UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

August 5, 2011

Mr. Peter Dietrich
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 05000361/2011003 and 05000362/2011003

Dear Mr. Dietrich:

On June 23, 2011, 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 inspection report documents the inspection findings, which were discussed on July 1, 2011, with Mr. D. Bauder, Vice President and Station Manager, and other members of your staff.

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

Based on the results of this inspection, the NRC has identified one self-revealing and five NRC identified issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has determined that violations are associated with these issues. Additionally, two licensee-identified violations, which were determined to be of very low safety significance, are 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 a noncited violations, consistent with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the violations or the significance of the noncited violations, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the

NRC Resident Inspector at the facility. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the 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 you choose to provide one 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, your response should not include any personal privacy or proprietary, information so that it can be made available to the Public without redaction.

Sincerely,

D. Allen for

Ryan E. Lantz, Chief
Project Branch D
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report 05000361/2011003 and 05000362/2011003
w/Attachment: Supplemental Information

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Resident Inspector (John.Reynoso@nrc.gov)
Branch Chief, DRP/D (Ryan.Lantz@nrc.gov)
Senior Project Engineer, DRP/D (Don.Allen@nrc.gov)
SONGS Administrative Assistant (Heather.Hutchinson@nrc.gov)
Project Engineer, DRP/D (Zachery.Hollcraft@nrc.gov)
Project Engineer, DRP/D (David.You@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
Project Manager (Randy.Hall@nrc.gov)
Branch Chief, DRS/TSB (Dale.Powers@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
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OEMail Resource
ROPreports
RIV/ETA: OEDO (Jack.McHale@nrc.gov)
RIV RSLO (Bill.Maier@nrc.gov)
NSIR/DPR/EP (Eric.Schrader@nrc.gov)

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SRI:DRP/	RI:DRP/	SPE:DRP/	C:DRS/EB1	C:DRS/EB2
GWarnick	JReynoso	DAllen	TRFarnholtz	NFO'Keefe
DAllen for	**DAllen for**		**/RA/**	**/RA/**
8/4/11	8/4/11	8/5/11	7/29/11	8/1/11
C:DRS/OB	C:DRS/PSB1	C:DRS/PSB2	AC:DRS/TSB	C:DRP/
MSHaire	MCHay	GEWerner	DPowers	RLantz
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-361, 50-362
License:	NPF-10, NPF-15
Report:	05000361/2011003 and 05000362/2011003
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:	March 25 through June 23, 2011
Inspectors:	S. Achen, Resident Inspector C. Alldredge, Health Physicist P. Elkmann, Senior Emergency Preparedness Inspector N. Greene, Health Physicist G. Guerra, Emergency Preparedness Inspector S. Hedger, Operations Engineer Z. Hollcraft, Project Engineer C. Osterholtz, Senior Operations Engineer J. Reynoso, Resident Inspector J. Tapp, Resident Inspector E. Uribe, Reactor Inspector G. Warnick, Senior Resident Inspector J. Watkins, Reactor Inspector
Approved By:	Ryan E. Lantz Project Branch D Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000361/2011003, 05000362/2011003; 03/25/2011 – 06/23/2011; San Onofre Nuclear Generating Station, Units 2 and 3; Adverse Weather Prot., Maint. Effectiveness, Maint. Risk Assessment & Emergent Work, Eval. Perm. Plant Modifications, Surv. Testing

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by region-based inspectors. Six Green noncited violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” The cross-cutting aspect is determined using Inspection Manual Chapter 0310, “Components Within the Cross Cutting Areas.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the failure of operations personnel to establish adequate compensatory measures to restore or maintain operability as required by Procedure SO123-XV-52, “Operability Determination and Functionality Assessments,” Revision 18. Specifically, on November 12, 2010, although engineering identified measures were required to maintain water level below the steam line in the auxiliary feedwater trenches, no measures had been taken to stage pumps or limit flows into the trenches. On May 5, 2011, as a result of the inspectors’ questions, the licensee established additional compensatory measures including blocking storm drains that flow into the trench and staging sump pumps. This issue was entered into the licensee’s corrective action program as Nuclear Notification NN 201448584.

  The performance deficiency is more than minor, and therefore a finding, because it is associated with the protection against external events attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. During a design basis flooding from a probable maximum precipitation event, the auxiliary feedwater pump could be rendered inoperable. Using NRC Inspection Manual 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” the finding screened to a Phase 2 significance determination because it involved a potential loss of safety function. A Phase 2 was not appropriate for this external event. The senior reactor analyst determined that the finding had very low significance. This was based on information received from the licensee indicating that the precipitation intensity required to render the turbine-driven auxiliary feedwater pump non-functional had a return frequency well below 1.0E-6/yr. In the case of clogged drains, less intense rain could affect the function of the pump, but would likely not cause a transient. A bounding risk estimate indicated that the delta core damage frequency of this scenario was less than 1.0E-7/yr. The finding was determined to have a cross-cutting aspect in the area of human performance associated with the decision-making component because operations personnel failed to verify the validity of underlying assumptions for operability decision-making [H.1(b)](Section 1R01).

- Green. The inspectors identified that work instructions to replace a safety-related steam generator differential pressure transmitter did not contain adequate instructions to ensure that the scope of work was defined and the installed configuration would satisfy environmental qualification requirements. This involved multiple examples of a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings.” The inspectors also identified that the licensee had failed to maintain procedures intended to address previous problems damaging delicate insulation needed to maintain environmental qualification, and had failed to plan modifications needed to implement a planned improvement to the environmental qualification configuration, challenging maintenance workers during transmitter replacement. The licensee has entered this issue into their corrective action program as Nuclear Notification NN 201477774.

  Failure to provide adequate work instructions to replace a safety-related steam generator differential pressure transmitter to ensure that the scope of work was defined and the installed configuration would satisfy environmental qualification requirements is a performance deficiency. The performance deficiency affected the procedure quality attribute of the Mitigating Systems Cornerstone. This finding is more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern in that inadequate work instructions could result in a failure to meet the environmental qualification in systems needed to mitigate accidents. This finding was determined to have very low safety significance during a Phase 1 significance determination because it involved a qualification deficiency that was confirmed not to result in loss of operability or functionality. This finding has a cross-cutting aspect in the resources component in the human performance area because the licensee failed to ensure that procedures and other resources were adequate to assure nuclear safety. Specifically, the licensee did not ensure that complete, accurate, and up-to-date design documentation, procedures, and work packages were provided to support replacement activities for generator differential pressure transmitter 2PDT-0979-2 [H.2(c)](Section 1R12).

- Green. A self-revealing noncited violation of 10 CFR 50.65(a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” was identified for the failure of work control and operations personnel to adequately assess and manage the increase in risk associated with maintenance on the Unit 3 fish elevator. Specifically, on March 29, 2011, a stop log was installed in the Unit 3 intake structure without informing the Unit 2 control room operators or establishing measures to maintain adequate Unit 2 saltwater flow to ensure the operability of the component cooling water system. Immediate corrective actions included verifying and monitoring Unit 2 train A component cooling water operability and taking actions to restore saltwater cooling flow and component cooling water/saltwater cooling heat exchanger differential pressure to normal. This issue was entered into the licensee's corrective action program as Nuclear Notification NN 201395115.

The performance deficiency is more than minor and therefore a finding because it is associated with the operating equipment configuration control attribute of the Mitigating Systems Cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," Flowcharts 1 and 2, the finding was determined to have very low safety significance because the incremental core damage probability deficit was less than 1E-6 and the incremental large early release probability deficit was less than 1E-7. This finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because work control and operations personnel did not communicate decisions and the basis for decisions to individuals that needed to know the information in order to perform work safely and take appropriate risk management actions [H.1(c)](Section 1R13).

- Green. The inspectors identified that the licensee did not provide adequate long-time over-current protection for charging pumps 2P190 and 2P191 feeder cables. The finding involved a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control", for failure to translate applicable regulatory requirements and the design basis into specifications, drawings, procedures, and instructions. The licensee entered this issue into their corrective action program as Nuclear Notification NN 201443248.

  Failure to provide adequate long-time over-current protection for the feeder cables for charging pumps 2P190 and 2P191 is a performance deficiency. The performance deficiency affected the Mitigating Systems Cornerstone. The performance deficiency is more than minor and therefore a finding, because if left uncorrected, it would have the potential to lead to a more significant safety concern in that possible mechanical problems with the pump or motor could cause the affected cables to exceed their current limit and cause cable damage without tripping the associated breaker. The finding was determined to have very low safety significance during a Phase 1 significance determination because it involved a design deficiency that was confirmed not to have resulted in a loss of operability or functionality. No crosscutting aspect was identified because this issue is not reflective of current performance, since this condition has existed since construction (Section 1R17).

- Green. The inspectors identified a noncited violation of Technical Specification 5.5.1.1, "Procedures," for the failure of operations personnel to follow the surveillance program requirements for control element assembly testing, when a satisfactory verification of control element assembly movement was not obtained. Specifically, on May 8, 2011, operations personnel failed to refer to the abnormal procedure and the applicable action statement for Technical Specification 3.1.5, "Control Element Assembly (CEA) Alignment," as required by Procedure SO23-3-3.5, "CEA/Reactor Trip Circuit Breaker Operability Testing," Revision 18, when a satisfactory verification of control element assembly movement was not obtained. The licensee assumed the inability to move a control element assembly was due to a control rod drive mechanism control system failure without establishing a technical basis. This issue was entered into the licensee's corrective action program as Nuclear Notification NN 201474804.

The performance deficiency is more than minor and therefore a finding, because, if left uncorrected, it would have the potential to lead to a more significant safety concern since using presumptions of operability with inadequate factual basis could result in a condition prohibited by technical specifications. The finding is associated with the Mitigating Systems Cornerstone. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding was determined to have very low safety significance because the finding: (1) was not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because operations personnel failed to use conservative assumptions in decision-making when evaluating test results to determine an appropriate course of action [H.1(b)](Section 1R22).

- <u>Green</u>. The inspectors identified a noncited violation of Technical Specification 5.5.1.1, "Procedures," for the licensee's failure to establish procedures for the inability to drive control rods. Specifically, from initial licensing to May 2011, Abnormal Operating Instruction SO23-13-13, "Misaligned or Immovable Control Element Assembly," did not contain guidance to address an immovable control element assembly. This issue was entered into the licensee's corrective action program as Nuclear Notification NN 201497724.

  The performance deficiency is more than minor and therefore a finding, because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the inspectors determined the finding to have very low safety significance because the finding: (1) was not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors reviewed this finding for cross-cutting aspects and none were identified since the deficiency has existed since initial licensing and is not reflective of current performance (Section 1R22).

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by the licensee were 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 the corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 2 remained at essentially full power for the entire inspection period.

Unit 3 began the inspection period at essentially full power. On May 22, 2011, power was reduced to approximately 93 percent to replace the motor for heater drain pump 3MP058. Following completion of the planned maintenance, the unit returned to essentially full power on May 27, 2011, and remained there for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Summer Readiness for Offsite and Alternate-ac Power

a. Inspection Scope

The inspectors performed a review of preparations for summer weather for selected systems, including conditions that could lead to loss-of-offsite power and conditions that could result from high temperatures. The inspectors reviewed the procedures affecting these areas and the communications protocols between the transmission system operator and the plant to verify that the appropriate information was being exchanged when issues arose that could affect the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the transmission system operator and the plant's operations personnel during off-normal or emergency events
- The explanations for the events
- The estimates of when the offsite power system would be returned to a normal state
- The notifications from the transmission system operator to the plant when the offsite power system was returned to normal

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. The inspectors performed a walkdown of the switchyard to evaluate material condition of the offsite power sources. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the attachment. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- May 17 through June 7, 2011, Units 2 and 3, scheduling of generation risk activities, saltwater cooling and circulating water system performance and heating and ventilation air conditioning system outages

These activities constitute completion of one readiness for summer weather affect on offsite and alternate-ac power sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the UFSAR for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, on April 25-29, 2011, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed an inspection of the protected area to identify any modification to the site that would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one external flooding sample as defined in Inspection Procedure 71111.01-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the failure of operations personnel to establish adequate compensatory measures to restore or maintain operability as required by Procedure SO123-XV-52, "Operability Determination and Functionality Assessments," Revision 18. Specifically, on November 12, 2010, although engineering identified measures were required to maintain water level below the steam line in the auxiliary feedwater trenches, no measures had been taken to stage pumps or limit flow into the trenches.

Description. Between March 28 and April 29, 2011, inspectors completed an assessment of licensee's activities and actions using the guidance in

Temporary Instruction TI-183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event." The assessment included areas of the plant susceptible to flooding as a result of design basis rainfall or probable maximum precipitation. The inspectors' review included pertinent documented corrective actions associated with safety-related plant components and systems. On November 12, 2010, Nuclear Notification NN 201200176 documented the auxiliary feedwater system steam piping trench could be compromised as a result of flooding during the maximum storm runoff period of a probable maximum precipitation event. Since the capacity of the storm drain was only 3 inches in one hour compared to the maximum storm runoff during a design basis event (probable maximum precipitation) of 7 inches in one hour period, the trench could fill with water. Water in contact with the steam supply pipe could result in steam condensation inside the pipe and impact auxiliary feedwater pump operability.

The operability determination, documented in Nuclear Notification NN 201200176, concluded that the inadequate trench storm drain capability was a nonconforming condition and a latent design issue dating from 1985. Engineering personnel also concluded compensatory measures were required for rainfall intensity of greater than 3 inches per hour. These measures included use of a temporary sump pump, which was not staged. In addition, the operability determination acknowledged that if weather conditions were extreme and an operator could not access the auxiliary feedwater trench, the steam pipe would be submerged and the pump would be rendered inoperable. Operations personnel failed to validate their assumptions that monitoring of the trench water level and subsequent operator actions to remove water from the trench was sufficient to ensure the auxiliary feedwater pump remained capable of performing its specified safety function during design basis flood conditions.

On May 4, 2011, the inspectors questioned the adequacy of the compensatory measures and the licensees' conclusion that the auxiliary feedwater pump was operable since the pump may not be capable of performing its design functions during a probable maximum precipitation event. The inspectors' concerns were documented in Nuclear Notification NN 201448584.

On May 5, 2011, as a result of the inspectors' concerns, the licensee implemented actions including installation of sandbag barriers, blockage of storm drains feeding into the auxiliary feedwater trench, and installation of 10 gpm sump pumps. Other actions included daily monitoring of impending weather predicted by the weather service to ensure that no severe weather was forecasted within the next 48 hours. If severe weather of greater than 3 inches of water per hour was anticipated, operations personnel would be prepared to use the installed sump pumps. If water level could not be maintained, the turbine driven auxiliary feedwater pump would be started and maintained running to ensure continued operability.

<u>Analysis</u>. The failure of operations personnel to establish adequate compensatory measures to maintain operability of the turbine driven auxiliary feedwater pump was a performance deficiency. The performance deficiency is more than minor and therefore a finding because it is associated with the protection against external events attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to

prevent undesirable consequences. Using NRC Inspection Manual 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," the finding screened to a Phase 2 significance determination because it involved a potential loss of safety function. A Phase 2 was not appropriate for this external event. The senior reactor analyst determined that the finding had very low significance. This was based on information received from the licensee indicating that the precipitation intensity required to render the turbine-driven auxiliary feedwater pump non-functional had a return frequency well below 1.0E-6/yr. In the case of clogged drains, less intense rain could affect the function of the pump, but would likely not cause a transient. A bounding risk estimate indicated that the delta core damage frequency of this scenario was less than 1.0E-7/yr. The finding was determined to have a cross-cutting aspect in the area of human performance associated with the decision-making component because operations personnel failed to verify the validity of underlying assumptions for operability decision-making [H.1(b)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, from November 12, 2010, until May 5, 2011, operations personnel failed to follow Procedure SO123-XV-52, "Operability Determination and Functionality Assessments," Revision 18. Specifically, operations personnel did not properly establish compensatory measures for flood protection of safety-related components of the auxiliary feedwater system for all postulated flood levels and conditions. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 201448584, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361; 05000362/2011003-01, "Inadequate Compensatory Measures for a Design Nonconformance."

1R04 Equipment Alignments (71111.04)

Partial Walkdown

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- April 21, 2011, Unit 3, turbine driven auxiliary feedwater system
- May 12, 2011, Unit 2, emergency diesel generator fuel oil system train A
- June 20, 2011, Unit 2, motor driven auxiliary feedwater system train B

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- March 29, 2011, Unit 2, emergency diesel generator building
- March 30, 2011, Unit 3, emergency diesel generator building
- March 31, 2011, Unit 3, penetration building, elevation 63 feet
- April 4, 2011, Unit 2, safety equipment building, rooms 2-5 and 15
- April 22, 2011, Units 2 and 3, permanently mounted post seismic fire fighting equipment in radioactive waste and safety equipment buildings
- June 14, 2011, Unit 2, main steam isolation valve area

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan.

The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On June 16, 2011, the inspectors observed a fire brigade activation for a fire near auxiliary feedwater pump P-504 in the Unit 3 auxiliary feedwater building. The observation evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of preplanned strategies; (9) adherence to the preplanned drill scenario; and (10) drill objectives.

These activities constitute completion of one annual fire-protection inspection sample as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the UFSAR, the flooding analysis, and plant procedures to assess susceptibilities involving internal flooding; reviewed the corrective action program to determine if licensee personnel identified and corrected flooding problems; inspected underground bunkers/manholes to verify the adequacy of sump pumps, level alarm circuits, cable splices subject to submergence, and drainage for bunkers/manholes; and verified that operator actions for coping with flooding can reasonably achieve the desired outcomes. The inspectors also inspected the areas listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- March 23, 2011, Units 2 and 3, manholes TLAX07, TLAX08, AOX318, AOX309, AOX303, AKX222 and 3 switchyard underground bunkers

- April 25 - 29, 2011, Units 2 and 3, safety equipment buildings

These activities constitute completion of one inspection sample of flood protection measures and one annual sample of cables located in underground bunkers and manholes as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On June 7, 2011, the inspectors observed a crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance

- Crew's clarity and formality of communications

- Crew's ability to take timely actions in the conservative direction

- Crew's prioritization, interpretation, and verification of annunciator alarms

- Crew's correct use and implementation of abnormal and emergency procedures

- Control board manipulations

- Oversight and direction from supervisors

- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to preestablished operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- April 28 through May 4, 2011, Units 2 and 3, improper operation of 480V load center 2B24 circuit breaker
- December 8, 2010 through May 25, 2011, Unit 2, replacement of steam generator differential pressure transmitter 2PDT-0979-2 per Work Order WO 800501116

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or -(a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective

actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

Introduction. The inspectors identified that work instructions to replace a safety-related steam generator differential pressure transmitter did not contain adequate instructions to ensure that the scope of work was defined and the installed configuration would satisfy environmental qualification requirements. This involved multiple examples of a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V.

Description. In May 2010, Unit 2 steam generator differential pressure transmitter 2PDT-0979-2 failed in service while the unit was operating. This transmitter is used to measure reactor coolant system flow. Each steam generator had four differential pressure transmitters for providing flow measurement inputs to the reactor protection system. These inputs are used to cause a reactor trip signal if multiple transmitters sense a low flow condition.

Steam generator differential pressure transmitter 2PDT-0979-2 was located inside containment, and is subject to environmental qualification (EQ) requirements because it could be exposed to a harsh environment during some postulated accidents. In consideration of protecting the transmitter's sensitive electronics, a Conax electrical connector seal assembly (ECSA) was used to exclude steam and water. Drawing M-39619, "EQ Configuration Detail Rosemount transmitter Model 1153GD9, 1153HD5, 1153HD6," and Drawing M-39625, "EQ Configuration Detail Rosemount transmitter Model 1153GD9, 1153HD5, 1153HD6," depicted two different approved EQ configurations (the original and the improved configurations, respectively). The original design had the Conax seal attached between the transmitter and the flexible conduit, while the improved design relocated the Conax seal to be between the flexible conduit and the junction box.

The inspectors confirmed that all steam generator differential pressure transmitters in both units meet the EQ configuration requirements of the original configuration (Drawing M-39619), with slight individual variations that did not impact the environmental qualification, by examining recent photographs taken for that purpose.

Work Order WO 800501116 had been issued to troubleshoot and then replace the failed transmitter. The inspectors determined that this work order included instructions that required workers to change the configuration from the original configuration to the improved configuration by requiring that the installation work be done in accordance with Drawing M-39625.

The inspectors identified multiple problems with the work instructions included in or referenced by Work Order WO 800501116 that had the potential to impact the ability of the system to meet applicable EQ requirements. The work order, originally written to troubleshoot a transmitter problem, was revised to replace the transmitter without providing a detailed work scope description or work instructions. The inspectors noted that the work order contained only general instructions to perform the work. The inspectors were concerned that this level of detail was inadequate to ensure that work steps ensured the proper configuration and the quality specified for a safety-related, environmentally qualified transmitter.

Step 5.4.3 of work order 800501116 required the installation of the new transmitter and ECSA/conduit configuration to be installed in accordance with Drawing M-39625. Note 1.2 of Drawing M-39625 instructs workers to attach the Conax ECSA to junction box or conduit in accordance with CONAX instruction manual or station maintenance Procedure SO123-I-4.61, without specifying which sections were required to be performed. The inspectors noted that some sections of that procedure were not performed. Specifically, the work order did not specify whether to re-use or replace the Conax seal assembly, and the completed work documents did not specifically document that the Conax seal assembly was replaced.

The inspectors reviewed the licensee's actions to address operating experience contained in NRC Information Notice 88-89, "Degradation of Kapton Insulation." This information notice was written to document problems that had occurred at San Onofre in Unit 1. The electrical leads that were part of the Conax ECSA were coated with Kapton insulation, which was somewhat delicate. The inspectors noted that the work instructions did not include steps to install polyolefin heat-shrink sleeves that had been provided by the vendor to protect the Kapton insulation as corrective action for the event described in NRC Information Notice 88-89. The inspectors were unable to determine from the completed work documents whether the sleeves had been installed. In response to this concern, the licensee performed a containment entry in both units to examine 12 of the 16 transmitter systems. This inspection determined that all 12 of the steam generator differential pressure transmitters examined had polyolefin sleeves installed. Because installation of the polyolefin sleeves was corrective action for previous problems, it appeared that the licensee's corrective actions may not have been fully effective since there was not a clear requirement in station procedures to ensure that the sleeves would be installed.

The inspectors noted that Work Order WO 800501116 required workers to install the replacement transmitter in accordance with Drawing M-39625. Maintenance workers interviewed by the inspectors indicated that it was not clear which details from these drawings were critical to quality or environmental qualification. Licensee staff stated that these drawings were intended to be representational of the typical configurations, rather than to be considered assembly drawings, and that minor configuration differences existed between the as-built transmitter configurations and the drawings. The licensee stated that these drawings were not intended to provide assembly guidance. The inspectors noted that the drawings did not include conduit couplings, conduit elbows, conduit reducing fittings, and heat shrink polyolefin sleeves. These parts were needed to meet the approved EQ configuration. The inspectors determined that these parts were installed with each transmitter without instructions to do so or documentation to show the as-left configuration.

Both Drawings M-39619 and M-39625 required installing a Conax ECSA in accordance with either the vendor installation manual (Conax Installation Manual IPS-725 Revision J) or Maintenance Procedure SO123-I-4.61. Work Order WO 800501116 showed that the work was performed using Maintenance Procedure SO123-I-4.61. The inspectors identified that Maintenance Procedure SO123-I-4.61 did not include the vendor manual requirements for installing and heat shrinking protective polyolefin sleeves over the Kapton-insulated wires, nor did it require use of a conduit coupling to protect the rigid part of the leads at the seal, both of which were required to be installed per the vendor manual. Therefore, the inspectors concluded that the maintenance procedure was inadequate.

The inspectors determined that both the existing and improved configurations were approved EQ configurations. However, the licensee had recognized in the early 1990s that the as-built configurations varied somewhat from transmitter to transmitter. These configurations were accepted in Non-Conformance Reports NCR 90050183 (Unit 2) and NCR 2-3114 (Unit 3) after additional testing was performed. At that time, the licensee decided to change all steam generator differential pressure transmitters to the improved configuration shown in Drawing M-39625 when the transmitters reach the end of their EQ qualified life. The inspectors determined that all of the steam generator differential pressure transmitters in both units were replaced in 2002 when the transmitters reached their end of qualified life, but none of the replacement transmitters were installed per Drawing M-39625. The inspectors noted that such a configuration change would involve more work and planning than was typical for routine maintenance. Each transmitter system had unique configuration details, so conversion would be expected to involve detailed modification planning. The inspectors determined that, even though this licensee-specified requirement existed for years, the licensee had not created engineering change packages or scheduled the work to implement the configuration changes required by Drawing M-39625. The licensee had decided not to change Drawing M-39619 to reflect the actual as-built configuration of the transmitters because they expected the plant to be changed to match Drawing M-39625. As a result, Drawing M-39619 did not show the individual variations, so it could only be considered representative.

The inspectors noted that the licensee had discretion to use either approved configuration. However, there were actions associated with both configurations that were left undone, and the failure to complete those actions created confusion for the maintenance workers who had to use the documents that were not updated.

- Changing all detectors to the improved configuration necessitated creating and scheduling detailed plant modifications, which was not done

- Allowing detectors to continue to use the original configuration necessitated drawing changes to show the as-built configuration details

- Having work instructions that did not contain detailed assembly details, but rather referred to Drawings M-39619 and M-39625, necessitated changes to those drawings to provide the assembly details needed to satisfy the EQ requirements

Despite recognizing the conflict created by these three issues, maintenance personnel did not document it in the corrective action program. Rather than requesting that the conflicts be addressed, maintenance personnel proceeded to do the work anyway.

While responding to the inspectors' concerns, the licensee identified that a work group supervisor had made pen-and-ink changes to Work Order WO 800501116. The changes revised the instructions that originally required installing the replacement transmitter in accordance with Drawing M-39625 (the improved configuration) to state that it should be installed per Drawing M-39619 (the existing configuration). This change violated Procedure SO123-I-1.3, which prohibited making pen-and-ink changes to work orders involving EQ components. This is further discussed in Section 4OA7. The inspectors determined through interviews that technical agreement to make the change was obtained from knowledgeable maintenance engineering personnel to install transmitter 2PDT-0979-2 in the original configuration.

The inspectors concluded that work planning to replace transmitter 2PDT-0979-2 was inadequate to ensure that it was installed in a quality manner. Specifically:

- The work order referenced entire procedures without specifying which parts were to be performed.

- The work order did not provide necessary assembly details, but instead used EQ drawings that were not intended for that purpose and did not contain all the necessary details.

- The work order required installing the transmitter in a configuration that required modifying the plant without providing the details to perform the modification.

- The work order did not require sufficient documentation of completed work to allow a reviewer to determine what the final configuration of the completed work was or that it met the necessary quality and EQ requirements.

These shortcomings necessitated containment entries and intrusive examination of the systems in order for inspectors to confirm that the final configuration met the proper quality and environmental qualification requirements, including passing appropriate post-maintenance testing.

The licensee has entered this issue into their corrective action program as Nuclear Notification NN 201477774.

Analysis. The failure to provide adequate work instructions to replace a safety-related steam generator differential pressure transmitter to ensure that the scope of work was defined and the installed configuration would satisfy environmental qualification requirements was a performance deficiency. The performance deficiency affected the procedure quality attribute of the Mitigating Systems Cornerstone. This finding is more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern in that inadequate work instructions could result in a failure to meet the environmental qualification in systems needed to mitigate accidents. Using the NRC Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance (Green) during a Phase 1 significance determination because it involved a qualification deficiency that was confirmed not to result in loss of operability or functionality.

This finding has a cross-cutting aspect in the resources component in the human performance area because the licensee failed to ensure that procedures and other resources were adequate to assure nuclear safety. Specifically, the licensee did not ensure that complete, accurate, and up-to-date design documentation, procedures, and work packages were provided to support replacement activities for generator differential pressure transmitter 2PDT-0979-2 [H.2(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Work Order WO 800501116, Drawing M-39619, "EQ Configuration Detail Rosemount Transmitter Model 1153GD9, 1153HD5, 1153HD6," Revision 4, Drawing M-39625, "EQ Configuration Detail Rosemount Transmitter Model 1153GD9, 1153HD5, 1153HD6," Revision 3, and Maintenance Procedure SO123-I-4.61, "Conax Seal Assembly-Removal, Cleaning, Inspection, Repair, and Installation," Revision 1, were used to replace Unit 2 steam generator differential pressure transmitter 2PDT-0979-2; an activity affecting quality.

Contrary to the above, in May 2010, the licensee failed to ensure that instructions, procedures, and drawings were appropriate to the circumstances for an activity affecting quality. Specifically, as part of work to replace Unit 2 steam generator differential pressure transmitter 2PDT-0979-2, instructions, procedures and drawings were inappropriate to ensure that the transmitter system would satisfy applicable EQ requirements as follows : (1) Work Order WO 800501116 did not contain an adequate work scope description or specify which sections of the referenced documents to execute; (2) Drawings M-39619 and M-39625 did not contain the assembly details needed to ensure that the approved EQ configuration would be met, and allowed performing work in accordance with either of two procedures which did not contain the same requirements; and (3) Maintenance Procedure SO123-I-4.61 did not contain critical requirements from the Conax electrical seal assembly vendor manual needed to ensure EQ requirements were met.

Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 201477774, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361; 05000362/2011003-02, "Inadequate Work Instructions to Ensure Environmentally Qualified Configuration."

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

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- April 5, 2011, Unit 3, intake structure stop log installation
- April 6, 2011, Unit 2, containment entry for planned maintenance on charging line to reactor coolant loop 1A control valve HV9203
- April 18-20, 2011, Units 2 and 3, main generator on-line Western Electricity Coordinating Council generator test

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. Findings

Introduction. A Green self-revealing noncited violation of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified for the failure of work control and operations personnel to adequately assess and manage the increase in risk associated with maintenance on the Unit 3 fish elevator, which resulted in a lack of appropriate risk mitigation actions and a challenge to the operability of the Unit 2 component cooling water system.

Description. On March 29, 2011, the licensee installed a stop log in the Unit 3 intake structure in order to perform planned maintenance on the Unit 3 fish elevator. The stop log installation caused an influx of debris into Unit 2 train A saltwater cooling pump which is located in the Unit 3 intake structure. This caused an unexpected increase in the Unit 2 component cooling water/salt water cooling heat exchanger differential pressure and a decrease in the associated saltwater cooling flow. Consequently, the unexpected influx of debris challenged the operability of the Unit 2 train A component cooling water system. In addition, Unit 2 train B component cooling water system was inoperable for emergent maintenance. Licensee personnel performed recovery actions and were able to return the degraded parameters to normal levels within approximately thirty minutes. If conditions degraded on the train A component cooling water system to the point where it became inoperable, the licensee would have been required to enter Technical Specification 3.0.3 and shut down the plant. During the work planning for the stop log installation, the risk assessments performed were narrowly focused and did not take into account the potential impact on nearby equipment and on both units. Operating experience from past stop log installations and the known disturbance of debris occurring during this work was not understood by licensee personnel in general, only certain individuals had that knowledge. The work activity was assessed as a high industrial risk but the nuclear safety risk was not addressed. No risk mitigating actions were in place to manage the increase in risk to the saltwater cooling and component cooling water systems, such as increasing the risk awareness to licensee personnel and establishing compensatory measures to ensure continued operability of the potentially affected safety-related systems. When the Unit 3 control room was made aware of the commencement of the activity, because the risk to the Unit 2 saltwater cooling system was not understood by the Unit 3 control room, the Unit 2 control room supervisor was not informed and the activity proceeded forward.

Analysis. The failure of work control and operations personnel to adequately assess and manage the risk associated with maintenance on the Unit 3 fish elevator was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the operating equipment configuration control attribute of the Mitigating Systems Cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," Flowcharts 1 and 2, the finding was determined to have very low safety significance (Green) because the incremental core damage probability deficit was less than 1E-6 and the incremental large early release probability deficit was less than 1E-7. This finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because work control and operations personnel did not communicate decisions and the basis for decisions to individuals that needed to know the information in order to perform work safely and take appropriate risk management actions [H.1(c)].

Enforcement. Title 10 CFR 50.65(a)(4), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," requires, in part, that the licensee assess and manage the increase in risk that may be associated with performing maintenance activities prior to performing the maintenance. Contrary to the above, on

March 29, 2011, work control and operations personnel failed to adequately assess and manage the increase in risk associated with the Unit 2 component cooling water system prior to performing maintenance on the Unit 3 fish elevator. As a result, appropriate risk mitigating actions were not implemented which challenged the operability of the Unit 2 component cooling water system. Immediate corrective actions included verifying and monitoring Unit 2 component cooling water train A operability and taking actions to restore saltwater cooling flow and component cooling water/saltwater cooling heat exchanger differential pressure to normal. In addition, the licensee reviewed all future activities related to the intake structure to prevent recurrence, required site-wide required reading of the requirements for risk assessment completion, and performing management oversight of communications between each unit's control room including to and from the work process center. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 201395115, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361; 05000362/2011003-03, "Failure to Appropriately Assess and Manage Risk for Work in Unit 3 Intake."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- December 17, 2010, Unit 2, salt water cooling pump fans 2MA-372 and 2MA-373 obstructions
- April 11, 2011, Units 2 and 3, potential miscalibrated loss of voltage signal relays installed in plant
- April 19, 2011, Units 2 and 3, improper operation of 480 volt breakers
- April 20-22, 2011, Unit 2, reactor coolant pump P004 upper seal degradation
- April 26-28, 2011, Units 2 and 3, train A component cooling water heat exchanger fouling and high delta pressure following cycling of saltwater water cooling pump 3MP307
- April 28 through May 11, 2011, Unit 3, high pressure safety injection pump instrument line fastener found broken

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and UFSAR to the licensee personnel's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six operability evaluations inspection samples as defined in Inspection Procedure 71111.15-05.

b. Findings

No findings were identified.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

a. Inspection Scope

Between December 17, 2010, and May 25, 2011, the inspectors reviewed the circumstances surrounding changes made to design basis calculations to de-rate the current-carrying capacity (ampacity) of the feeder cables for charging pumps 2P190 and 2P191. The inspectors reviewed calculations, drawings, pump performance curves, pump testing procedures, breaker testing procedures, maintenance records, and modifications to fire-wrap raceways. The inspectors had extensive discussions with the design engineering personnel on the history associated with this issue.

b. Findings

Introduction. The inspectors identified that the licensee did not provide adequate long-time over-current protection for charging pump 2P190 and 2P191 feeder cables. The finding involved a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for failure to translate applicable regulatory requirements and the design basis into specifications, drawings, procedures, and instructions.

Description. The charging pump motor electrical power circuits were designed as safety-related equipment with 100 horsepower motors. The system provides borated water makeup to the reactor coolant system, performing makeup, reactivity control, chemistry control, and a number of other functions.

Basic electrical circuit design philosophy calls for providing protection that will de-energize the circuit before serious damage occurs in the event of unusual conditions, such as one that would cause excess electrical current. For a motor, the protective setpoint is normally based on the motor characteristics, and the associated power cable(s) would be selected with a current rating that exceeds the protective setpoint so that the cable is not the limiting component in the circuit. The original design for the charging pump circuits was to select components for a 40 year life. Therefore, since the pump would run at 60 amps and need no more than 70 horsepower, a 100 horsepower motor was selected that could run continuously for 40 years at 120 amps, and cables that could carry 242 amps continuously for 40 years were installed.

The original design for the circuits associated with charging pumps 2P190 and 2P191 met the above design principles, although the architect-engineering firm did not select the most appropriate feeder breaker to protect the motor.

Late in the construction period, the plant was modified to install Cerablanket wrapping in certain locations to meet the separation requirements of Regulatory Guide 1.75, "Physical Independence of Electric Systems." The cables for charging pumps 2P190 and 2P191 were wrapped with Cerablanket where the six cables were routed together inside a 3-inch conduit. The Cerablanket material was installed to prevent electrical contact between cables. Cerablanket acts as a thermal insulator, so it does not allow the rigid metal conduit to dissipate/conduct the internal heat that is generated by the enclosed conductors as efficiently as it would in air. Therefore, when energized, the cables will be at a higher temperature than if the conduit were not insulated. In order to prevent the conductors and insulation from being damaged, the maximum allowable ampacity of the conductors must be reduced (de-rated). Bechtel recognized this and modified the calculations to administratively limit the affected cables' maximum ampacity to retain the intended 40 year life. This was not done for every cable tray; Bechtel used judgment to identify only the most limiting cable tray for each cable, and then calculated the current limit for that one location. Bechtel lowered the administrative current limit to 223 amps in Calculation E4C-051, "Low Voltage Power Circuit Breaker Settings," in 1977. The location used for de-rating calculations was later found by the licensee to not be the most limiting location for charging pump 2P190 and 2P191 feeder cables.

In 1999 the licensee updated Calculation E4C-051 to include all cable segments in the calculations. The licensee identified that the limiting path was a wrapped 3-inch conduit, (designated OU2GUAF04) located in room 305, which contained the six single conductor (3 conductors per motor) 3/0 American Wire Gauge feeder cables for both charging pumps 2P190 and 2P191 from train A (charging pump 2P191 is a "swing pump" that can be powered from either train). The revised calculation de-rated the feeder cable ampacity to 101 amps based upon an assumed cable rated temperature of 90°C. However, the license determined in 2010 that the installed cable was actually rated for 75°C, and further de-rated the feeder cable ampacity to 96.3 amps. This resulted in the current limit for the cable to be below the protective setpoint and below the maximum rated motor current (full load motor current). It is important to note that the full load motor current is above the normal running current, and intentionally includes a design capacity for the motor to safely run above the expected running current.

The feeder breakers for the charging pumps had a long-time overcurrent trip setpoint of 150 percent of full load current (180 amps). The original/existing feeder breakers could not be adjusted to a trip setting below 180 amps. Calculation E4C-099 specified that the long-time trip setpoint should be set between 125 and 140 percent for motors with a 1.15 service factor. While this higher breaker trip setting did not provide the normal protection for the motor, it did provide protection for the cables prior to de-rating, since the original cable ampacity was greater than 180 amps.

The licensee performed a number of evaluations, in 1999 and again in 2010 after revising the cable ampacity calculations, when they recognized that the cable ampacity was less than the breaker long-time overcurrent setpoint. The charging system design is such that the positive-displacement pump will pump water at a steady flow rate under all conditions, so motor current should not significantly depart from the normal operating current of 60 amps. Since the normal operating current was below the de-rated limit for the cable, the licensee concluded that it was acceptable to leave the breaker settings as originally set in Calculation E4C-099, “SR 480V Power Circuit Breaker Settings,” and did not change the breaker setting.

The inspectors concluded that the licensee did not adequately consider abnormal operation when considering the proper level of protection.

In 2006, the NRC component design basis inspection team concluded that the licensee had failed to provide adequate protection for a number of de-rated cables, including the charging pump 2P190 and 2P191 feeder cables. In response, the licensee adopted a new philosophy that departed from the original design intent. Rather than consider the cables to be expected to last for 40 years (the original license period for the plant), the licensee placed cables that had low margin due to de-rating into a predictive maintenance program to monitor their performance for aging effects.

This predictive maintenance program recorded and trended motor running currents to ensure they did not exceed the de-rated cable ampacity. The inspectors confirmed that no test had shown any change in operating current compared to the baseline data. This program also performs megger testing of the cable insulation, which had shown no degradation. While this performance-based method could detect aging effects from unplanned departures from normal running conditions, the inspectors concluded that this would only be useful in identifying slowly-developing aging effects.

The inspectors reviewed the charging pump circuit, and concluded that the feeder breaker long-time overcurrent setpoint did not provide adequate protection. Specifically, the long-time overcurrent setpoint of 180 amps could permit the cables, which have a de-rated ampacity of 96.3 amps, to be exposed to over-currents such that they could exceed their 75°C rated temperature causing cable degradation and reduced life. It is possible to have a physical problem with the charging pump or its motor that causes the motor to run at an abnormally high current. This could include a bad bearing, tight pump packing, or mechanical rubbing. As long as the current remains below 180 amps the breaker would not trip. The inspectors reviewed the maintenance history of the charging pumps and concluded that the charging pump feeder breakers have not seen this high current. The inspectors concluded that such a condition would be detected since the charging pump motor currents are displayed in the control room where they are monitored by the operators, and operators monitor the performance of the charging pumps during rounds each shift.

Through a combination of calculations performed by the inspectors and the licensee and a cable test performed by the licensee, the inspectors concluded the cable could exceed its 75°C rated temperature and experience premature aging of the cable insulation during problems that cause abnormal operation at high current below the long-time overcurrent trip setpoint.

At the time of the inspection, the licensee had Engineering Changes ECP 80071317 and 800239532 approved for replacing the breakers for the charging pumps. These engineering changes were being performed to address other problems associated with the existing breakers. The licensee documented the issue in Nuclear Notification NN 201443248 and planned to expedite the replacement of these breakers by the end of 2011. The licensee was also evaluating removing the existing Cerablanket insulating material and replacing it with a different material that will provide more ampacity margin for the cable.

Analysis. The failure to provide adequate long-time over-current protection for the feeder cables for charging pumps 2P190 and 2P191 was a performance deficiency. This performance deficiency affected the design control and plant modifications attributes of the Mitigating Systems Cornerstone. The performance deficiency is more than minor and therefore a finding because, if left uncorrected, it would have the potential to lead to a more significant safety concern in that possible mechanical problems with the pump or motor could cause the affected cables to exceed their current limit and cause cable damage without tripping the associated breaker. Using the NRC Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance (Green) during a Phase 1 significance determination because it involved a design deficiency that was confirmed not to result in loss of operability or functionality.

No crosscutting aspect was identified because this issue is not reflective of current performance, since this condition has existed since construction.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. It further requires that changes shall be subject to design control measures commensurate with the original design.

The Updated Final Safety Analysis Report, Appendix 3.2A, states that the onsite power system is designed in accordance with IEEE Standard 308-1974. IEEE 308-1974 states that protective devices shall be provided to limit the degradation of the Class 1E power systems.

Section 5.3.2 of Calculation E4C-099, "Safety Related 480V Power Circuit Breaker Settings," Revision 3, stated: "The thermal overload protection of cables shall be evaluated by comparing the long time pick-up settings for each trip device with the cable ampacity; a cable is considered to be protected by a trip device if the maximum pick-up of the long-time unit is less than the rated ampacity of the cable."

Contrary to the above, since original construction, the licensee has failed to ensure that all applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures and instructions. Specifically, Calculation E4C-099, "SR 480V Power Circuit Breaker Settings," Revision 3, failed to ensure that the supply breakers for charging pumps 2P190 and 2P191 had a long time pick-up setting lower than the de-rated ampacity of 96.3 amps in order to assure the required protection of the feeder cables.

Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 201443248, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361; 05000362/2011003-04, "Failure to Provide Adequate Long-Time Over-Current Protection for the Feeder Cables for Charging Pumps 2P190 and 2P191."

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- May 4, 2011, Unit 3, train A component cooling water heat exchanger
- May 5, 2011, Unit 3, postmaintenance test of main steam supply to auxiliary feedwater pump 3K007 isolation valve
- May 10, 2011, Unit 2, control element assembly CEA 25 operability test per Procedure SO23-3-3.5, "CEA/Reactor Trip Circuit Breaker Operability Testing," Revision 18

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed
- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and technical specifications to ensure that the surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- April 14, 2011, Units 2 and 3, post-seismic and B.5.b fire fighting equipment inspection per Procedure SO123-XIII-54, "Fire Equipment Inspection," Revision 23
- April 20, 2011, Units 2 and 3, annual and monthly inspections of skid mounted water pump MP1058 per Maintenance Orders MOs 800617591 and 800643846
- April 22, 2011, Unit 3, semiannual test of ESF subgroup relays K-112A, K-625A, and K-725A
- May 5, 2011, Unit 2, nuclear power excore channel A calibration
- May 8, 2011, Unit 2, Procedure SO23-3-3.5, "CEA/Reactor Trip Circuit Breaker Operability Testing," Revision 18
- June 8, 2011, Unit 2, high pressure safety injection charged piping monthly verification per Procedure SO23-3-3.8, "Safety Injection Monthly Tests," Revision 26

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

1. Control Element Assembly Operability Surveillance Test

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.5.1.1, "Procedures," for the failure of operations personnel to follow the surveillance program requirements for control element assembly testing, when a satisfactory verification of control element assembly movement was not obtained.

Description. On May 8, 2011, operations personnel performed a surveillance to demonstrate operability of the Unit 2 control element assemblies (CEAs) to satisfy the requirements of Technical Specification Surveillance Requirement 3.1.5.3. Control element assembly operability was demonstrated by exercising each full length CEA not fully inserted at least 5 inches in any one direction. The surveillance was performed using Procedure SO23-3-3.5, "CEA/Reactor Trip Circuit Breaker Operability Testing," Revision 18. At approximately 1800 hours, when attempting to withdraw CEA 25, the CEA stopped after 1 step with no illumination of the upper electrical limit light. Operations personnel verified that the CEA moved out 1 step, then attempted one more withdrawal and one step insertion with no further movement noted. Consequently, satisfactory verification of control element assembly movement was not obtained per Procedure SO23-3-3.5. Just prior to testing CEA 25, instrument and control personnel had suspended obtaining coil traces since it was close to the end of their shift. As a result, control rod drive mechanism system performance information was not immediately available to operations personnel when CEA 25 failed to move.

Procedure SO23-3-3, “Operations Surveillance Program Requirements,” Revision 16, Section 6.5, “Unsatisfactory Surveillance Actions,” stated that the senior reactor operator or operations supervisor must be immediately notified when acceptance criteria results were unsatisfactory, and to review and follow the applicable technical specification requirements. Procedure SO23-3-3.5, Step 3.2, stated, in part, “IF test results are not satisfactory, THEN refer to SO23-13-13, Misaligned Control Element Assembly, AND follow the Action statements of Tech. Spec. LCO 3.1.5 and LCS 3.1.105.” Rather than follow the above requirements of Procedures SO23-3-3 and SO23-3-3.5 when satisfactory verification of control element assembly movement was not obtained, operations personnel applied an initial presumption of operability based on their belief that CEA 25 was still trippable, and therefore operable. The belief that CEA 25 was still trippable was based on an assumption that the issue was associated with control rod drive mechanism control system and that the CEA was only electrically immovable and remained trippable and aligned. An additional basis was that there were no initial indications that the CEA was unable to be tripped, even though the test to demonstrate freedom of movement could not be completed satisfactorily. The inspectors reviewed the initial presumption of operability documented in Nuclear Notification NN 201453658 and concluded that operations personnel established their basis for CEA operability using unconfirmed assumptions that lacked factual information. Further, the presumption that CEA 25 was still trippable was the justification used by operations personnel to not follow the procedure requirements for a failed surveillance, but instead, suspend the test while troubleshooting the equipment issue proceeded.

At approximately 2220 hours, instrument and controls personnel performed equipment troubleshooting by obtaining and reviewing coil traces that provided factual evidence to confirm operations personnel’s assumptions and establish reasonable expectation that the CEA was still trippable and operable. The inspectors observed that from approximately 1800 hours, when operations personnel were unable to demonstrate freedom of movement through satisfactory surveillance testing, until approximately 2220 hours, when troubleshooting determined that the cause was due to an issue with the control rod drive mechanism control system, CEA 25 was in a state of indeterminate operability. Specifically, the initial presumption of operability was largely based on assumptions that relied on future troubleshooting results, rather than being based on specific sets of facts to provide a reasonable expectation of operability.

The inspectors reviewed Operations Standards Manual OSM-12, “Operator Fundamentals,” Revision 13, Section 8.0, and observed that decision-making by operations personnel in response to their inability to move CEA 25 was not consistent with the following conservative decision-making structure listed in Section 8.2.1:

- Equal and sufficient consideration is given to all potential consequences and/or hazards that may affect nuclear safety

- Operating experience and/or previous experience applicable to the plant condition are well known, understood, and in particular, how the event relates to the present circumstances
- A questioning attitude and personal review of mutually supporting indications, alarms, and plant response support the expected results
- Data or information is factual and precise, allowing for a complete and thorough understanding of the consequences
- As the level of uncertainty increases, so must the level of conservatism

Analysis. The failure of operations personnel to follow surveillance procedures when testing safety-related structures, systems, or components to verify operability was a performance deficiency. The performance deficiency is more than minor and therefore a finding because, if left uncorrected, it would have the potential to lead to a more significant safety concern since using presumptions of operability with inadequate factual basis to make operability decisions with respect to procedure and technical specification compliance could result in a condition prohibited by technical specifications. The finding is associated with the Mitigating Systems Cornerstone. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding was determined to have very low safety significance because the finding: (1) was not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because operations personnel failed to use conservative assumptions in decision-making when evaluating test results to determine an appropriate course of action [H.1(b)].

Enforcement. Technical Specifications 5.5.1.1, "Procedures," 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 8.b, requires procedures for the performance of surveillance tests, inspections, and calibrations. Procedure SO23-3-3.5, "CEA/Reactor Trip Circuit Breaker Operability Testing," Revision 18, implemented the requirements of Technical Specification Surveillance Requirement SR 3.1.5.3. Contrary to the above, on May 8, 2011, operations personnel failed to accomplish an activity affecting quality in accordance with prescribed instructions, procedures, or drawings. Specifically, operations personnel failed to follow Procedure SO23-3-3.5, to refer to Abnormal Operating Instruction SO23-13-13, "Misaligned or Immovable Control Element Assembly," Revision 13, and the applicable action statement for Technical Specification 3.1.5, "Control Element Assembly (CEA) Alignment," when a satisfactory verification of control element assembly movement was not obtained per Technical Specification Surveillance Requirement 3.1.5.3. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification

NN 201474804, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361/2010005-05, "Failure to Follow Procedure Requirements for a Failed Surveillance."

2. Lack of Adequate Procedures to Respond to the Inability to Drive Control Rods

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.5.1.1, "Procedures," for the licensee's failure to establish adequate procedures for the inability to drive control rods. Specifically, from initial licensing to May 2011, Abnormal Operating Instruction SO23-13-13, "Misaligned or Immovable Control Element Assembly," did not contain guidance to address an immovable control element assembly.

Description. During follow up of an incident that occurred on May 8, 2011, where operations personnel were unable to move CEA 25 while attempting to satisfy Surveillance Requirement SR 3.1.5.3, to verify "trippability" of control rods, inspectors identified a deficiency in the licensee's Abnormal Operating Instruction SO23-13-13, "Misaligned or Immovable Control Element Assembly," Revision 13. When the surveillance was not satisfied, operations personnel determined that Abnormal Operating Instruction SO23-13-13 was not applicable to an immovable control element assembly. Inspectors identified, through interviews and statements provided by operations personnel, that the abnormal operating instruction was intended for misaligned control element assemblies and did not provide appropriate guidance for when a control rod was found to be immovable.

The inspectors reviewed Surveillance Operating Instruction SO23-3-3.5, "CEA/Reactor Trip Circuit Breaker Operability Testing," Revision 18, as well as training documentation for operations personnel, and identified several discrepancies of references to the title of the abnormal operating instruction. All references made to Abnormal Operating Instruction SO23-13-13 in the documents reviewed by inspectors referenced the procedure as "Misaligned Control Element Assembly," excluding the "Immovable" portion of the title. In a further review of training documentation for initial operator licensing as well as requalification training for licensed operations personnel, inspectors identified that training being provided to operations personnel did not include adequate guidance on how to respond to the inability to drive control rods utilizing Abnormal Operating Instruction SO23-13-13.

Inspectors reviewed the content of Abnormal Operating Instruction SO23-13-13, and determined that the procedure would not adequately provide guidance for operations personnel to respond appropriately to an immovable control element assembly or the inability to drive control rods. The licensee acknowledged the procedural deficiency and generated Nuclear Notification NN 201497724 to evaluate and correct the deficiency.

Analysis. The failure to provide adequate procedures for combating emergencies and other significant events was a performance deficiency. The performance deficiency is more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the inspectors determined the finding to have very low safety significance because the finding: (1) was not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors reviewed this finding for cross-cutting aspects and none were identified since the deficiency has existed since initial licensing and is not reflective of current performance.

Enforcement. Technical Specifications 5.5.1.1, "Procedures," 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 6.m, recommends procedures for combating emergencies and other significant events, including the inability to drive control rods. Contrary to this requirement, from initial licensing to May 2011, the licensee failed to establish, implement, and maintain adequate written procedures for the inability to drive control rods. Specifically, Abnormal Operating Instruction SO23-13-13, "Misaligned or Immovable Control Element Assembly," Revision 13, referenced "immovable" in the title, however, the procedural guidance provided to operations personnel to combat and recover from the inability to drive control rods was inadequate. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 201497724, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361; 05000362/2011003-06, "Lack of Adequate Procedures to Respond to the Inability to Drive Control Rods."

Cornerstone: Emergency Preparedness

1EP1 Exercise Evaluation (71114.01)

a. Inspection Scope

The inspectors reviewed the objectives and scenario for the 2011 biennial emergency plan exercise to determine if the exercise acceptably tested major elements of the emergency plan. The scenario simulated a catastrophic diesel generator failure and oil fire, mechanical failure of a reactor coolant pump, a leak on a reactor coolant system charging line that increases to a loss of coolant (loss of the RCS Barrier), a faulted vital electrical power bus, a loss of the fuel clad (loss of fission product barrier), a pipe failure inside containment leading to an excessive steam demand event, and a radiological release (loss of the Containment Barrier) to the environment via failures of inboard and outboard valves in the containment main purge line, to demonstrate the licensee personnel's capability to implement their emergency plan.

The inspectors evaluated exercise performance by focusing on the risk-significant activities of event classification, offsite notification, recognition of offsite dose consequences, and development of protective action recommendations, in the Simulator Control Room and the following dedicated emergency response facilities:

- Technical Support Center
- Operations Support Center
- Emergency Operations Facility

The inspectors also assessed recognition of, and response to, abnormal and emergency plant conditions, the transfer of decision making authority and emergency function responsibilities between facilities, onsite and offsite communications, protection of emergency workers, emergency repair evaluation and capability, and the overall implementation of the emergency plan to protect public health and safety and the environment. The inspectors reviewed the current revision of the facility emergency plan, emergency plan implementing procedures associated with operation of the licensee's emergency response facilities, procedures for the performance of associated emergency functions, and other documents as listed in the attachment to this report.

The inspectors compared the observed exercise performance with the requirements in the facility emergency plan, 10 CFR 50.47(b), 10 CFR Part 50, Appendix E, and with the guidance in the emergency plan implementing procedures and other federal guidance.

The inspectors attended post-exercise critiques in each emergency response facility to evaluate the initial licensee self-assessment of exercise performance. The inspectors also attended a subsequent formal presentation of critique items to plant management. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one sample as defined in Inspection Procedure 71114.01-05.

b. Findings

No findings were identified.

1EP7 Exercise Evaluation (71114.07)

Emergency Preparedness Component, of the Force-on-Force Exercise Evaluation

a. Inspection Scope

The inspectors observed licensee performance during the site emergency preparedness exercise/drill in the Control Room (Shift Manager's Office). The inspectors observed communications, event classification, and event notification activities by the simulated shift manager. The inspectors reviewed the emergency preparedness-related corrective actions from a previous inspection conducted by the NRC's Office of Nuclear Security and Incident Response to determine whether they had been completed and adequately addressed the cause of the previously-identified weakness. The inspectors also observed portions of the post-exercise/drill critique to determine whether their observations were also identified by the licensee's evaluators. The inspectors verified that minor issues identified during this inspection were entered into the licensee's corrective action program.

These activities constitute the completion of one sample as defined in Inspection Procedure 71114.07-05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS04 Occupational Dose Assessment (71124.04)

a. Inspection Scope

This area was inspected to: (1) determine the accuracy and operability of personal monitoring equipment; (2) determine the accuracy and effectiveness of the licensee's methods for determining total effective dose equivalent; and (3) ensure occupational dose is appropriately monitored. The inspectors 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 inspectors interviewed licensee personnel, performed walkdowns of various portions of the plant, and reviewed the following items:

- External dosimetry accreditation, storage, issue, use, and processing of active and passive dosimeters
- The technical competency and adequacy of the licensee's internal dosimetry program
- Adequacy of the dosimetry program for special dosimetry situations such as declared pregnant workers, multiple dosimetry placement, and neutron dose assessment
- Audits, self-assessments, and corrective action documents related to dose assessment since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.04-05.

b. Findings

No findings were identified.

2RS05 Radiation Monitoring Instrumentation (71124.05)

a. Inspection Scope

This area was inspected to verify the licensee is assuring the accuracy and operability of radiation monitoring instruments that are used to: (1) monitor areas, materials, and workers to ensure a radiologically safe work environment; and (2) detect and quantify radioactive process streams and effluent releases. The inspectors 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 inspectors interviewed licensee personnel, performed walkdowns of various portions of the plant, and reviewed the following items:

- Selected plant configurations and alignments of process, postaccident, and effluent monitors with descriptions in the UFSAR and the offsite dose calculation manual
- Select instrumentation, including effluent monitoring instrument, portable survey instruments, area radiation monitors, continuous air monitors, personnel contamination monitors, portal monitors, and small article monitors to examine their configurations and source checks
- Calibration and testing of process and effluent monitors, laboratory instrumentation, whole body counters, postaccident monitoring instrumentation, portal monitors, personnel contamination monitors, small article monitors, portable survey instruments, area radiation monitors, electronic dosimetry, air samplers, continuous air monitors
- Audits, self-assessments, and corrective action documents related to radiation monitoring instrumentation since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.05-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the First Quarter 2011 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Emergency ac Power System (MS06)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - emergency ac power system performance indicator for Units 2 and 3 for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports, and NRC integrated inspection reports for the period of March 25, 2010, through March 24, 2011, to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two mitigating systems performance index - emergency ac power system samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - High Pressure Injection Systems (MS07)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - high pressure injection systems performance indicator for Units 2 and 3 for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports, and NRC integrated inspection reports for the period of March 25, 2010, through March 24, 2011, to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two mitigating systems performance index - high pressure injection system samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.4 Mitigating Systems Performance Index - Heat Removal System (MS08)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - heat removal system performance indicator for Units 2 and 3 for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, mitigating systems performance index derivation reports, and NRC integrated inspection reports for the period of March 25, 2010, through March 24, 2011, to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two mitigating systems performance index - heat removal system samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.5 Drill/Exercise Performance (EP01)

a. Inspection Scope

The inspectors sampled licensee submittals for the drill/exercise performance, performance indicator for the period July 2010 through March 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revisions 5 and 6. The inspectors reviewed the licensee's records associated with the performance indicator to verify that the licensee accurately reported the indicator in accordance with relevant procedures and the NEI guidance. Specifically, the inspectors reviewed licensee records and processes including procedural guidance on assessing opportunities for the performance indicator; assessments of performance indicator opportunities during designated Control Room Simulator training sessions and performance during the 2011 biennial exercise, and other drills. The specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the drill/exercise performance sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.6 Emergency Response Organization Drill Participation (EP02)

a. Inspection Scope

The inspectors sampled licensee submittals for the emergency response organization drill participation performance indicator for the period July 2010 through March 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revisions 5 and 6. The inspectors reviewed the licensee's records associated with the performance indicator to verify that the licensee accurately reported the indicator in accordance with relevant procedures and the NEI guidance. Specifically, the inspectors reviewed licensee records and processes including procedural guidance on assessing opportunities for the performance indicator, performance during drills, and revisions of the roster of personnel assigned to key emergency response organization positions. The specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the emergency response organization drill participation sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.7 Alert and Notification System (EP03)

a. Inspection Scope

The inspectors sampled licensee submittals for the alert and notification system performance indicator for the period July 2010 through March 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revisions 5 and 6. The inspectors reviewed the licensee's records associated with the performance indicator to verify that the licensee accurately reported the indicator in accordance with relevant procedures and the NEI guidance. Specifically, the inspectors reviewed licensee records and processes including procedural guidance on assessing opportunities for the performance indicator and results of periodic alert notification system operability tests. The specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the alert and notification system sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee's corrective action program because of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. <u>Findings</u>

No findings were identified.

.2 <u>Daily Corrective Action Program Reviews</u>

a. <u>Inspection Scope</u>

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. <u>Findings</u>

No findings were identified.

.3 <u>Selected Issue Follow-up Inspection</u>

a. <u>Inspection Scope</u>

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action item documenting the issue listed below. 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.

- June 9, 2011, Unit 2, heater drain pump wrong unit human performance event as described in Nuclear Notification NN 201482168
- March 23 through May 18, 2011, Units 2 and 3, reviewed human performance events for proper identification of clock reset per Procedure SO123-XV-HU-1, "Human Performance Program," Revision 12

These activities constitute completion of two in-depth problem identification and resolution samples as defined in Inspection Procedure 71152-05.

b. Findings

No findings were identified.

4OA3 Event Follow-up (71153)

.1 Event Follow Up

a. Inspection Scope

The inspectors reviewed the below listed event for plant status and mitigating actions to: (1) provide input in determining the appropriate agency response in accordance with Management Directive 8.3, "NRC Incident Investigation Program"; (2) evaluate performance of mitigating systems and licensee actions; and (3) confirm that the licensee properly classified the event in accordance with emergency action level procedures and made timely notifications to NRC and state/governments, as required.

- May 13, 2011, Unit 2, K7 relay failure on emergency diesel generator 2G003

Documents reviewed by the inspectors are listed in the attachment.

These activities constitute completion of one inspection sample as defined in Inspection Procedure 71153-05.

b. Findings

No findings were identified.

.2 Event Report Review

a. Inspection Scope

The inspectors reviewed the four below listed Licensee Event Reports and related documents to assess: (1) the accuracy of the Licensee Event Report: (2) the appropriateness of corrective actions; (3) violations of requirements; and (4) generic issues.

b. Observations and Findings

1. (Closed) Licensee Event Report 05000361/2010-001-00, "Broken Manual Valve Prevents Timely Condensate Storage Tank Isolation"

On January 26, 2010, the handwheel actuator for manual isolation valve 2HV-5715 failed during the biannual Inservice Test. This valve is closed within 90 minutes of a seismic event to maintain condensate storage tank T-120 inventory in compliance with decay heat removal license commitments. The licensee identified the failure was due to a lack of lubrication and corrosion of the handwheel stem. In addition, it was identified that the surveillance test frequency had been changed in 2004 from quarterly to biennial without implementing periodic lubrication of the stem. The timely closure of the valve in

90 minutes was questioned by the inspectors during baseline inspection activities in May 2010. The licensee determined the valve closure could require as long as 140 minutes, which rendered the valve non-functional in the as-found condition. These baseline inspection activities resulted in the issuance of a finding and associated noncited violation of Technical Specification 3.7.6 for failure to perform preventative maintenance, including lubricating the valve actuator's components necessary to manually close valve 2HV-5715.

This issue was previously reviewed by the inspectors, and results of the review are documented in Section 4OA2.5c of NRC Inspection Report 05000361; 05000362/2010006. A Green noncited violation was identified and documented as NCV 05000361; 05000362/2010006-03, "Lack of Preventive Maintenance Results in Valve Failure and Inoperable Condensate Storage Tank."

The inspectors reviewed the licensee's submittal and determined that the report adequately documented the summary of the event including the potential safety consequences and corrective actions required to address the failure of the manual valve handwheel. No additional findings were identified during the review of this event as documented in the licensee event report. This licensee event report is closed.

2. <u>(Closed) Licensee Event Report 05000361; 05000362/2010-003-00, "Typographical Error Results in Conflicting TS Actions and TS Violation"</u>

On July 15, 2010, the licensee identified a typographical error associated with limiting conditions for operation and required actions in Technical Specification 3.6.3, "Containment Isolation Valves." The licensee reportability assessment found that this typographical error resulted in a conflict with the application of the required actions when one or more containment isolation valves are discovered to be inoperable. The expected action is that operations personnel should have ensured that one of the containment isolation valves was isolated in 4 hours. Contrary to this requirement, when a containment isolation valve was deemed inoperable, it was the licensee's practice not to comply with the action depending on the type of containment isolation valve. As a result, during the past three years, the licensee determined that control room operators had not been applying the required actions associated with Technical Specification 3.6.3, "Containment Isolation Valves."

The condition was caused by typographical error during the initial implementation of the licensee improved technical specifications in August 1996. As an interim corrective action, the licensee issued a priority reading to instruct the control room operators to enter the required 4 hour action whenever conditions required it. As part of the scheduled technical specification upgrade, the licensee plans to correct the Technical Specification 3.6.3 wording. The failure to meet technical specifications is being treated as a minor violation because this failure to implement the action has no impact to safety equipment and caused no safety consequences. This failure to comply with technical specification requirement constitutes a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. This licensee event report is closed.

3. <u>(Closed) Licensee Event Report 05000361; 05000362/2010-004-00, "EDG Ventilation Fan Nose Cone Corrosion Results in Fan Damage"</u>

   This issue was reviewed by the inspectors, and results of the review are documented in Section 1R15 of NRC Inspection Report 05000361; 05000362/2010004. A Green noncited violation was identified and documented as NCV 05000361; 05000362/2010004-03, "Failure to Promptly Identify and Correct a Condition Adverse to Quality Associated with Safety-related Emergency Ventilation Fans." No additional findings were identified during the review of this event as documented in the licensee event report. This licensee event report is closed.

4. <u>(Closed) Licensee Event Report 05000362/2011-002-00, "As-Found Condition of LOVS Relays Not Within Technical Specification Limits"</u>

   On March 1, 2011, loss of voltage signal relay 3A0615-127F1 associated with the safety-related 4.16kV bus failed surveillance testing since setpoints were found outside the allowable technical specification voltage range. Relay 3A0615-127F1 was replaced and returned to service. The failure was initially attributed to voltage drift based on information at the time. On April 7, 2011, the licensee identified through subsequent evaluation that the failure was due to improper calibration prior to relay installation in the plant on January 28, 2011. Based on the identification of the improper calibration, the licensee retested all loss of voltage signal relays in Units 2 and 3. The relays associated with Unit 2 safety-related electrical busses were found acceptable, and one additional relay (loss of voltage signal relay 3A0615-127F2) was found with setpoints outside the allowable technical specification voltage range on April 8, 2011. Based on the investigation, the licensee determined that relay 3A0615-127F2 had previously been installed in the plant during Mode 5 conditions on January 28, 2011, and improperly calibrated during the surveillance test performed on February 1, 2011. Relay 3A0615-127F2 was properly calibrated, retested satisfactorily, and returned to service. The licensee's evaluation of the condition determined that an undervoltage condition would have been detected within the design margins ensuring the safety function was performed by the mis-calibrated relays.

   The performance deficiency is more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. The inspectors evaluated the significance of this finding using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets. The inspectors determined the finding to have very low safety significance (Green) because the finding: (1) was not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This licensee-identified finding involved a violation of Technical Specification 3.3.7, "Diesel Generator – Undervoltage Start." The enforcement aspects of the violation are discussed in Section 4OA7. This licensee event report is closed.

4OA5 Other Activities

.1 (Open) NRC Temporary Instruction 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"

As documented in Section 1R04 of Inspection Report 05000361; 05000362/2010002, the inspectors performed a complete system alignment inspection of the Unit 2 high pressure safety injection. In addition, in Section 1R04 of Inspection Report 05000361; 05000362/2011002, the inspectors performed a complete system alignment inspection of the Unit 3 containment spray system. These inspections count towards the completion of Temporary Instruction TI 2515/177, which will be closed in a later inspection report.

a. Inspection Scope

Between January 22, 2010, and March 24, 2010, the inspectors performed a complete system alignment inspection of the Unit 2 safety injection system. Also on March 2, 2011, the inspectors performed a complete system alignment inspection of the Unit 3 containment spray system. These inspections were in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdowns was consistent with the items identified during the inspector's independent walkdown (TI 2515/177, Section 04.02.c.3).

In addition, the inspectors verified that the licensee had isometric drawings that describe the system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings:

- High point vents were identified
- High points that do not have vents were acceptably recognizable
- Other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation
- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified
- All pipes and fittings were clearly shown
- The drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution

The inspectors verified that piping and instrumentation diagrams accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the piping and instrumentation diagrams were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.b).

This inspection effort counts towards the completion of TI 2515/177, which will be closed in a later inspection report.

b. Findings

No findings were identified.

.2 (Closed) NRC Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

a. Inspection Scope

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included: (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

b. Findings

Inspection Report 05000361; 05000362/2011009 (ML11133A113) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues. A finding associated with this inspection is documented in Section 1R01.

.3 (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"

On May 20, 2011, the inspectors completed a review of the licensee's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine: (1) whether the SAMGs were available and updated; (2) whether the licensee had procedures and processes in place to control and update its SAMGs; (3) the nature and extent of the licensee's training of personnel on the use of SAMGs; and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for San Onofre Nuclear Generating Station were provided as Enclosure 11 to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 26, 2011 (ML111470264).

.4 <u>(Closed) Violation 05000361/2009004-02, "Failure to Assess and Manage Risk for Maintenance that Could Impact Offsite Power Components"</u>

The inspectors reviewed the licensee's reply to Notice of Violation; EA-09-270, dated December 9, 2009, and Root Cause Evaluation RCE 200596804, "NRC Cited Violation for Repeat Finding of Inadequate Qualitative Risk Management per Maintenance Rule Risk Management Program," to assess the licensee's effectiveness and progress in the corrective actions that were developed as a result of the violation. The inspectors determined that the corrective actions were adequate to address the root and contributing causes of the violation, and to prevent recurrence. Therefore, this violation is closed.

4OA6 Meetings

<u>Exit Meeting Summary</u>

On April 15, 2011, the inspectors presented the results of onsite inspection of the 2011 biennial emergency preparedness exercise to Mr. P. Dietrich, Senior Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On May 25, 2011, the inspectors presented the plant modifications inspection results to Mr. D. Bauder, Vice President and Station Manager, and other members of the licensee staff. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On June 9, 2011, the inspectors presented the results of the radiation safety inspections to Mr. G. Kline, Senior Director of Engineering, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On July 1, 2011, the inspectors presented the inspection results to Mr. D. Bauder, Vice President and Station Manager, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section 2.3.2 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as noncited violations.

.1 Technical Specification 3.3.7, “Diesel Generator – Undervoltage Start,” requires that four channels of loss of voltage function and four channels of degraded voltage function auto-initiation instrumentation per diesel generator be operable. Contrary to the above, from January 28 to April 8, 2011, Unit 3 was operated with one or two train B 3A0615 loss of voltage signal channels inoperable for a period of time longer than allowed by Technical Specification 3.3.7. This issue was entered into the licensee’s corrective action program as Nuclear Notifications NNs 201410580 and 201412748.

.2 Criterion V of 10 CFR Part 50, Appendix B, requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. Maintenance Procedure SO123-I-1.3, “Work Activity Guidelines,” Revision 25, Step 6.20.1, prohibited making pen and ink changes to work orders that implemented EQ requirements. Changes to these types of work orders required amending. Contrary to the above, on May 7, 2010, the instrumentation and controls work group supervisor made pen and ink changes to the instructions contained on Work Order WO 800501116 which changed the intent of the work order from installing the new transmitter in the EQ configuration shown on Drawing M-39625 to installing it in the EQ configuration shown in Drawing M-39619. This change involved a failure to perform work affecting quality in accordance with work instructions. The finding is greater than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern. The finding is of very low safety significance because it is a design or qualification deficiency confirmed not to have resulted in a loss of operability or functionality. The issue was entered into the licensee’s corrective action program as Nuclear Notification NN 201198467.

<u>SUPPLEMENTAL INFORMATION</u>
KEY POINTS OF CONTACT

<u>Licensee Personnel</u>

T. Adler, Manager, Maintenance/Systems Engineering
J. Allen, Supervisor, Nuclear Training
B. Arbour, Manager, Operations Training
J. Armas, Supervisor, Maintenance Engineering Fluid Process
D. Axline, Project Manager, Nuclear Regulatory Affairs
D. Bauder, Vice President and Station Manager
K. Bergquist, I & C Technician
G. Bhashyan, Technical Specialist
C. Cates, Manager, Recovery
B. Corbett, Director, Performance Improvement
B. Culverhouse, Manager, Offsite Emergency Preparedness
J. Davis, Manager, Plant Operations
T. Dibbins, Technician, Instrumentation and Control Maintenance
D. Dick, Supervisor, Chemistry
R. Elsasser, Manger, Training
W. Fargo, Senior Nuclear Engineer
G. Fausett, ALARA Coordinator, Health Physics
G. Ferrigno, Supervisor, Health Physics
W. Frick, Manager, Nuclear Safety Culture
K. Gallion, Manager, Onsite Emergency Preparedness
S. Genschaw, Manager, Maintenance & Construction Services
S. Gianell, Supervisor, Onsite Emergency Preparedness
L. Green, Technical Specialist, Employee Concerns
R. Hilton, Technician, Health Physics
L. Hinostroza, Foreman, Instrumentation and Control Maintenance
P. Imlay, I & C Technician
D. Inouye, BACCP Engineer Program Owner
G. Johnson, Jr., Senior Nuclear Engineer, Maintenance/Systems Engineering
K. Johnson, Manager, Design Engineering
G. Kline, Senior Director, Engineering and Technical Services
T. Knippelberg, I & C Technician
M. Lewis, Manager, Health Physics
J. Madigan, Director, Site Recovery
A. Mahindrakar, ISI Manager, Maintenance Engineering
A. Martinez, Manager, Corrective Action Program
M. McBrearty, Project Manager, Nuclear Regulatory Affairs
T. McCool, Plant Manager
R. Mejia, Supervisor, Instrumentation and Control Maintenance
L. Mosher, Manager, Communications
L. Pepple, ALARA General Foreman, Health Physics
W. Poirier, Manager, Operations
N. Quigley, Manager, Maintenance/System Engineering
T. Rak, Engineering Manager, Design Engineering

R. Richter, Engineering Supervisor, Fire Protection
M. Russell, Health Physicist, Health Physics
C. Ryan, Manager, Maintenance
S. Sewell, Technical Support, Health Physics
S. Smith, Supervisor, I & C Technician
M. Stevens, Engineer, Nuclear Regulatory Affairs
R. St. Onge, Director, Nuclear Regulatory Affairs
R. Treadway, Manager, Compliance
S. Vaughan, ALARA Manager, Health Physics
D. Yarbrough, Director, Plant Operations

NRC Personnel

R. Mathew, Acting Branch Chief, Electrical Engineering Branch, NRR
M. Runyan, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000361/2011003-01 05000362/2011003-01	NCV	Inadequate Compensatory Measures for a Design Nonconformance (Section 1R01)
05000361/2011003-02 05000362/2011003-02	NCV	Inadequate Work Instructions to Ensure Environmentally Qualified Configuration (Section 1R12)
05000361/2011003-03 05000362/2011003-03	NCV	Failure to Appropriately Assess and Manage Risk for Work in Unit 3 Intake (Section 1R13)
05000361/2011003-04 05000362/2011003-04	NCV	Failure to Provide Adequate Long-Time Over-Current Protection for the Feeder Cables for Charging Pumps 2P190 and 2P191 (Section 1R17)
05000361/2011003-05	NCV	Failure to Follow Procedure Requirements for a Failed Surveillance (Section 1R22)
05000361/2011003-06 05000362/2011003-06	NCV	Lack of Adequate Procedures to Respond to the Inability to Drive Control Rods (Section 1R22)

Closed

05000361/2010-001-00	LER	Broken Manual Valve Prevents Timely Condensate Storage Tank Isolation (Section 4OA3)
05000361/2010-003-00 05000362/2010-003-00	LER	Typographical Error Results in Conflicting TS Actions and TS Violation (Section 4OA3)
05000361/2010-004-00 05000362/2010-004-00	LER	EDG Ventilation Fan Nose Cone Corrosion Results in Fan Damage (Section 4OA3)
05000362/2011-002-00	LER	As-Found Condition of LOVS Relays Not Within Technical

		Specification Limits (Section 4OA3)
05000361/2009004-02	VIO	Failure to Assess and Manage Risk for Maintenance that Could Impact Offsite Power Components (Section 4OA5)

Discussed

2515/177	TI	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01) (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION / DATE
SO23-XX-29.1	Seasonal Readiness	2
DBD-SO23-120	Design Basis Document 220kv system	0
OM-6.9.1	Substation Inspections	April 26, 2000
SO23-6-30	Switchyard Inspection and Operation	31
SO23-13-4	Operation During Major System Disturbances	17

NUCLEAR NOTIFICATIONS

NUMBER	
200697485	201497144

Section 1R04: Equipment Alignment

PROCEDURES

NUMBER	TITLE	REVISION
SO23-3-3.60.6	Auxiliary Feedwater Pump and Valve Testing	20
SO23-3-3.16	Auxiliary Feedwater System Monthly Tests	14
SO23-2-13.1	Diesel Generator Alignments	8
SO123-0-A4	Configuration Control	17
SO123-XV-15	Maintaining Plant Status Control	2
SO23-2-4	Auxiliary Feedwater System Operation	32

NUCLEAR NOTIFICATIONS

NUMBER				
201015075	201460996	201403969	201354351	201253729
201124869	200944010	201286485	201507209	200784013

WORK ORDERS

NUMBER
800547169

DRAWINGS

NUMBER	TITLE	REVISION
40110B	Diesel Generator System (Train A)	37
40116A	Diesel Fuel Storage System	13
40160A	P&ID Auxiliary Feedwater System	43
40160B	P&ID Auxiliary Feedwater Steam Supply	24

MISCELLANEOUS

NUMBER	TITLE	REVISION
SD-SO23-780	System Description - Auxiliary Feedwater System	12
SD-SO23-750	System Description: Part III Fuel Oil System	19
Tech Spec 3.8.3/B	Diesel Fuel Oil, Lube Oil, and Starting Air	211
SD-SO23-780	Auxiliary Feedwater System	13

Section 1R05: Fire Protection

PROCEDURES

NUMBER	TITLE	REVISION
SD-SO23-590	System Description - Fire Protection System	15
SO123-FP-1	Fire Protection Order	9
SO23-XV-4.13	Control of Work and Storage Area within Protected Area	10
SO23-XIII-48	Spray/Sprinkler System Inspection	9
SO23-XIII-57	Barrier Inspections	19
SO23-13-21	Fire	23
SO123-XIII-4.10.3	Fire Department Fire Fighting Response	8

SO23-XIII-4.100	Units 2 and 3 Fire Monitoring Systems (FMS) Computer Use and Impairment Scope Identification	19
SO123-XIII-4.10.1	Fire Department Communications, Protected Area (PA) Entry and Radiologically Controlled Area Entry	7
SO123-XIII-4.10.2	Fire Department Dispatch	14
SO23-XIII-4.13	Inspection for Control of Combustibles and Transient Fire Loads	3
SO23-XV-4.13	Control of Work and Storage Areas Within the Protected Area	10
SO23-XIII-50	Fire Door Inspection	15

NUCLEAR NOTIFICATIONS

NUMBER				
201395547	201399175	201403373	201505409	201504960
201505337	201322896	201482210	201322899	

WORK ORDERS

NUMBER
060601174

DRAWINGS

NUMBER	TITLE	REVISION
2-013	Pre-Fire Plans - Unit 2 Diesel Generators and Diesel Fuel Transfer Pump Rooms A & B	7
3-045	Pre-Fire Plans - Unit 3 Diesel Generators and Diesel Fuel Transfer Pump Rooms A & B	7
3-037	Pre-Fire Plans, Unit 3 Penetration and Fuel Handling	6
2-006	Pre-Fire Plans, Unit 2 Safety Equipment Building, Elevation -15' to 8'	6
2-007	Pre-Fire Plans, Unit 2 Safety Equipment Building, Elevation -15'	5
	Pre-Fire Plans, Units 2 and 3	6
3-043	Pre-Fire Plans	5
2-009	Pre-Fire Plans: Main Steam Isolation Area	6

MISCELLANEOUS

NUMBER	TITLE	DATE
Impairment 11030075-00	2-YD-30-CCZ6 Combustible Control Zone Outside of the Unit 2 Diesel Generator Building	March 29, 2011
TCR 11030019-1	40 lbs of plastic tarp over the scaffolding in Combustible Control Zone #6	March 29, 2011

Section 1R06: Flood Protection Measures

NUCLEAR NOTIFICATIONS

NUMBER	
201400829	201388166

WORK ORDERS

NUMBER
800523888

DRAWINGS

NUMBER	TITLE	REVISION
2-006	Pre-Fire Plans, Unit 2 Safety Equipment, El. -15'6" to 8'	6
2-007	Pre-Fire Plans, Unit 2 Safety Equipment, El. -15'-6"	5

Section 1R11: Licensed Operator Requalification Program

PROCEDURES

NUMBER	TITLE	REVISION
OSM-9	Standard EOI Good Practices and Strategies	7
OSM-14	Operations Department Expectations	5

Section 1R12: Maintenance Effectiveness

PROCEDURES

NUMBER	TITLE	REVISION
SO123-I-1.3	Work Activity Guidelines	25
SO123-I-4.59	Wire/Cable Inspection	15 EC1
SO123-II-1.1.2	Surveillance Requirement, Plant Protection System, Channel Functional Test	8

SO123-I-4.61	CONAX Seal Assembly-Removal, Cleaning, Inspection, Repair, and Installation	EC 1-4

NUCLEAR NOTIFICATIONS

NUMBER				
201425098	201330175	201330137	201164842	201230901
201164841	201200498	201230987	201230852	201201604
201198467	200922182	201477774		

WORK ORDERS

NUMBER
800501116

DESIGN BASIS DOCUMENTS

NUMBER	TITLE	REVISION
DBD-SO23-TR-EQ	Environmental Qualification topical Report pages:19, 21,23, 24, and 25	9

DRAWINGS

NUMBER	TITLE	REVISION
LOOP 2PDT0979-2 Sheet 1	Loop Diagram SG E088 (SG) CH B RCS DIFF PRESS	2
LOOP 2PDT0979-2 Sheet 2	Loop Diagram SG E088 (SG) CH B RCS DIFF PRESS Bistable Outputs	0
39619 Sheet 1	EQ Configuration Detail Rosemount transmitter Model 1153GD9, 1153HD5, 1153HD6	4
39619 Sheet 2	EQ Configuration Detail Rosemount transmitter Model 1153GD9, 1153HD5, 1153HD6	2
39619 Sheet 3	EQ Configuration Detail Rosemount transmitter Model 1153HD6	2
39625	EQ Configuration Detail Rosemount transmitter Model 1153GD9, 1153HD5, 1153HD6	3

VENDOR MANUALS

NUMBER	TITLE	REVISION
IPS-725	CONAX Corporation Installation Manual for Electric Conductor Seal Assemblies with Long Body for Pipe Thread Equipment Interface	J

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
SO23-302-18-M54	Failure Analysis Report	0
SONG Unit 2 & 3 M37631	EQ Documentation Package for Rosemount Model 1153 Series D Transmitters Section C.1	7
SONG Unit 2 & 3 M37631	EQ Documentation Package for Rosemount Model 1153 Series D Transmitters	16
900500183	Non-Conformance Report (Action Request)	00
2-3114	Non-Conformance Report (Action Request)	00
CS-E03	Construction Specification for Cable Splicing, Terminations and Supports	22
FSAR §7.2.1.1.1.12	Reactor Protective System	April 2009
FSAR §7.2.2.2.12	Reactor Protective System	April 2009
	List of Transmitters with Similar EQ Configuration to 2PDT-0979-2	
	Field Verification Sheets for Unit 2 Transmitters: 2PDT09781, 09782, 09783, 09784, 09791, 09792, 09793, and 09794 with Photographs	April 1991
	Photographs of Unit 3 Transmitters: 3PDT0978-1, -2, -3, -4, 3PDT0979-1, -2, -3, and -4	2010
	Photographs of Unit 2 Transmitters	2008

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

PROCEDURES

NUMBER	TITLE	REVISION
SO123-IT-1	Infrequently Performed Tests or Evolutions	13
SO23-XX-8	Integrated Risk Management	8
SO23-XX-10	Maintenance Rule Risk Management Program Implementation	7
SO23-XX-35	Protected Equipment	3
SO23-XX-28	On-Line Work Management Process	6

NUCLEAR NOTIFICATIONS

NUMBER	
201408361	201395115

WORK ORDERS

NUMBER		
800680841	800680889	800670023

Section 1R15: Operability Evaluations

PROCEDURES

NUMBER	TITLE	REVISION
SO2-II-11.1A-2	Unit 2 ESF Train A Channel (Online) Test of Loss of Voltage (LOVS), Degraded Voltage (SDVS, DGVSS) and Sequencing Relays and Circuits	11

NUCLEAR NOTIFICATIONS

NUMBER				
201248977	201249165	201425098	201433743	201423504
201076929	201427560	201425804	200862977	201439669
201448381	201450398	201448328	201448466	201412748
201410580	201352641			

Section 1R17: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

PROCEDURES

NUMBER	TITLE	REVISION
SO23-I-2.58.1	480v ABB K-Line – Breaker Secondary and Primary Current Testing	4
SO123-I-4.59	Wire/Cable Inspection	15 EC1
SO23-3-3.60.5	Surveillance Operating Instruction Charging Pump and Valve testing	10

NUCLEAR NOTIFICATIONS

NUMBER

201443248 201513478 201540343

WORK ORDERS

NUMBER

060400439-01 011001639 060400439 070201139

DRAWINGS

NUMBER	TITLE	REVISION
30118	One Line Diagram 480V Loadcenter 2B04 (ESF) and 2B24 (ESF)	21

CALCULATIONS

NUMBER	TITLE	REVISION / DATE
E4C-050	Low Voltage Power Circuit Breaker Settings	December 5, 1977
E4C-061 Sheets1, 4, and 5 of 131	Cable Ampacity Derating Calculations	3
E4C-051	600V Power Cable Ampacity for 480V Load Center w/Maintained Spacing	15
M-DSC-235	Diesel Generator Load Verification Mechanical Equipment BHP Requirements	August 31, 1999
E4C-051.1	Class 1E 600V Power Cable ampacity for 480V Load Center feeders	1
E4C-099	SR 480V Power Circuit Breaker Settings	3

E4C-051	Non-Class 1E 600V Power Cable Ampacity for 480V Loadcenter w/Maintained Spacing	16

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
SO23-928-16	Pump Performance Curves for Charging Pumps	16
	Thermal Ceramics Product Data Information on Cerablanket	June 20, 1995
	Walkdown Record of Motor Nameplate Data for Unit 2 Charging Pumps	March 7, 1990

Section 1R19: Postmaintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
SO23-3-3.30	Inservice Valve Testing Program	20
SO23-2-17.2	Component Cooling Water System Removal/Return To Service Evolutions (Online or Outage)	12
SO23-2-8.1	Saltwater Cooling System Removal/Return To Service Evolutions (Online or Outage)	12
SO23-3-3.30.4	Main Steam to K-007 Online Valve Test	13

WORK ORDERS

NUMBER	
800432664	800697184

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
SO23-3-3.43.30	ESF Subgroup Relays K-112A, K-625A, and K-725A Semiannual Test	5
SO23-11-5.1	Excore Neutron Safety Channel Calibration	14
SO23-XIII-73	Skid Mounted Pump Test (SA2301MP1058)	2 EC 1
OSM-12	Operator Fundamentals	13
SO23-3-3	Operations Surveillance Program Requirements	16

NUCLEAR NOTIFICATIONS

NUMBER

201444814	201430861	201432333	201496401	201474804
201469661	201453658			

WORK ORDERS

NUMBER

800340111

DRAWINGS

NUMBER	TITLE	REVISION
Loop 2-JE0001-1	Loop Diagram Excore channel A Log Power	2
Loop 2-JE0002-1	Loop Diagram Excore channel A Linear Power	1

MISCELLANEOUS

NUMBER	TITLE	DATE
M3-3766	Ronan X88 Calibrator,	Calibration due date September 30, 2011
Loop 2JYK099	I/C Loop Verification Data Record	Calibration due date May 5, 2011
	CEDMCS Troubleshooting Plan	0
LER 05000301/2010-003-00	Technical Specification Required Shutdown	
	Priority 2 Reading	May 20, 2011
TB-06-17	CDRM Transitory Misstepping Due to Crud	December 12, 2006
	Operator Logs	May 8-9, 2011
Compliance Clarification 122	CEA Operability	November 3, 1988
CEOG-91-434	Relaxation of Tech Spec on Movable CEA Operability	August 9, 1991

Section 1EP1: Exercise Evaluation

PROCEDURES

NUMBER	TITLE	REVISION
SO123-VIII-0.200	Emergency Plan Drills and Exercise	12

SO123-VIII-1	Recognition and Classification of Emergencies	33
SO123-VIII-10	Emergency Coordinator Duties	29
SO123-VIII-10.1	Station Emergency Director Duties	29
SO123-VIII-10.2	Corporate Emergency Director Duties	20
SO123-VIII-10.3	Protective Action Recommendations	12
SO123-VIII-10.4	Technical Support Center Manager Duties	3
SO123-VIII-30.3	OSC Operations Coordinator Duties	6
SO123-VIII-30.4	Emergency Services Coordinator Duties	10
SO123-VIII-30.7	Emergency Notifications	13
SO123-VIII-40.1	OSC Health Physics Coordinator Duties	27
SO123-VIII-40.100	Dose Assessment	14
SO123-VIII-60.1	OSC Security Coordinator Duties	20
SO123-VIII-80	Emergency Group Leader Duties	15
SO23-12-1	Standard Post Trip Actions	24
SO23-12-3	Loss of Coolant Accident	22
SO23-12-11	EOI Supporting Attachments	9
SO23-12-9	Functional Recovery	27
SO23-3-2.28	Containment Combustible Gas Control System	16
SO23-V-5	SONGS Severe Accident Management Guidelines	3
SO123-XXI-1.11.3	Emergency Plan Training Program Description	23
SO123-XXI-TPD-HP	Health Physics Personnel Training Program Description	1
	Sequence of Events, 2007 Biennial Exercise	
	Sequence of Events, 2009 Biennial Exercise	

NUCLEAR NOTIFICATIONS

NUMBER

200519920	200609759	200663392	200704908	200751228
200950516	200950657	200950710	200952579	200967789
200971007	200974676	201018867	201057894	201061405
201120923	201135750	201222692	201238990	201256001
201313073	201318584	201319127	201382899	201386644
201397091	201402502	201410779	201417282	201419054

201419545	201421373	201421620	201421630	201421684
201421703	201421723	201421756	201422595	201422956
201423076				

Section 2RS04: Occupational Dose Assessment

PROCEDURES

NUMBER	TITLE	REVISION
SO123-VII-20	Health Physics Program	16
SO123-VII-20.6	External Occupational Exposure Monitoring	10
SO123-VII-20.6.1	Calculation of Dose from Skin Contamination	6
SO123-VII-20.7	Internal Occupational Exposure Monitoring	7
SO123-VII-20.9	Radiological Surveys	9
SO123-VII-20.9	Radiological Surveys	12
SO123-VII-20.9.4	Survey and Release of Personnel	10
SO123-VII-20.10	Radiological Work Planning and Controls	19
SO123-VII-20.14.1	Health Physics Instrumentation Program	7
SO123-VII-20.15	Radiation Protection for Unborn Children	3
SO123-VII-20.20.2	Dosimetry Performance Testing	5
SO123-GHP-1	Radiation Protection Program for Unborn Children	8

NUCLEAR NOTIFICATIONS

NUMBER				
200586800	200619747	200625730	200630851	200644258
200647686	200668103	200690938	200691444	200713282
200734647	200738544	200738754	200753725	200757164
200773301	200776301	200786897	200792729	200793832
200815894	200817886	200829762	200862011	200894911
200930254	200952722	200954014	200970295	201018334
201044941	201048788	201145187	201157960	201163788
201172706	201183020	201178812	201218602	201227205
201231265	201239161	201246691	201249572	201275964
201277312	201292790	201375227	201387057	201483798
201497586				

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
1Q10	Health Physics Division Performance Assessment Report	April 30, 2010
2Q10	Health Physics Division Performance Assessment Report	July 15, 2010
3Q10	Health Physics Division Performance Assessment Report	October 28, 2010
4Q10	Health Physics Division Performance Assessment Report	January 31, 2011

MISCELLANEOUS

NUMBER	TITLE	DATE
Survey #100305-017	Complete Report	April 14, 2010
Survey #091004-043	SONGS Radiological Survey	October 4, 2009
Survey #110426-007	SONGS Radiological Survey	April 26, 2011
HP(123)100-5	Neutron Dose Estimate Worksheet	April 27, 2011

Section 2RS05: Radiation Monitoring Instrumentation

PROCEDURES

NUMBER	TITLE	REVISION
SO123-VII-20	Health Physics Program	16
SO123-VII-20.14.1	Health Physics Instrumentation Program	7
SO123-VII-20.14.2.1	Operation of Common Portable Survey Instruments	11
SO123-VII-20.14.4.1	Operation of Common Portable Count Rate Meters	4
SO123-VII-20.14.5.6	REM 500 Neutron Survey Instrument	0
SO123-VII-20.14.6.4	Calibration of NNC Gamma-60Portal Monitor	9
SO123-VII-20.14.7.1	Calibration of Common Portable Air Samplers	5
SO23-XXV-4.19	Plant Vent Stack/Waste Gas Holdup System Loop 2/3 RE 7808G Channel Calibration	4
SO23-XXV-9.309	Plant Vent Stack/Waste Gas Holdup System Loop 2/3 RE 7808G Electronic and Transfer Isotopic Calibration	3

SO23-XXV-9.337	Setting of General Atomics Digital Radiation Monitor Alarm Setpoints and Process Flow Substitute Values	1
SO123-XXV-9.338	Alarm Setpoints Calibration: General Atomics Analog Radiation Monitors	3

NUCLEAR NOTIFICATIONS

NUMBER				
200534443	200678700	200792729	201024997	201199682
200553819	200687188	200793832	201044941	201218395
200555600	200728442	200815894	201123655	201218602
200556508	200734647	200817886	201133879	201246691
200573295	200738754	200818672	201139701	201256107
200586800	200771834	200868850	201146642	201258504
200587331	200772438	200952722	201146721	201315012
200619747	200773301	200954014	201156968	201327799
200623200	200782128	200976227	201170450	201372347

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
200819630	SONGS Radiation Monitoring System Performance Trending Program	2010
	Benchmark Report of Instrumentation Program: Columbia vs. SONGS	May 18, 2010
2Q10	Health Physics Division Performance Assessment Report	July 15, 2010
3Q10	Health Physics Division Performance Assessment Report	October 28, 2010
4Q10	Health Physics Division Performance Assessment Report	January 31, 2011

RADIATION PROTECTION INSTRUMENTATION CALIBRATIONS

IDENTIFICATION MODEL NO.	INSTRUMENT TYPE	CALIBRATION DATE
3RE7817	Blowdown Process System Neutralization Sump Discharge Radiation Monitor	July 29, 2010
3RE7821	Turbine Area Sump Radiation Monitor	February 7, 2011
2/3RE7808G	Plant Vent Stack Particulate, Iodine, Noble Gas Radiation Monitor	October 16, 2009
2RE7865A/B/C	Containment Purge and Plant Vent Stack Radiation Monitor – Low/Mid/High Range	October 1, 2010
635	Small Article Monitor II	July 1, 2010

4A	Gamma-60 Portal Monitor	December 12, 2010
8	Gamma-60 Portal Monitor	December 20, 2010
113675	Ludlum 177-1 Count Rate Meter	May 4, 2011
6112	AMP-100 Portable Underwater Survey Meter	February 16, 2011
1266	AMS-4 Continuous Airborne Monitor	May 13, 2011
1387	AMS-4 Continuous Airborne Monitor	May 20, 2011
0903	ASPI Neutron Rem Ball and Survey Meter	June 6, 2011

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
	Offsite Dose Calculation Manual	4
	Radioactive Effluent Release Report	2010
SO123-III-5.8	Units 2/3 Liquid Monitor Setpoint Data Transmittal	December 6, 2010
SO123-III-5.9	Units 2/3 Gaseous Monitor Setpoint Data Transmittal	December 6, 2010
	RE7865 Containment Purge Isokinetic Flow Schemes	June 9, 2011

Section 4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION
SO123-VIII-0.301	Emergency Telecommunications Testing	14
SO123-VIII-0.302	Onsite Emergency Siren Test	5
SO123-VIII-0.401	Emergency Preparedness Performance Indicators	2
SO123-VIII-1	Recognition and Classification of Emergencies	33, 34
SO123-VIII-10.3	Protective Action Recommendations	12
SO123-VIII-30.7	Emergency Notifications	13
SO123-XVIII-10	Community Alert Siren System, System Description and Operation Guide	13-1
SO123-XVIII-10.1	Community Alert Siren System, Biweekly Silent Test	9-1
SO123-XVIII-10.3	Community Alert Siren System, Quarterly Growl Test	13-1
SO123-XVIII-10.4	Community Alert Siren System, Response to a Report of an Inadvertent Siren Activation	6-1

SO123-XVIII-10.5	Community Alert Siren System, Annual Activation Test Procedures	10-1
SO123-XVIII-10.6	Community Alert Siren System, Inspection and Maintenance	8
SO23-XV-24	Quarterly NRC Performance Indicator (PI) Process	9
SO23-3-3.60.2	Auxiliary Feedwater Pump and Valve Testing	20

NUCLEAR NOTIFICATIONS

NUMBER		
201065100	201330175	201344602

WORK ORDERS

NUMBER
800432671

MISCELLANEOUS

NUMBER	TITLE	REVISION
	San Onfore Nuclear Generating Station Emergency Plan	
	Lesson Plan Encode Number EP4100, “Dose Assessment”	5
	Lesson Plan Encode Number 710SED, “Technical Support Center Training”	1-4
	Lesson Plan Encode Number 710EGL, “Operations Support Center”	1
	Qualification Walk Through (QWT) Encode Number 7L1QWT, TSC Health Physics Leader	2
	Qualification Walk Through (QWT) Encode Number 7V2QWT, OSC Health Physics Coordinator	2
	Lesson Plan Encode Number RQ1013, “NEI 99-01 Emergency Classification Training”	1-4
	Lesson Plan Encode Number EPT-020, “OSC HP Staff Positions and Duties”	2-1

Section 4OA2: Identification and Resolution of Problems

NUCLEAR NOTIFICATIONS

NUMBER				
201352641	201412077	201445911	201393301	201400711
201350486	201395115			

MISCELLANEOUS

NUMBER	TITLE	DATE
	Station Stand Up	May 31, 2011
	Plant Daily Brief	June 1-3, 2011
	Station Stand Up	June 7, 2011
	Memo to Site Personnel from Director M&CS	May 27, 2011
	Operational Alignment	June 6, 2011

Section 4OA3: Event Follow-Up

PROCEDURES

NUMBER	TITLE	REVISION
SO123-II-11.152	Circuit Device Tests and Overall Functional Test	17

NUCLEAR NOTIFICATIONS

NUMBER				
200765235	201457104	201412748	201410580	201352641

DRAWINGS

NUMBER	TITLE	REVISION
30342, Sheet 6	Diesel Generator 2G003 Control System	11

Section 4OA5: Other Activities

NUCLEAR NOTIFICATIONS

NUMBER		
200596804	200556120	200559128