuel handling building pump room emergency air conditioning Unit 2E441 Phase B thermal overload tripped for no apparent reason with the fan turned off. The inspectors noted that six spurious trips of other thermal overloads had occurred since December 2005. These overloads were associated with the Unit 3 fuel handling building post accident cleanup room emergency air conditioning Unit 3E371, the Unit 2 fuel handling building pump room emergency air conditioning Units 2E441 and 2E442, and the Unit 2 component cooling water Pump 2P024 room emergency air conditioning Unit 2E453. All of these thermal overloads were subsequently changed out for larger devices in 2005 because of chronic problems with spurious trips.

The inspectors reviewed the history of spurious thermal overload trips and discovered that five previous apparent cause assessments (ACEs) had been performed since January 2001 to identify and correct spurious trips associated with thermal overloads. A 2001 ACE identified equipment aging as the cause, and directed that replacement thermal overloads be installed. A 2002 ACE identified degraded cabling lugs as the cause, and the lugs were replaced. A 2003 ACE identified the cause as insufficient margin in the trip settings, which were adjusted. A 2004 ACE attributed a series of spurious trips to warm weather. Finally, a 2005 ACE identified that the thermal overloads were undersized, and that new, larger thermal overloads should be installed. The licensee upgraded 64 thermal overloads to a larger capacity model in December 2005. However, the inspectors concluded that the ACEs and the associated corrective actions generated by the licensee had been ineffective in resolving the problem.

The licensee performed a root cause evaluation as part of RCE070901311 initiated in response to the thermal overload failures. Procedure SO123-XV-50, “Corrective Action Process,” Revision 7, directs a root cause evaluation for significant problems and to prevent recurrence of the consequences of these problems. The inspectors concluded a root cause evaluation was appropriate since Procedure SO123-XV-50 specifies criteria for a root cause that include safety equipment failures with generic operability issues and long-standing problems requiring escalation for resolution. The inspectors determined these criteria were met based on the generic implications involving failures of safety related equipment and the numerous apparent causes that had been performed since January 2001 that had failed to correct the issue. The inspectors therefore concluded the failure of the thermal overloads represented a significant condition adverse to quality.

The licensee implemented a detailed plan for testing the thermal overloads and X-rayed the internals to determine if a design defect had previously gone undetected. The licensee discovered that two mechanisms in concert with each other were causing the spurious trips. Thermal overloads associated with small motors had a tendency to trip early due to higher than expected current levels going through the overloads while the associated line voltage was high in the normal band. Also, the X-ray analysis revealed that approximately 20 percent of the sample had insufficient melting alloy, contributing to a thermal overload tripping on lower current.

The licensee established a plan to replace the affected thermal overloads with properly sized components that would be X-rayed for sufficient melting alloy verification prior to installation. However, the licensee concluded sufficient margin existed in a group of 75 thermal overloads, including those associated with the Unit 3 saltwater cooling pump room intake structure fans.

On August 8, 2007, the intake structure fan for the Unit 3 saltwater cooling pump room tripped. The cause was subsequently determined to be a defective thermal overload on the Phase C portion due to insufficient solder material in the thermal overload. The thermal overload was replaced, and temperature in the Unit 3 saltwater cooling pump never approached its design value of 98°F. The licensee has since replaced all 75 susceptible thermal overloads that were previously scoped out of the corrective action process.

<u>Analysis</u>. The failure of the licensee to properly scope corrective actions to prevent the premature tripping of thermal overloads for safety-related equipment was considered a performance deficiency. The finding was determined to be more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and it affected the cornerstone objective by challenging the availability and capability of safety-related components. Using the Manual Chapter 0609, “Significance

Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance (Green) because it did not result in an actual loss of a system safety function, a loss of a single train of safety equipment for greater than its technical specification allowed outage time, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. The cause of the finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program (P.1(c)) because the licensee failed to thoroughly evaluate the extent of condition of insufficient solder material on safety-related thermal overloads.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to ensure that for significant conditions adverse to quality, corrective actions are taken to preclude repetition. Contrary to this, from February 6 through August 8, 2007, the licensee failed to take corrective actions to preclude repetition of the premature tripping of thermal overloads for safety-related equipment, a significant condition adverse to quality. This finding has been entered into the licensee's corrective action program as AR 070800454. Due to the licensee's failure to restore compliance from previous NCV 05000361;05000362/2006005-04, within a reasonable time after the violation was identified, this violation is being cited as a Notice of Violation consistent with Section VI.A of the Enforcement Policy: VIO 05000361; 05000362/2007005-04, "Failure to Prevent Recurrence of Premature Tripping of Square D Thermal Overloads."

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semi-annual trend review of repetitive or closely related issues that were documented to identify trends that might indicate the existence of more safety significant issues, specifically in the areas of procedural compliance and human performance. The inspectors review consisted of the six month period from June 25, 2007, through December 31, 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed corrective action program items associated with human performance improvement, and met with representatives from the San Onofre human performance improvement team at regular intervals. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. Documents reviewed by the inspectors are listed in the attachment.

b. Findings

No findings of significance were identified. However, the inspectors noted that the licensee continued to attempt to implement human performance initiatives to prevent personnel errors. The licensee indicated that a stand alone performance improvement plan would be implemented by January 31, 2008.

4OA5 Other

.1 Temporary Instruction 2515/166, "Pressurized Water Reactor Containment Sump Blockage," San Onofre Nuclear Generating Station, Unit 2

Temporary Instruction 2515/166 was performed at San Onofre Nuclear Generating Station, Unit 2. The results of inspection phase of Temporary Instruction 2515/166 for Unit 2 are subsequently documented in this report. Temporary Instruction 2515/166 for both Unit 2 and Unit 3 will be closed out after the completion and verification of modification commitments for Unit 2 containment sumps at the end of Refueling Outage 15.

Listed below are the commitments and actions taken by the licensee:

1. Design and procurement of replacement sump screens

   Actions Taken

   Engineering Change Packet ECP#040301974-11 dated Jul 17, 2006, provides for the design changes of containment sump to address sump blockage concerns. This engineering change packet has undergone NRC review and supplemental responses to the NRC are to be received no later than February 29, 2008, per letter to Nuclear Energy Institute (NEI) from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors," dated November 30, 2007. Materials for the sump screens have been procured and are currently being installed during Refueling Outage RF15, with modifications expected to complete at the end of the outage.

2. Resolution of potential susceptibility of emergency core cooling system and containment spray system pump mechanical seal to increased leakage due to debris mix passing through the seals

   Actions Taken

   The licensee has completed calculations to evaluate seal leakage due to debris ingestion. This action has undergone NRC review and supplemental responses to the NRC are to be received no later than February 29, 2008, per letter to NEI from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors," dated November 30, 2007.

3. Resolution of potential susceptibility of ECCS and CSS pump mechanical seal cyclone separators to debris blockage

Actions Taken

The licensee has completed calculations to evaluate seal leakage due to debris ingestion. This action has undergone NRC review and supplemental responses to the NRC are to be received no later than February 29, 2008, per letter to NEI from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors," dated November 30, 2007.

4. Development of a reduced qualified protective coatings zone of influence (ZOI)

Actions Taken

ALION-CAL-SONGS2933-02, Revision 1 "San Onofre Units 2 and 3 GSI-191 Containment Recirculation Sump Evaluation: Debris Generation Calculation," documents the assumptions and methodology that the licensee applied to determine the ZOI and debris generated for each postulated break. This evaluation has undergone NRC review and supplemental responses to the NRC are to be received no later than February 29, 2008, per letter to NEI from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated November 30, 2007.

5. Validation of the 8 percent head loss margin adjustment factor for chemical effects (SONGS uses Trisodium Phosphate (TSP) as a post-LOCA pH buffering agent, and pertinent debris loads are primarily mineral wool fibrous insulation, making NRC's Integrated Chemical Effects Test (ICET) 2 generally applicable, but the licensee stated that chemical effects values were subject to follow-on sump screen vendor testing, and SCE evaluations and walkdowns).

Actions Taken

Chemical effect tests were completed by Alion Science and Technology, and directly observed by the NRC, in Warrenville, Illinois on August 17 - 18, 2006. Open items from the NRC review are to be addressed and supplemental responses to the NRC are to be received no later than February 29, 2008, per letter to NEI from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors," dated November 30, 2007.

6. Containment insulation configuration control to ensure the amounts and types of insulation remain within acceptable debris loading design margins

Actions Taken

The licensee has removed microtherm insulation on four different piping segments in containment. This insulation is to be replaced by reflective metal insulation where appropriate. Mineral wool insulation on the steam generators is to be replaced with RMI during the steam generator replacement activities in 2009. These actions have undergone NRC review and supplemental responses to the NRC are to be received no later than February 29, 2008, per letter to NEI from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors" dated November 30, 2007.

7. Replace sump screens at SONGS Unit 2 during Refueling Outage Cycle 15

<u>Actions Taken</u>

Work currently ongoing and expected to be completed by the end of the refueling outage.

8. Removal of microporous insulation on piping to be completed coincident with sump screen replacement.

<u>Actions Taken</u>

Work currently ongoing and expected to be completed by the end of the refueling outage.

9. Modification fo steel grates at the entry to the bioshield to reduce the potential for debris blockage and resultant hold-up of recirculating water to be completed coincident with sump screen replacement.

<u>Actions Taken</u>

Work currently ongoing and expected to be completed by the end of the refueling outage.

4OA6 <u>Meetings, Including Exit</u>

On November 9, 2007, the engineering inspectors presented the results of the permanent plant modifications inspection and the evaluation of changes, tests, or experiments inspection to Dr. R. Waldo and others who acknowledged the findings.

On November 30, 2007, the health physics inspectors presented inspection results to Mr. J. Reilly and others who acknowledged the findings.

On December 3, 2007, the inspector discussed the inspection results of the licensed operator annual requalification examination with Mr. B. Arbour, Training Supervisor. A telephone exit was held with Mr. Arbour, on December 3, 2007. The licensee acknowledged the findings presented in both the briefing and the final exit meeting.

On December 13, 2007, the inspectors presented the results of this inservice inspection to J.T. Reilly, Vice-President Engineering and Technical Services, and other members of licensee management. Licensee management acknowledged the inspection findings.

On December 21, 2007, and on February 13, 2008, the inspectors presented the quarterly inspection results to Mr. R. Ridenoure and others who acknowledged the findings.

The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Licensee Technical Specification Section 5.5.1.1.a requires applicable procedures recommended in Regulatory Guide 1.33. Revision 2, Appendix A, February 1978. Section 7e of the Appendix requires procedures for access control and a radiation work permit system. Radiation Exposure Permit A081997001/200117-8 requires workers to wear radiological protective clothing for entry into contaminated areas, such as shoe covers and gloves. Contrary to this requirement, there were three examples of security officers entering contaminated areas without the required protective clothing. The first example occurred on October 9, 2007, when two security guards entered a posted contaminated area in Unit 3, Room 411 of the penetrations building, without the required radiological protective clothing. The second example occurred on November 12, 2007, when a security guard entered a posted contaminated area in Unit 2, Room 209 without the required radiological protective clothing. The third example occurred November 13, 2007, when a security guard entered a posted contaminated area in Unit 2, Room 209 without the required radiological protective clothing. In all three examples, the area postings had changed and with inattention to detail, the officers entered the areas without the required radiological protective clothing. This issue was entered into the licensee's corrective action program (Action Requests 071000551, 071100759, and 071100760). This finding is of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Axline, Technical Specialist, Nuclear Regulatory Affairs
D. Breig, Manager, Engineering Standards and Excellence
B. Corbett, Manager, Health Physics
J. Hirsch, Manager, Maintenance
K. Johnson, Manager, Design Engineering
R. Ridenoure, Vice President, Nuclear Generation
L. Kelly, Engineer, Nuclear Regulatory Affairs
C. McAndrews, Manager, Nuclear Oversight and Assessment
N. Quigley, Manager, Mechanical/Nuclear Maintenance Engineering
J. Reilly, Vice President, Engineering and Technical Services
A. Scherer, Manager, Nuclear Regulatory Affairs
R. St. Onge, Manager, Maintenance and Systems Engineering
T. Vogt, Manager, Special Projects
D. Wilcockson, Manager, Plant Operations
C. Williams, Manager, Compliance
T. Yackle, Manager, Operations
O. Flores, Manager, Chemistry
J. Morales, Manager, Projects
M. Cooper, Manager, Maintenance and Systems Engineering
S. Gardner, Nuclear Engineer, Nuclear Regulatory Affairs
A. Mahindrakar, Technical Specialist/Scientist, Maintenance and Systems Engineering
J. Valsvig, Technical Specialist/Scientist, Maintenance and Systems Engineering
M. McDevitt, Senior Nuclear Engineer, Engineering and Technical Services
P. Chang, Nuclear Engineer, Maintenance Engineering
A. Matheney, Senior Nuclear Engineer, Engineering and Technical Services
M. Wade, Westinghouse Representative
M. Short, Director Nuclear Oversight and Assessment
J. Todd, Manager, Nuclear Oversight and Regulatory Affairs

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000361; 05000362/2007005-04	NOV	Failure to Prevent Recurrence of Premature Tripping of Square D Thermal Overloads (Section 4OA2.2)

Opened and Closed

05000361; 05000362/2007005-01	NCV	Failure to Properly Implement Maintenance Rule Requirements for Emergency Diesel Generators (Section 1R12)
05000362/2007005-02	NCV	Failure to Implement Procedural Requirements for Modificaitons in the Auxiliary Feedwater Steam Supply Trench (Section 1R15)
05000362/2007005-03	NCV	Failure to Follow a Radiation Exposure Permit Requirement (Section 2OS1)

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

In addition to the documents called out in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Section 1R02: Evaluations of Changes, Tests, or Experiments

10 CFR 50.59 Evaluations

020701289-37	Auxiliary steam system radwaste condensate return line rad monitor flow valve change - Fix position of Condensate Return Valve 2/3FV-7546 and remove 2/3FIC-7546	Revision 0
050801215-08	Change to the U3C14 Core Fuel Loading Pattern	Revision 0
060101335-13	Reduction in the number of Dome Air Circulator Fans Credited for Containment Sprayed and Unsprayed Region Mixing.	Revision 0
060401009-06	One-time change to the testing frequency for the High Pressure Turbine Stop and Control Valves	Revision 0

060700747-13	Perform Calculation to evaluate the effects of air pocket on Engineered Safety Feature pump performance.	Revision 0
060700747-18	Perform Calculation to evaluate the effects of air pocket on Engineered Safety Feature pump performance.	Revision 1
060800698-13	Engineering design work by Bechtel to support steam generator replacement - Remove one Containment Hydrogen Recombiner E146 for one cycle of operation to facilitate Steam Generator Replacement	Revision 0
060800698-44	Change to UFSAR Section 8.1, paragraph 8.1.4.3.14.B	Revision 0

<u>10 CFR 50.59 Screenings</u>

040400696-17	Add ECP vent line at AFW pump motor outboard bearing housing to eliminate oil leak	09/25/2007
041100092-79	Need to Evaluate U-2 CCW Fisher Butterfly valve concerning valve taper pin issue	
050300070-05	Install Steam Trap in Auxiliary Steam Cross-tie header	
050901044-40	Technical specification bases change to allow substituting B00X for battery B007 and B008 for temporary battery outage	11/01/2005
050901044-43	Technical specification bases change to allow substituting B00X for battery B007 and B008 for temporary battery outage	11/03/2005
050901044-61	Phase I of the Class 1E DC system upgrade	10/27/2005
050901044-61	Technical specification bases change to allow substituting B00X for battery B007 and B008 for temporary battery outage (update)	12/16/2005
050901044-82	Technical specification bases change to allow substituting B00X for battery B007 and B008 for temporary battery outage	03/20/2006
051000132-06	Update AOV Program Procedure to update valve IST Procedure.	
051200901-07	Installation of a flow orifice downstream of 2PCV4716	07/25/2006
060200607-18	Add DC shunts to batteries 2B007 and 2B009 for monitoring current	06/08/2006

060200607-51	Add DC shunts to batteries 2B007 and 2B009 for monitoring current - Addition of an 800 Amp, 100 mV DC shunt at the positive polarity of battery B00X	08/02/2006
060400474-04	Modify required actions in procedure SO23-5-1.7 to require MODE 3 entry for 1-3 inoperable MSSVs per steam generator	04/10/2006
060400474-12	Modify required actions in procedure SO23-5-1.7 to require MODE 3 entry for 1-3 inoperable MSSVs per steam generator	04/14/2006
060400474-32	Modify required actions in procedure SO23-5-1.7 to require MODE 3 entry for 1-3 inoperable MSSVs per steam generator	07/27/2006
060400474-41	Modify required actions in procedure SO23-5-1.7 to require MODE 3 entry for 1-3 inoperable MSSVs per steam generator	10/04/2006
060500070-14	ECP# 060500070-10: Replace 3P123 Feeder Breaker	05/052006
060500211-21	Replace vertical air tank S31319MV048	05/18/2006
060500211-38	Replace vertical air tank S31319MV048	06/16/2006
060500211-43	Replace vertical air tank S31319MV048	08/10/2006
060600089-84	Increase Thermal Overload size in breakers 2BY37, 3BY37, 3BZ33	09/18/2006
060800603-02	Replace existing R3, R4 potentiometers with a new model in AVR for EDG.	01/24/2007
060800603-16	Replace existing R3, R4 potentiometers with a new model in AVR for EDG.	01/24/2007
060800603-29	Replace existing R3, R4 potentiometers with a new model in AVR for EDG.	03/07/2007
061001071-19	Use of new E4C-109 battery short circuit methodology	03/28/2007
061001842-82	Upsize Thermal Overloads to avoid Spurious Trips	11/15/2006
061100895-11	Material condition of Generator Neutral Grounding Resistor is poor.	
061101272-04	Install Lifting Eye Pad on beam to allow in-line lift capability when changing out safety valve.	

070200876-05	Code upgrade installation for CENTS computer code version 06100	02/26/2007
070200876-06	Code upgrade installation for TORCGEOM computer code version 1.0.5	03/26/2007
070200876-07	Code upgrade installation for REX computer code version 2.1.6	09/20/2007
070200876-08	Code upgrade installation for CORD computer code version 1.3.7	09/20/2007
070700512-06	Lower the Set Point of the concerned instruments and provide Control Room indication of actual pressure.	

Calculations

E4C-112, CCN 72	Class 1E 480V MCC Protection Calculation	Revision 1
E4C-112, ECN A46476	Class 1E 480V MCC Protection Calculation	Revision 1
E4C-112,CCN 55	Class 1E 480V MCC Protection Calculation	Revision 1
M-0012-039	ESF Pump Suction with Entrained Air after RAS (Recirculation Actuation Signal)	Revision 0
N-4061-001	Post-Loss Of Coolant Accident Summary of Low Populated Zones and Offsite Doses	Revision 2
N-4061-002	Post-Loss Of Coolant Accident Containment Leakage - Control Room and Offsite Doses	Revision 1

Action Requests

050901044	060200607	060400474	060800603	061001071

Section 1R04: Equipment Alignment

Procedures

SO23-3-2.6	“Shutdown Cooling System Operation”	Revision 24
SD-SO23-780	“Auxiliary Feedwater System”	Revision 10
SD-SO23-120	“6.9 kV, 4.16 kV and 480 V Electrical Distribution Systems”	Revision 16
SO23-5-1.8.1	“Shutdown Nuclear Safety”	Revision 17

Drawings and Calculations

SD-SO23-740	“Shutdown Cooling System”	Revision 17
40160A	“Auxiliary Feedwater System - No. 1305"	Revision 43
40160B	“Auxiliary Feedwater Steam Supply System - No. 1301"	Revision 21
40160C	“Auxiliary Feedwater System Hydraulic Valves 2HV-4714 & 4731 Control Fluid System No. 1305"	Revision 7
40160X	“Auxiliary Feedwater System No. 1305 and Auxiliary Feedwater Steam Supply System No. 1301"	Revision 4

Section 1R05: Fire Protection

Procedures

2-013	“Unit 2, diesel generator pre-fire plans”	Revision 4
3-0345	“Unit 3, diesel generator pre-fire plans”	Revision 4
2-007	“Unit 2, Safety Equipment Building (-)15'6" elevation”	Revision 3
UFHA 2/3-7.0-2SE	“Updated Fire Hazard Analysis”	May 2007

Action Requests

070901019 070901022

Section 1R08: Inservice Inspections

Procedures

Number	Title	Revision
SO23-XXVII-20.51	Visual Examination Procedure for Operability of Nuclear Components and Supports and Conditions Relating to Their Functional Adequacy	2
SO23-XXVII-20.48	Liquid Penetrant Examination	1
SO23-XXVII-30.13	Risk-Informed Ultrasonic Examination of Class 1 Austenitic Piping Welds	0
SO23-XXVII-30.6	Ultrasonic Examination of Austenitic Piping Welds	2
SO23-XXVII-30.9	Ultrasonic Examination of Dissimilar Metal Piping Welds	2

PDI-UT-10	PDI Generic Procedure for the Ultrasonic Examination of Dissimilar Metal Welds	C
9022	Reactor Coolant System Alloy 600 Material Management Program	5
SO23-XXXIII-8.16	Reactor Coolant System Alloy 600 Inspection	5
SO23-3-2.34	Containment Access Control, Inspections and Airlocks Operation	20
SO123-XXIV-10.1	Engineering Change Package	15
SO123-0-A4	Configuration Control	9
SO23-1-1.11.1	Plant Maintenance Procedure for Coating Service Level 1 Application	6
SO23-XV-23.1.1	Containment Cleanliness/Loose Debris Inspection	1
SO23-V-8.17	Containment Coatings Assessment	1
QA-46	Qualification and Certification of NDE and Visual Examination Personnel per ASME Section XI	0
WSI QAP 9.21	Liquid Penetrant Examination	1
SI-UT-126	Phased Array Ultrasonic Examination	3
T4EN51	Non-RCS Alloy 600 Boric Acid Leakage, Inspection and Evaluation	1
T4EN52	RCS Alloy 600 Boric Acid Leakage, Inspection and Evaluation	0
SO23-V-8.15 ISS2	Containment Boric Acid Leak Inspection	2
SO23-V-8.18	Reactor Coolant System (RCS) Leak Monitoring and Investigation Guide	0
SO23-XV-85	Boric Acid Corrosion Control Program	1
SO23-XXXIII-8.16	Reactor Coolant System Alloy 600 Inspection	5
SO23-XXVII-3.51.9	IntraSpec UT Analysis Guidelines	5
SO23-XXVII-3.51.2	IntraSpec Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations	5
SO23-XXVII-3.51.4	IntraSpec Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time-of-Flight Ultrasonic, Longitudinal Wave & Shear Wave	5
SO23-XXVII-3.51.3	IntraSpec Eddy Current Analysis Guidelines	6

SO23-I-2.53	Containment Emergency Sump Inspection Surveillance	7
SO 123-I-11.1	Welding Filler material control	9

Corrective Action Documents

AR 070500261	AR 071101172	AR 071101173	AR 070500262
AR 070500263	AR 070500265	AR 071200384	AR 071200384
AR 060100998	AR 060101057	AR 060100961	AR 071200751
AR 071200830	AR 060901108-89		

Calculations

Number	Title	Revision
SONG-10Q-301	Weld Overlay Sizing for Pressurizer Surge Nozzle	2

Drawings

Number	Title	Revision
SONG-10Q-02	Pressurizer Surge Nozzle Weld Overlay Design and Buffer Layer, Shts 1 and 2	1
403974	Construction Drawing Surge, SONGS, Unit 2, Shts 1 and 2	0
S2-1203-ML-229	Letdown Heat Exchanger E-602 to Line 100: UA 2TV-0223, Sht 1	12
S2-1203-ML-498	Component Cooling Water, Sht 1	0

Examination Technique Specification Sheets (ETSS)

San Onofre Nuclear Generating Station ETSS	Qualifying EPRI ETSSs
ETSS #1	96004.1, 96005.2, 96008.1, 96012.1, 24013.1, 20511.1
ETSS #9	23514.1, .2, .3
ETSS #3	20510.1, 20511.1, 21409.1, 21410.1, 21998.1, 22401.1, 96703.1
ETSS #4	20510.1, 20511.1, 21409.1, 21410.1, 21998.1, 22401.1, 96703.1

ETSS #5	96008.1, 96511.2
ETSS #6	96511.2, 99997.1

Welding Procedure Specifications and Corresponding Procedure Qualification Reports

WPS 08-08-T-001-Butter SS, Revision 0: PQRs 08-08-T-009, 08-08-TS-001, 8.8.6-OKG, and 08-08-TS-002

WPS 03-08-T-804-Bottom, Revision 0: PQRs A08202.3-3, 43-43-T-001, 03-03-T-803, and A843256-52

WPS 1-GT-SM, Manual GTAW and/or SMAW of P-Number 1 CS, Revision 1: PQRs 51, 112, and 153

Miscellaneous

Number	Title	Revision
RPA 02-0080	Quantification of Containment Latent Debris	1
ECP#04031974-74	Microtherm Insulation to RMI Change-out ECP; Unit 2	
ECP# 04031974-58	Microtherm Insulation to RMI Change-out ECP; Unit 3	
ECP# 04031974-12	Sump Screen Installation and Bioshield Gate Modification ECP; Unit 2	
ECP#04031974-11	Sump Screen Installation and Bioshield Gate Modification ECP; Unit 3	
	Letter to NRC from SCE: NRC Generic Letter 2004-02 Response To NRC Request For Information San Onofre Nuclear Generating Station Units 2 and 3	March 7, 2005
	Letter to SCE from NRC: San Onofre Nuclear Generating Station Units 2 and 3-Request For Additional Information (RAI) Related to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Sump Recirculation At Pressurized-Water Reactors" (TAC NOS. MC4714 and MC4715)	June 2, 2005
	Letter to NRC from SCE: NRC Generic Letter 2004-02 Response To NRC Request For Additional Information	July 5, 2005
	Letter to NRC from SCE: NRC Generic Letter 2004-02 San Onofre Nuclear Generating Station Units 2 and 3	September 1, 2005

Letter to SCE from NRC: San Onofre Nuclear Generating Station, Units 2 and 3, Request For Additional Information RE: Response to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Sump Recirculation At Pressurized-Water Reactors" (TAC NOS. MC4714 and MC4715)	February 9, 2006
Letter to PWR Owners Group from NRC: Alternative Approach for Responding to the Nuclear Regulatory Commission Request for Additional Information Letter RE: Generic Letter 2004-02 (TAC NOS. See Enclosure)	March 26, 2006
Letter to PWR Owners Group from NRC: Alternative Approach for Responding to the Nuclear Regulatory Commission Request for Additional Information Letter RE: Generic Letter 2004-02 (TAC NOS. See Enclosure)	January 4, 2007
San Onofre Nuclear Generating Station Units 2 and 3-Report on Results of Staff Audit of Corrective Actions to Address Generic Letter 2004-02 (TAC NOS. MC4714 and MC4715)	May 16, 2007
Letter to NEI from NRC: Plant-Specific Requests for Extension of Time to Complete One or More Corrective Actions for Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors"	November 8, 2007
Letter to NEI from NRC: Supplemental Licensee Responses to Generic Letter 2004-02, "Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors"	November 30, 2007
ASNTCP-189-1995, ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel, 1995 Edition	
Request For Relief ISI-3-25, Use of Structural Weld Overlay and Associated Alternative Repair Techniques	
NRC Safety Evaluation for Request For Relief ISI-3-25	June 12, 2007
Weld Data Sheet, Pressurizer Surge Line Nozzle – Weld ID DMW 02-005-031	

	Welder Bead Logs for ER308L and Alloy 52M deposition on Unit 2 Pressurizer Surge Nozzle	
	Steam Generator Degradation Assessment for the Cycle 15 Refueling Outages in 2007 and 2008	November 29, 2007
	EA-03-009, Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors	February 11, 2003
	EPRI Report 1010087, Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139) August 2005	
	Certificate of Compliance dated 5/29/07 for ASME Code Section II SFA5.9 Class ER 308/308L welding material used on sacrificial layer on pressurizer surge nozzle	
	Certificate of Compliance 06369301 for ASME Code Section II, Part C SFA-5.14 Inconel 52M welding material used to deposit weld overlay on pressurizer surge nozzle	
	WSI Traveler No. 104532-TR-004 Pressurizer Surge Nozzle Repair Work Steps	0
	San Onofre Nuclear Generating Station Unit 3 Boric Acid Corrosion Control Program (BACCP) Health Report for Cycle 13: 12/29/2004 - 12/12/2006 May 8, 2007	
Letter from T. G. Hiltz (NRC) to R. M. Rosenblum (SCEC)	San Onofre Nuclear Generating Station Units 2 and 3 Re: Third 10-year Inservice Inspection Interval Request ISI-3-25, Use of Structural Weld Overlays and Associated Alternative Repair Techniques (TAC NOS MD2579 and MD2580)	June 12, 2007
Guide 5	System Component Walkdown	1
Generic Letter 88-05	Boric Acid Corrosion of Carbon Steel Pressure Boundary Components in PWR Plants	March 17, 1988
Information Notice 86-109, Supplement 3	Degradation of Reactor Coolant System Boundary Resulting from Boric Acid Corrosion	January 5, 1995
90022	Southern California Edison San Onofre Nuclear Generating Station Units 2 and 3: Reactor Coolant System Alloy 600 Material Management Program Plan	5

Section 1R07A: Heat Sink Performance

SO23-I-8.94	"Component Cooling Water Heat Exchanger Cleaning and Inspection"	Revision 8

Action Requests

071000587 071200968

Maintenance Orders

06040726000

Section 1R11: Licensed Operator Requalification

Procedures

Lesson Plan 2RS767	"Reactor Startup (Simulator)"	Revision 1
Lesson Plan 2RS768	"Plant Startup - Power Ascension from Mode 2 to 20% Power (Simulator)"	Revision 1

Action Requests

071000587

Maintenance Orders

06040726000

Section 1R12: Maintenance Effectiveness (Quarterly)

Procedures

SO23-3-3.23	"Diesel Generator Monthly and Semi-annual Testing"	Revision 30

Action Requests

070300161

Maintenance Orders

070300161-02   070300161-04

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

SO23-5-1.4	"Plant Shutdown to Hot Standby"	Revision 13
SO23-5-1.3.1	"Plant Startup from Hot Standby to Minimum Load"	Revision 26
Shutdown Nuclear Safety Program	"Defense in Depth Planning Sheets Unit 3 Cycle 14 Fall Midcycle Outage"	Revision 0
SO23-5-1.8.1	"Shutdown Nuclear Safety"	Revision 16
SO123-VIII-1	"Recognition and Classification of Emergencies"	Revision 26
SO123-XX-6	"Operator Work Around Program"	Revision 5
SO23-15-52.A	"Annunciator Panel 52A - FWCS/SBCS"	Revision 7
SO23-3-2.10	"Main Steam Isolation Valve Operation"	Revision 16
SD-SO23-110	"220 kV Switchyard System"	Revision 16
SSSPG-SO123-G-10	"Assessment of Offsite Capabilities Following a Natural Disaster"	Revision 0

Drawings and Calculations

SO23-507-6A-3-3	"MSIV, FWIV, and FWBV Hydraulic Dump Valve"	Revision M
SO23-507-6A-5-3	"MSIV, FWIV, and FWBV Hydraulic Dump Valve"	Revision M
40156FSO3	"High Pressure Feedwater System Feedwater Isolation Valve 3HV4051 Electro-Hydraulic Actuation System"	Revision 13
40141GSO3	"Main Steam System Electro-Hydraulic Valve 3HV-8204 System"	Revision 15
40141G	"Main Steam System Electro-Hydraulic Valve 2HV-8204 System"	Revision 17
M3C14 DID #1	"Barrier Map - Unit 3 Auxiliary Building (El. 50')"	Revision 0
M3C14 DID #1	"Barrier Map - Unit 3 Safety Equipment Building (El. 15'-6" & 5'-3")"	Revision 0

M3C14 DID #3	“Barrier Map - Train A Shutdown Cooling - Unit 3 Auxiliary Building (El. 50')”	Revision 0
M3C14 DID #3	“Barrier Map - Train A Shutdown Cooling - Unit 3 Safety Equipment Building (El. 15'-6" & 5'-3")”	Revision 0
M3C14 DID #3	“Barrier Map - Train B Shutdown Cooling - Unit 3 Auxiliary Building (El. 50')”	Revision 0
M3C14 DID #3	“Barrier Map - Train B Shutdown Cooling - Unit 3 Safety Equipment Building (El. 15'-6" & 5'-3")”	Revision 0
UFSAR Fig. 8.2-1	“One line Diagram - Switchyards	Revision 16

Action Requests

071000609 070500815 071100595 071201499 071000250

Section 1R15: Operability Evaluations

Procedures

SO23-2-16	“Operation of Waste Water systems”	Revision 20
SO23-20-4	“Auxiliary Feedwater System Operation”	Revision 22
Vendor Spec	“Kanaline SR PVC Hose”	undated
Vendor Spec	“Prosser Standard-Line Submersible Dewatering Pumps Series: 9-01000 & 9-01300"	June 2003
Vendor Spec	“Prosser Standard-Line Submersible Dewatering Pumps Series: 9-50000"	March 2001
SO23-3-3.31.6	“Main Feedwater System Valve Test”	Revision 7
SO23-3-3.31.4	“Main Steam Valve Testing - Offline”	Revision 7
SO123-XV-5.1	“Temporary Modification Control”	Revision 8
SO23-2-16	“Use of Temporary Sump Pumps”	Revision 20
SO123-XV-52	“Functionality Assessments and Operability Determinations”	Revision 7
SO23-3-3.60.4	“Saltwater Cooling Pump and Valve Testing”	Revision 9

Drawings and Calculations

40160A	“Auxiliary Feedwater System”	Revision 43

40160B	"Auxiliary Feedwater Steam Supply System"	Revision 21
DCP 52	"Plant design package to add trench eductor to TDAFW"	Revision 0

Action Requests

070500586	051200901	070500815	071100965	071000309	070500578
071000901					

Section 1R17: Permanent Plant Modifications (71111.17A)

Engineering Change Packages

060400474-40	Modify required actions in procedure SO23-5-1.7 to require MODE 3 entry for 1-3 inoperable MSSVs per steam generator	Revision 09/27/2006
060800177-07	Replacement of Diesel Generator Temperature Switch per SEE 000036	Revision 00
061001379-84	Install CCW Bypass Flow around the Unit 3 Letdown Heat Exchanger	Revision 00
061001842-16	Replace Existing TOL for Breaker 2BZ17	Revision 00
061001842-46	Replace Existing TOL for Breaker 3BZ25	

Drawings

S3-1023-ML-229, Sht 1	Letdown Heat Exchanger, Line 100: Valve 3TV-0223	Revision 15
S3-1203-ML-498, Sht 1	Component Cooling Water Line S3-1203-ML-498-4"-D-LL1 Sys 1203	Revision 0
S3-1203-ML-228, Sht 1	S3-1203-ML-228-8"-D-LL1, From Line 099 Valve 138 to Letdown Heat Exchanger	Revision 13
40123BS03	Reactor Coolant Chemical & Volume Control System No. 1208	Revision 29

Permanent Plant Modifications

020701289-37	Fix Position of Condensate Return Valve 2/3FV7546 and Remove 2/3FIC-7546	01/15/2007
040400696-17	Add ECP vent line at AFW pump motor outboard bearing housing to eliminate oil leak	09/25/2007

050901044-40	Technical specification bases change to allow substituting B00X for battery B007 and B008 for temporary battery outage	11/01/2005
051200901-07	Installation of a flow orifice downstream of 2PCV4716	07/25/2006
060500211-21	Replace vertical air tank S31319MV048	05/18/2006
060800603-29	Replace existing R3, R4 potentiometers with a new model in AVR for EDG.	03/07/2007
061101272-04	Install Pad Eye on beam over Safety Valve 3PSV0200	08/28/2007

Procedures

SO123-XV-44	10 CFR 50.59 and 72.48 Program	Revision 8

Tech Spec Amendments

PCN 576	Request to revise Main Steam Safety Valve Requirements and Actions (T.S. 3.7.1)	11/07/2006

Section 1R19: Postmaintenance Testing

Procedures

SO23-3-3.31.4	"Main Steam Isolation Valve-Offline Testing"	Revision 7
SO23-3-3.31.6	"Main Feedwater System Valve Test"	Revision 7
SO23-XXVII-33.14	"Procedure for the Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds"	Revision 1
WSI 104125-TR-004	"SONGS Pressurizer Surge Nozzle Repair Work Steps"	Revision 0
SO23-3-3.60.4	"Saltwater Cooling Pump and Valve Testing"	Revision 9
SO23-3-3.31.10	"Reactor Coolant Gas Vent System Test"	Revision 13

Miscellaneous

006-07	"Repair/Replacement Plan for Weld Overlay Repair to Pressurizer Surge Nozzle"	Revision 0
WPS -03-08-T-804-Bottom	"Weld Procedure Specification for Inconel to Stainless Steel"	Revision 0

WPS-08-08-T-001-ButterSS	“Weld Procedure Specification for Stainless Steel Butter”	Revision 0

“WPS-08-08-T-001-ButterSS Bead Log”

“WPS-03-08-T-804-Bottom Bead Log”

Section 1R20: Refueling and Outage Activities

Procedures

SO23-5-1.4	“Plant Shutdown to Hot Standby”	Revision 13
SO23-5-1.5	“Plant Shutdown from Hot Standby to Cold Shutdown”	Revision 28
SO23-3-1.8	“Draining the Reactor Coolant System”	Revision 26
SO23-5-1.8	“Shutdown Operations (Mode 5 and 6)”	Revision 17
SO23-3-3.29	“Determination of Reactor Shutdown Margin”	Revision 18
SO23-3-2.6	“Shutdown Cooling System Operation”	Revision 24
SO23-I-3.5	“Refueling Sequence”	Revision 14
SO23-5-1.3	“Plant Startup from Cold Shutdown to Hot Standby”	Revision 30
SO23-5-1.7	“Operating Instruction”	Revision 35
SO23-13-15	“Loss Of Shutdown Cooling”	Revision 16
SO23-V-8.15	“Containment Boric Acid Inspection”	Revision 2
	“ M3C14 Defense In Depth Planning Sheets”	Revision 0

Action Requests

071200870 071200486

Section 1R22: Surveillance Testing

Procedures

SO23-3-3.30.8	“Normal HVAC and Radiation Monitor Online Valve Test”	Revision 5
SO23-3-3.30.3	“Component Cooling Water Seismic Makeup Valve Test”	Revision 11
SO23-3-3.30.2	“Train A Saltwater Cooling Valve Test”	Revision 5
SO23-3-3.60.1	“High Pressure Safety Injection Pump 2MP-018 Testing”	Revision 7

SO23-3-3.60.3	"Component Cooling Water Pump 2MP-024 Test"	Revision 8
SO23-3-3.60	"Inservice Pump Testing Program"	Revision 8

Section 1R23: Temporary Plant Modifications

Procedures

ECP-07100097-3	"Replace grounded pressurizer heater S31201ME616 with pressurizer heater S31201ME614"	Revision 0

Drawings and Calculations

32631	"Elementary diagram reactor pressurizer backup heaters E124"	Revision 13
32632	"Elementary diagram reactor pressurizer backup heaters E128"	Revision 27
32171	"One line diagram pressurizer heaters distribution panels"	Revision 16
SO23-919-2-D58	"Heater element assembly"	Revision 4

Section 1EP6 Drill Evaluation

Procedures

SO123-VIII-1	"Emergency plan implementing procedures"	Revision 26
	"Emergency plan Drill 0704"	October 3, 2007
	"SONGS Emergency Plan"	Revision 16
SO123-0-A7	"Notification and Reporting of Significant Events"	Revision 5

Section 2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Action Request Documents

061001562, 061100484, 061101431, 070700048, 070700545, 070701137, 070701389, 070800826, 071000512, 071000551, 071000551, 071100267, 071100759, 071100760

Audits, Self-Assessments, Observations, and Surveillance Reports

Health Physics Division Self-Assessment Reports for First, Second, and Third Quarter 2007
Leader Observation Program Records from May through November 2007
SCES-006-07

Procedures

HP-I-2	Reactor Mode Change Checklist, Revision 14
SO123-VII-20	Health Physics Program, Revision 12
SO123-VII-20.6.1	Calculation of Dose from Skin Contamination, Revision 4
SO123-VII-20.7	Monitoring Internal Radiation Exposure, Revision 6
SO123-VII-20.9	Radiological Surveys, Revision 8
SO123-VII-20.9.6	Laboratory Analysis of Health Physics Air Samples, Revision 2
SO123-VII-20.11	Access Control Program, Revision 9
SO123-VII-20.11.1	Radiological Posting, Revision 8

Radiation Exposure Permits

A0707562000/200159, A0727070026, A0727070032/200101-12, A0819970001/200117-8

Miscellaneous

Selected Radiological Surveys during initial entry to Unit 2 Containment Refueling Outage
Unit 2 Shutdown Cooling Posting Plan

Section 2OS2: ALARA Planning and Controls (71121.02)

Action Request Documents

070400180, 070401109, 070401115, 070501042, 070600855, 070800568, 071101117, 071101118, 071101120, 071101121, 071101122, 071101124

Audits, Self-Assessments, Observations, and Surveillance Reports

Health Physics Division Self-Assessment Reports for First, Second, and Third Quarter 2007
Leader Observation Program Records from May through November 2007
SCES-006-07 and SOS-007-07

Procedures

HP-I-2	Reactor Mode Change Checklist, Revision 14
SO123-VII-20	Health Physics Program, Revision 11
SO123-VII-20.4	ALARA Program, Revision 4
SO123-VII-20.4.1	ALARA Design Change Reviews, Revision 4
SO123-VII-20.10	Radiological Work Planning and Controls, Revision 10

Radiation Exposure Permits

A0727070026, A1018940021

Miscellaneous

Reactor Coolant System Cobalt-58 Clean Up Curve for Unit 3 Midcycle 14

Unit 2 Refueling Cycle 15 ALARA Daily Current Performance for November 26 through 29, 2007

Section 4OA1: Performance Indicator Verification (71151)

Procedures

SO23-XV-24	Quarterly NRC Performance Indicator (PI) Process, Revision 5	
	"San Onofre Nuclear Generating Station; Station Performace Report"	2nd Quarter 2007
	"San Onofre Nuclear Generating Station; Station Performace Report"	3rd Quarter 2007

Miscellaneous

Quarterly Radiation Doses at the Site Boundary (Effluent Releases) for 2006 and 2007

Worker exposure records for radiological controlled area entries greater than 100 millirem

Section 4OA2: Identification and Resolution of Problems

Procedures

Policy Note 14	"Human Performance Strategic Plan"	November 9, 2007

LIST OF ACRONYMS

AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AR	Action Request
AVR	Automatic Voltage Regulator
BACC	boric acid corrision control
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
LER	Licensee Event Report
NCV	noncited violation
NDE	nondestructive examination
SSC	structure, system, and component
TS	Technical Specification
UFHA	Updated Fire Hazards Analysis
UFSAR	Updated Final Safety Analysis Report
VUHP	vessel upper head penetration