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

July 30, 2010

EA-10-125

Mr. Ross T. Ridenoure
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 PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION REPORT 05000361/2010006; 05000362/2010006 AND NOTICE OF VIOLATION

Dear Mr. Ridenoure:

On April 23, 2010, the U. S. Nuclear Regulatory Commission (NRC) completed the onsite portion of a team inspection at your San Onofre Nuclear Generation Station. Additionally, the inspectors performed in-office inspections through June 17, 2010. The enclosed report documents the inspection findings discussed with you and members of your staff during an exit briefing on June 17, 2010.

The inspection examined activities conducted under your license as they relate to identification and resolution of problems, safety and compliance with the Commission's rules and regulations and with the conditions of your operating license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. The inspectors also interviewed a representative sample of personnel regarding the condition of your safety conscious work environment.

When compared with the findings from the previous inspection conducted in September 2008, the findings from this inspection indicate that the corrective action program effectiveness has declined. As previously discussed in the past 5 NRC assessment letters your staff's ability to thoroughly evaluate problems such that the resolutions effectively address the causes and extent of conditions is of concern. Your efforts to reverse the trend of substantive crosscutting issues in both the human performance and problem identification and resolution areas have not shown to be effective.

The inspection identified a number of issues that your staff had previous opportunities to identify. The inspectors noted that even after issues were discussed with your staff thorough evaluations were not consistently completed. We noted examples where your staff's evaluations for deficient components failed to fully address component safety functions for all applicable design basis accident scenarios.

The inspectors reviewed the status of site corrective actions related to the areas of human performance and problem identification and resolution described in your letters to the NRC dated April 21, October 29, and October 30, 2009.

The inspectors noted examples where due dates were exceeded and different actions were performed from those specified in the plan. As a result, the NRC identified a finding related to your failures to meet the actions discussed in the above referenced letters. During the next public meeting, that is currently being scheduled, you should address the status of your site corrective actions and additional controls put in place to effectively monitor their execution. You should also plan to address the causes for the inability to reverse the poor human performance and problem identification and resolution trends.

This report documents ten NRC identified noncited violations, one NRC identified cited violation, one self-revealing violation, and one finding, all of very low safety significance (Green). Additionally, one licensee-identified violation is also discussed in this report. Because of the very low safety significance of the violations and because they were entered into your corrective action program, the NRC is treating these violations as noncited violations consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest these 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 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, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the San Onofre Nuclear Generating Station.

The NRC-identified violation is cited in the enclosed Notice of Violation (Enclosure 1). The violation involved the failure to revise and maintain in effect adequate procedures following plant modifications. Although determined to be of very low safety significance (Green), this violation is being cited in the Notice of Violation because not all of the criteria specified in Section VI.A.1 of the NRC Enforcement Policy for a noncited violation were satisfied. Specifically, San Onofre Nuclear Generating Station failed to restore compliance within a reasonable time after previously-identified noncited violations were identified in NRC Inspection Report 05000361; 05000362/2009003-02 and 05000361; 05000362/2009009-02. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

If you disagree with the crosscutting 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 San Onofre Nuclear Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web-site at www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

Dale A. Powers

for Michael C. Hay, Chief
Technical Support Branch
Division of Reactor Safety

Dockets: 50-361; 50-362
Licenses: NPF-10; NPF-15

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 05000361/2010006; 05000362/2010006
   w/Attachment: Supplemental Information

cc (w/Enclosures):
Chairman, Board of Supervisors
County of San Diego
1600 Pacific Highway, Room 335
San Diego, CA 92101

Gary L. Nolff
Assistant Director-Resources
City of Riverside
3900 Main Street
Riverside, CA 92522

Mark L. Parsons
Deputy City Attorney
City of Riverside
3900 Main Street
Riverside, CA 92522

Gary H. Yamamoto, P.E., Chief
Division of Drinking Water and
 Environmental Management
1616 Capitol Avenue, MS 7400
P.O. Box 997377
Sacramento, CA 95899-7377

Michael L. DeMarco
San Onofre Liaison
San Diego Gas & Electric Company
8315 Century Park Ct. CP21C
San Diego, CA 92123-1548

Director, Radiological Health Branch
State Department of Health Services
P.O. Box 997414 (MS 7610)
Sacramento, CA 95899-7414

The Mayor of the City of San Clemente
100 Avenida Presidio
San Clemente, CA 92672

James D. Boyd, Commissioner
California Energy Commission
1516 Ninth Street (MS 34)
Sacramento, CA 95814

Douglas K. Porter, Esquire
Southern California Edison Company
2244 Walnut Grove Avenue
Rosemead, CA 91770

Albert R. Hochevar
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

Steve Hsu
Department of Health Services
Radiologic Health Branch
MS 7610, P.O. Box 997414
Sacramento, CA 95899-7414

R. St. Onge
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

Chief, Technological Hazards Branch
FEMA Region IX
1111 Broadway, Suite 1200
Oakland, CA 94607-4052

Institute of Nuclear Power Operations (INPO)
Records Center
700 Galleria Parkway SE, Suite 10
Atlanta, GA 30339

NOTICE OF VIOLATION

Southern California Edison Company
San Onofre Nuclear Generating Station

Docket No: 50-361; 50-362
License No: NPF-10; NPF-15
EA-10-125

During an NRC inspection, conducted from April 5 to April 23, 2010, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Technical Specification 5.5.1.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," Appendix A, recommends procedures for the operation of certain plant systems.
>
> Contrary to the above, prior to April 23, 2010, Southern California Edison Company failed to maintain written procedures as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Specifically, the licensee failed to ensure that following modifications made to the instrument air system the affected system procedures where either suspended, put on administrative hold, or otherwise restricted from use until the required changes were implemented. As a result, several procedures with known technical deficiencies were inappropriately available for use following plant modifications.

This violation is associated with a Green Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Southern California Edison Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector San Onofre Nuclear Generating Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to Notice of Violation EA-09-270," and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time. If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the

NRC website at www.nrc.gov/reading-rm/pdr.html or www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the basis for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information).

Dated this 30th day of July 2010.

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-361; 50-362
License:	NPF-10; NPF-15
Report:	05000361/2010006; 05000362/2010006
Licensee:	Southern California Edison Co.
Facility:	San Onofre Nuclear Generating Station
Location:	5000 So. Pacific Coast Highway San Clemente, California
Dates:	April 5 through June 17, 2010
Team Leader:	M. Vasquez, Senior Reactor Inspector, Technical Support Branch, DRS
Inspectors:	C. Long, Senior Resident Inspector R. Smith, Senior Resident Inspector S. Walker, Senior Reactor Inspector E. Ruesch, Resident Inspector S. Matharu, Senior Electrical Engineer G. Tutak, Project Engineer
Accompanied By:	G. Wilson, Chief, Electrical Engineering Branch S. Marquez, Nuclear Safety Professional Development Program
Approved By:	Michael C. Hay, Chief Technical Support Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR05000361/2010006; 05000362/2010006; October 1, 2008, through April 23, 2010: San Onofre Nuclear Generating Station "Biennial Baseline Inspection of the Identification and Resolution of Problems."

The report covers a 2-week period of onsite inspection by two senior resident inspectors, a senior electrical engineer, a senior reactor inspector, a reactor inspector, and a project engineer. Following the onsite inspection additional in-office reviews were performed through June 17, 2010. The findings from this inspection include ten Green NRC identified noncited violations, one Green self revealing violation; one Green cited violation, and one Green finding. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management 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.

Identification and Resolution of Problems

The inspectors reviewed approximately 300 condition reports, work orders, engineering evaluations, root and apparent cause evaluations, and other supporting documentation to determine if problems were being properly identified, characterized, and entered into the corrective action program for evaluation and resolution. The inspectors reviewed a sample of system health reports, self-assessments, trending reports and metrics, and various other documents related to the corrective action program.

When compared with the findings from the previous inspection conducted in September 2008, the findings from this inspection indicate that the corrective action program effectiveness has declined. As previously discussed in the past five NRC assessment letters, the licensee's ability to thoroughly evaluate problems such that the resolutions effectively address the causes and extent of conditions is of concern. The licensee's efforts to reverse the trend of substantive crosscutting issues in both the human performance and problem identification and resolution areas have not shown to be effective.

Additionally, the inspection identified a number of issues that the licensee's staff had previous opportunities to identify. The inspectors noted that even after issues were discussed with the licensees' staff, thorough evaluations were not consistently completed. We noted examples were the evaluations for deficient components failed to fully address the component safety functions for all applicable design basis accident scenarios.

The inspectors determined that the licensee adequately evaluated industry operating experience for relevance to the facility, and entered applicable items in the corrective action program. The inspectors noted that operating experience was considered in cause evaluations. The inspectors noted that following the review of operating experience the licensee failed to consistently incorporate the knowledge into procedural guidance and design calculations.

In February 2010, the inspectors found that several work groups at San Onofre did not feel free to raise safety concerns

without fear of retaliation. This was documented in NRC Inspection Report 050000361; 05000362/2009009 dated March 2, 2010, and in the NRC's Chilling Effect Letter dated March 2, 2010.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a noncited violation of Technical Specification 5.5.1.1.a involving the failure of control room operators to follow San Onofre Procedure SO123-0-A1, "Conduct of Operations." These included failures to: implement alarm response procedure place-keeping, announce alarms to the control room supervisor, stop conversations when an alarm annunciated and cleared, perform three-way communication during pre-job briefing, review the summarize, anticipate, foresee, evaluate and review questions during a pre-job brief, review the prerequisites of a procedure prior to use, and remain cognitive of the re-activity change evolution by a control room supervisor. This issue was entered into the licensee's corrective action program as Nuclear Notification 200871332, and operations management immediately began actions to institute a recovery plan to improve operator performance.

  The finding was more than minor because it was associated with the Initiating Events Cornerstone attribute of human performance, and it affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and that challenge critical safety functions during shutdown, as well as during power operations. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that the transient initiator did not contribute to both the likelihood of a reactor trip and to the likelihood that mitigation equipment or functions would not be available. As a result, the issue was of very low safety significance (Green). The finding has a crosscutting aspect in the area of human performance associated with the work practices because the licensee did not ensure supervisory and management oversight of work activities. [H.4(c)](Section 40A2.5e)

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.5.1.1.a involving the failure to maintain adequate instructions in San Onofre Procedure SO23-3-2.4, "RCS Purification and De-borating Ion Exchanger Operation," Revision 21 to control borating of ion exchangers. The failure to maintain an adequate procedure resulted in an unplanned power reduction by control room operators. This issue was entered into the licensee's corrective action program as Nuclear Notification 200721702. Immediate corrective actions included revising the procedure and operator crew training.

The finding was more than minor because it was associated with the Initiating Events Cornerstone attribute of human performance, and it affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and that challenge critical safety functions during shutdown, as well as during power operations. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that the transient initiator did not contribute to both the likelihood of a reactor trip and to the likelihood that mitigation equipment or functions would not be available. As a result, the issue was of very low safety significance (Green). The finding has a crosscutting aspect in the area of human performance associated with the work practices because licensee supervisory personnel did not ensure activities associated with re-activity control were performed in a controlled manner such that nuclear safety was assured.
[H.4(c)](Section 40A2.5f)

- Green. The inspectors identified a noncitied violation of Technical Specification 5.5.1.1.a involving the failure to follow procedural guidance of SO123-XX-11, "Switchyard Work Performance." Specifically, the inspectors identified temporary equipment stored in the switchyard that was not tethered or otherwise secured in accordance with the procedure. The licensee entered a notification in its corrective action program as Nuclear Notification 200870138, and removed or secured the items.

  This finding is more than minor because it impacts the protection against the external factors attribute of the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and power operations. Using the Inspection Manual Chapter 0609 "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This finding also has a human performance crosscutting aspect associated with the work control component in that personnel failed to appropriately plan work activities involving job site conditions which may impact plant structures, systems and components. [H.3(a)] (Section 40A2.5k)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," involving the failure to follow procedural requirements for performing operability determinations. Specifically, the licensee's operability evaluation for a degraded turbine-driven auxiliary feedwater pump steam admission valve failed to address all the specified safety functions of the affected component as described in the Final Safety Analysis Report and design basis documents. For example, the operability determination incorrectly stated that manual closure of the valves was not a credited safety function and incorrectly assumed nonsafety-related instrument air would always be available to close the valves. This finding was entered into the licensee's corrective action program as Nuclear Notifications 200869281 and 200887620. The licensee's corrective actions included re-performing the evaluation and emphasizing with licensee staff the importance of ensuring all design basis information is considered in operability evaluations.

The finding was more than minor because it impacted the Mitigating Systems Cornerstones and its objective to ensure the availability and reliability of equipment that responds to initiating events. Using Inspection Manual Chapter 0609 the issue screened to a Phase 3 analysis because it represented a loss of safety function for greater than the allowed technical specification allowed outage time and it screened to greater than Green using the Phase 2 pre-solved worksheet. The senior reactor analyst determined that this finding was of very low safety significance (Green) based on a bounding calculation which assumed inoperability of the component for a year. The senior reactor analyst determined that the combined significance of these scenarios was a delta-core damage frequency of 1.3E-7/yr and a delta-large early release frequency of 4.2E-8/yr. Therefore the violation was determined to be of very low safety significance (Green). The analyst determined that the cause of the finding has a crosscutting aspect in the area of human performance associated with decision making. Specifically, the licensee utilized unsupportable assumptions in its evaluation that were not consistent with the Final Safety Analysis Report or the valve vendor manual. [H.1.b](Section 40A2.5a)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" in that the licensee failed to translate design basis information into procedures for the turbine-driven auxiliary feedwater pump steam admission valves. Specifically, the licensee did not translate into procedures the design requirements to manually close and gag the valves within 30 minutes in response to high energy line breaks, a fire in the auxiliary feedwater pump room, or a steam generator tube rupture event. This issue was entered into the licensee's corrective action program as Nuclear Notification 200887620. Immediate actions included posting a leveraging device for operators to use should it be necessary, training operators, and scheduling lubrication of the valves.

  The finding is more than minor because it impacted the Mitigating Systems Cornerstones and its objective to ensure the availability and reliability of equipment that responds to initiating events. The analyst screened the issue to more than one cornerstone due to its effect on early release (steam generator tube rupture), fire protection, and mitigating systems (high energy line break). The analyst performed a Phase 3 analysis that considered the effects of a high energy line break in the pump room, a steam generator tube rupture, and fires in the pump room and auxiliary feedwater pipe tunnel. The analyst determined that the combined significance of these scenarios was a delta- core damage frequency of 5.E-9/yr and a delta- large early release frequency of 1.6E-9/yr. Therefore, the violation was determined to be of very low safety significance (Green). The inspectors determined that cause of the finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program. Specifically, the licensee had previous opportunities to identify this problem when the valve was removed from the in-service testing program and when they evaluated relevant external operating experience. [P.1(a)](Section 40A2.5b)

- Green. The inspectors identified a noncited violation of Technical Specification 3.7.6, which requires, in part, that Condensate Storage Tank T-120 be operable. Specifically, the tank isolation valve 2HV5715 had been inoperable for a period greater than the allowed outage time of seven days while Unit 2 was in Modes 1, 2, and 3. The valve isolates nonseismic piping from the tank and is required to be manually closed within 90 minutes following a seismic event. The licensee had not performed preventive maintenance on the valve resulting in the valve failing to close during an in-service test on January 26, 2010. This finding was entered into the licensee's corrective action program as Nuclear Notification 200765235. The licensee's corrective actions included repairing the isolation valve.

  This finding is more than minor because it impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Phase 1, "Initial Screening and Characterization of Findings," a Phase 2 analysis was performed because the condensate storage, Tank T-120, was inoperable greater than that allowed in technical specifications. Phase 2 analysis resulted in a potential greater than Green issue therefore, a Phase 3 was performed.

  The analyst performed a Phase 3 using San Onofre seismic information and fragility data associated with the piping that could not be isolated because of the failed condition of valve 2HV5715. The frequency of a seismic event that would cause a pipe break and drain tank T-120 was estimated to be 2.7E-5/yr. Given a seismic event that causes a loss of offsite power (nearly 100 percent of seismic events that rupture the piping would also cause a loss of offsite power), operators are compelled by procedure to cool down and initiate shutdown cooling. The amount of water that is protected with valve 2HV5715 failed to open, which includes inventory from tank T-121 and water below the break line in tank T-120, given that operators close the working manual isolation valve within 30 minutes, is more than what is needed to get to shutdown cooling in natural circulation with only 1 of 2 steam generator atmospheric dump valves in operation, even if there is a 4-hour hold time at hot standby. The analyst estimated that the failure probability of operators to cool down and initiate shutdown cooling is 1.0E-2. Therefore, assuming a zero base case, the estimated delta- core damage frequency of the finding is 2.7E-5/yr. (1.0E-2) = 2.7E-7/yr.

The inspectors also determined that the cause of the finding has a crosscutting aspect in the area of human performance associated with resources in that the licensee did not ensure that equipment was available and adequate to assure nuclear safety by minimization of long-standing equipment issues in that the valve was not being maintained through a preventive maintenance program. [H.2(a)](Section 40A2.5c)

- Green. The inspectors identified a cited violation of Technical Specification 5.5.1.1.a, involving the failure to maintain adequate written procedures. Specifically, as of April 23, 2010, the licensee's controls over its backlog of procedure change requests associated with plant modifications were inadequate to prevent licensee personnel from using outdated procedures with known technical errors in the plant. The performance deficiency of failing to control the backlog of procedure changes, such that procedures with known technical errors were in use in the plant were previously identified by the NRC on two occasions and were documented as noncited violations 05000361; 05000362/2009003-09 and 2009009-02. Because the licensee failed to restore compliance within a reasonable time after the previous noncited violations were identified, this violation is being cited in a Notice of Violation in accordance with Section VI.a.1 of the NRC's Enforcement Policy. This finding was entered into the licensee's corrective action program as Nuclear Notification 200888919. The licensee's corrective action included immediate actions to administratively suspend these procedures until they could be revised and to evaluate changes needed to its program to prevent recurrence.

  The failure to maintain procedures covered by Regulatory Guide 1.33 is a performance deficiency. The finding is of more than minor significance because, if left uncorrected, the failure to maintain and control procedures would have the potential to lead to a more significant safety concern. Using Inspection Manual Chapter 0609, Phase 1,"Initial Screening and Characterization of Findings," the finding was determined to have a very low safety significance because the finding did not result in a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component, because problems were not thoroughly evaluated, such that the resolutions addressed the causes and extents of condition. This includes properly classifying and prioritizing conditions adverse to quality. [P.1(c)](Section 40A2.5h)

- Green. Two examples of a noncited violation of 10 CFR 50.65(a)(1) were identified involving the failure to monitor the unavailability time associated with equipment failures which were maintenance induced. The first example involved maintenance inadvertently bending the fuse holder contacts such that there was a loose connection on the power supply on the turbine-driven auxiliary feedwater pump resulting in its failure. The second example involved the failure to perform maintenance associated with a condensate storage tank isolation valve resulting in its failure during in-service testing. In both cases, if the licensee had assessed the unavailability time due to the maintenance induced failures, the systems would have exceeded the 10 CFR 50.65(a)(2) monitoring criteria, necessitating the systems to be placed in 10 CFR 50.65(a)(1) goal setting. The licensee's corrective actions included evaluating its procedures to prevent recurrence, and re-evaluating these systems to determine the impact of accounting for unavailable time.

  This finding is more than minor because it affects the equipment performance attribute of the Mitigating Systems Cornerstone per Inspection Manual Chapter 612, Appendix B. Using Inspection Manual Chapter 0609, Phase 1, "Initial Screening and Characterization of Findings," the inspectors determined the finding to be of very low safety significance (Green) because they did not represent the loss of a system safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of the finding was determined to have a crosscutting aspect in the area of human performance. Specifically, personnel failed to use a formal decision making process to determine how to count unavailable hours for the maintenance rule. [H.1(a)](Section 40A2.5i)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," in that, from October 2008 to April 2010, the licensee failed to promptly identify and correct potentially degraded motor-driven relays in safety-related systems and components. Specifically, after identifying a degraded relay affecting an emergency diesel generator, the licensee replaced all similar relays in the other diesel generators but failed to evaluate the use of these potentially degraded relays in other safety-related systems. The licensee entered this issue into the corrective action program as Nuclear Notification 200146292, and developed a plan to replace the 62 degraded relays that were installed in other safety-related equipment.

  This finding was more than minor because it impacted the equipment performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609.04, Phase 1, "Initial Screening and Characterization of Findings," the inspectors determined the finding to be of very low safety significance (Green) because it did not represent the loss of a system safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with the decision-making component, in that the licensee did not use conservative assumptions in making decisions about the extent of condition [H.1(b)](Section 40A2.5j)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the failure to translate nonconservative errors in calculations and procedures identified during review of external operating experiences. The first example involved the sizing calculation for the condensate storage tank failing to account for effects of auxiliary feedwater pump heat during recirculation. The second example involved the failure to update procedural guidance concerning the adverse effects of placing the low pressure safety injection system into operation following use of the residual heat removal system in the shutdown cooling mode of operation above 200°F. This issue was entered into the licensee's corrective action program as Nuclear Notification 200886265. The licensee initiated actions to correct its procedure and calculation for each instance.

  The finding is of more than minor significance because it adversely affects the design control attribute of the mitigating systems cornerstone objective. Using Inspection Manual Chapter 0609.04, Phase 1, "Initial Screening and Characterization of Findings," the finding was determined to have a very low safety significance (Green) because the finding did not result in a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee failed to implement and institutionalize operating experience information, including vendor recommendations, through changes to plant processes, procedures, equipment, and training programs. [P.2(b)](Section 40A2.5l)

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a noncited violation of Technical Specification 5.5.1.1.a, "Scope," involving the failure to establish procedures for component cooling water system alignments such that leakage of radionuclides to the environment would be monitored during all operational alignments of component cooling water. Specifically, radiation monitors could be aligned to only one train of component cooling water at a time and the licensee's procedures had no provision for monitoring the second train when both trains were in-service. This finding was entered into the licensee's corrective action program as Nuclear

- Notification 200871387, and actions were implemented to require periodic grab sampling of the train which was not being monitored.

  The inspectors determined that this finding was more than minor because this issue impacted the Public Radiation Protection Cornerstone and its objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Specifically, the radiation monitors for component cooling water were not sufficient to ensure adequate release measurements. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609.04 and determined that the finding screened to Inspection Manual Chapter 0609, Appendix D, "Public Radiation Safety Significance Determination Process." The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix D, and determined that the finding was of very low safety significance (Green) because dose did not exceed Appendix I criteria. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that the plant operators did not have a low threshold for identifying deficiencies in procedures. [P.1(c)](Section 4OA2.5g)

Cornerstone: Miscellaneous

- Severity Level IV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73, "Licensee Event Report System," in which the licensee failed to submit a licensee event report within 60 days following discovery of an event meeting the reportability criteria. On January 26, 2010, the valve which isolates nonseismic piping from condensate storage tank T-120 failed its in-service test when the hand wheel stem snapped after a leveraging device was used in an attempt to close the valve. This isolation valve, 2HV5715, must be closed within 90 minutes of an operating basis earthquake in order to prevent the loss of condensate storage tank T-120 water inventory from a line break in the nonseismic portion of the condensate system. The failure of this valve resulted in a condition prohibited by Technical Specification 3.7.6 and therefore was reportable. This finding was entered into the licensee's corrective action program as Nuclear Notification 200888616, and the licensee was taking actions to send a licensee event report to the NRC for this event.

  The inspectors determined that traditional enforcement was applicable to this issue because the NRC's regulatory ability was affected. Specifically, the NRC relies on the licensee to identify and report conditions or events meeting the criteria specified in regulations in order to perform its regulatory function. The inspectors determined that this finding was not suitable for evaluation using the significance determination process, and as such, was evaluated in accordance with the NRC Enforcement Policy. The finding was reviewed by NRC management, and because the violation was determined to be of very low safety significance, was not repetitive or willful, and was entered into the corrective action program, this violation is being treated as a Severity Level IV noncited violation consistent with the NRC Enforcement Policy. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that the licensee failed to appropriately evaluate corrective maintenance as a basis for past operability. [P.1(c)](Section 4OA2.5d)

- Green. The inspectors identified a Green finding associated with the licensee's failure to meet the actions described to the NRC in letters dated April 21, 2009, and October 29 and 30, 2009, addressing corrective actions to improve site performance in the areas of human performance and problem identification and resolution. Specifically, 16 actions were not implemented on time and a number of actions were modified from what was previously described, all prior to informing the NRC. These findings were documented in Nuclear Notification 200848923.

  The inspectors determined that the licensee's failure to perform actions as documented in its plan to the NRC was more than minor because if left uncorrected could result in a more significant safety concern. Using Inspection Manual Chapter 0609, Appendix M, this finding was reviewed by NRC management and was determined to be of very low safety significance (Green). This finding has a crosscutting aspect in the areas of human performance. (Section 4OA2.5m)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers (condition report numbers) are listed in Section 4OA7.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution (71152)

The inspectors based the following conclusions on the sample of corrective action documents that were initiated in the assessment period, which ranged from October 1, 2008, to the end of the on-site portion of this inspection on April 23, 2010.

.1 Assessment of the Corrective Action Program Effectiveness

a. Inspection Scope

Approach and Scope: The inspectors visited San Onofre Nuclear Generating Station from December 14 through 17, 2009, to review the sites corrective action and maintenance backlogs. The backlog review included corrective actions, maintenance actions and administrative actions involving pending procedure changes.

The results of these reviews were used to select issues involving risk important systems and operator actions that would be reviewed during future inspections.

The following areas were identified for future inspection:

- Agastat relay failures
- Medium voltage breakers conditions (safety related and nonsafety related)
- Switchyard breaker maintenance practices
- Switchyard transformers conditions
- Backlog of pending procedure changes
- Component power supplies problems
- Aged electrolytic capacitors
- High relay and breaker auxiliary contact resistance
- Electrical grounds
- Boric acid leaks
- Emergency core cooling system voids

- Reactivity control (chemical and volume control system)
- Mitigating systems performance indicator trending
- Component cooling water system voids
- Component cooling water pump in runout conditions
- Drifting undervoltage relays setpoint
- Auxiliary feedwater pump problems
- Battery room hydrogen monitors
- Discolored 4kV and 480V Cables (and thermography results)
- DC Bus 3D1 low voltage
- High pressure safety injection swing pump logic problems
- Charging pump oil leaks
- Emergency diesel generators degraded conditions
- Pending plant modifications
- Control room annunciator problems
- Operator workarounds/operator burdens

The inspectors reviewed approximately 300 condition reports, including associated root cause, apparent cause, and direct cause evaluations, that were initiated between October 1, 2008, and April 5, 2010, to determine if problems were being properly identified, characterized, and entered into the corrective action program for evaluation and resolution. The inspectors also reviewed system health reports, operability determinations, self assessments, trending reports, metrics, and various other documents related to the corrective action program. The inspectors reviewed work requests and attended the licensee's corrective action review board and closure review board meetings to assess the reporting threshold and prioritization processes. The inspectors' review included verifying that the licensee considered the full extent of cause and extent of condition for problems, as well as how the licensee assessed generic implications and previous occurrences. The inspectors assessed the timeliness and effectiveness of corrective actions, completed or planned, and looked for additional examples of similar problems.

The inspectors also reviewed a sample of corrective action documents that addressed past NRC-identified violations to ensure that the corrective actions addressed the issues as described in the inspection reports. The inspectors reviewed a sample of corrective actions closed to other corrective action documents to verify that corrective actions were appropriate and timely.

The inspectors considered risk insights to focus the sample selection and plant tours on risk significant systems and components. Based on this review, the samples reviewed by the inspectors focused on, but were not limited to, these systems. The inspectors also expanded its review to include five years of evaluations involving the salt water cooling system and various electrical components to determine whether problems were being effectively addressed. The inspectors conducted a walkdown of these systems to assess whether problems were identified and entered into the corrective action program.

b. Assessments

i. Assessment - Effectiveness of Problem Identification

In general, the inspectors found that the licensee has been identifying problems and entering them into their corrective action program at appropriately low thresholds. For example, San Onofre Nuclear Generating Station personnel had identified and initiated over 20,000 nuclear notifications into the corrective action process in 2009. The inspectors identified many examples of failures to document problems into the corrective action program resulting in missed opportunities for the licensee to identify problems and adverse trends. In addition, there were several issues that took significant NRC interaction with site staff in order for them to recognize the problem. Examples of ineffective identification of issues include the following:

- The licensee failed to identify design basis information regarding the steam admission valves to the turbine auxiliary feedwater pump. On April 5, 2010, inspectors identified a concern that the valves might not be able to be manually closed due to the apparent lack of lubrication and rust on the Unit 3 valve stems (3HV8200 and 3HV8201). These valves are normally held open under spring pressure and are normally closed with nonsafety-related instrument air. In cases where instrument air is not available, the valve may be closed manually by rotating a hand wheel approximately 24-25 rotations. The inspectors reviewed design basis documents and the Final Safety Analysis Report and found that the valves must be manually closed within 30 minutes for certain accident sequences where instrument air is not available. Based on the inspectors' concern that manually closing the valve would be challenged with the lack of lubrication, San Onofre Nuclear Generating Station conducted an operability determination on April 10, 2010. However, the inspectors found that the operability evaluation was inadequate and did not consider design basis information. After significant NRC interaction, the licensee consulted with the vendor and found that the valves could not be manually closed even under ideal lubrication conditions because the force required to manually turn the hand wheel exceeded the licensee's guideline for the amount of force an individual could be expected to exert. Prior to the inspectors' questioning, the licensee had failed to identify the force needed to manually close the valve as well as other design basis information. (Section 40A2.5b)

- San Onofre Nuclear Generating Station failed to identify that the failure of isolation valve 2HV5715 was reportable to the NRC. This valve isolates nonseismic piping from the seismic piping on condensate storage tank T-120. This valve must be closed within 90 minutes of an operating basis earthquake to prevent the tank from draining its water through a postulated break in the nonseismic piping. On January 26, 2010, an operator attempted to perform the 2-year in-service test to manually stroke the valve by rotating its hand wheel. When the hand wheel would not turn, the operator followed procedure and contacted the control room to obtain permission to use a leveraging device to turn the hand wheel. When the operator used the leveraging device, the hand wheel sheared off. San Onofre Nuclear Generator Station reportability determination concluded the event was not reportable because a mechanic could be called to disassemble the valve actuator and manually close the valve with a wrench. During the weeks of April 5 and April 19, the inspectors informed San Onofre Nuclear Generator Station staff that this use of corrective maintenance was inappropriate to consider for reportability determination. The licensee maintained this position through a "white paper" developed on May 7, 2010. Subsequently, the inspectors contacted the licensee and referred the licensee to the specific guidance in NUREG 1022, whereby the licensee changed its position. (Section 40A2.5d)

- The licensee failed to identify that a nuclear notification had not been written, as required by procedure, to document that a leveraging device had been used when an operator sheared the hand wheel off of the isolation valve (2HV5715) which isolates nonseismic piping from the seismic piping on condensate storage tank T-120. (Section 40A2.5c)

- The inspectors questioned the ability of plant equipment operators to identify plant problems during plant tours as a result of knowledge deficiencies identified by the inspectors. On April 7, 2010, inspectors observed an experienced plant equipment operator performing his daily rounds for several hours.

The inspectors found that the nonlicensed operator did not demonstrate fundamental knowledge regarding such items as separation distances between scaffolding and safety-related equipment, expected panel configurations, and requirements for standard items like chocking carts. As a result, the inspectors determined that given these knowledge weaknesses exhibited by an experienced equipment operator they were limited in their ability to identify plant problems.

- The inspectors identified that some plant personnel appeared to accept degraded or unacceptable conditions rather than identifying the condition through the corrective action process and getting them corrected. Examples included: (1) the common use of leveraging devices which can mask degraded conditions; (2) there were a number of control room alarms that had not been cleared in preparation for the Unit 2 startup from the steam generator replacement outage; (3) the inspectors identified that one control room alarm had been locked in for four days because data on a computer card needed to be downloaded; (4) after inspectors questioned control room operators about the vibration and loose parts monitor alarm, control room staff realized that they were in day 5 of a 30-day action statement required by licensee controlled specifications; and (5) the inspectors identified that unsecured equipment in the switchyard that had been there for months in violation of licensee procedures even though operators had been performing routine rounds and others had been going in and out of the area.

- The inspectors noted that operators were not sensitive to a condition involving the failure to have adequate procedures to ensure that for all operational alignments of the component cooling water system radiation monitoring would be in effect to detect system leakage. (Section 40A2.5g)

.ii <u>Assessment - Effectiveness of Prioritization and Evaluation of Issues</u>

The inspectors found many instances where the licensee had correctly prioritized and evaluated issues. In fact, there was objective evidence that the quality of cause evaluations had improved during this inspection period. However, the inspectors also found that San Onofre Nuclear Generating Station continued to have significant challenges performing these actions consistently. While most initial operability determinations were appropriate, the inspectors identified several examples where poor evaluations were performed. The following are examples of ineffective or inadequate evaluation of issues:

- San Onofre Nuclear Generating Station staff performed an inadequate evaluation of the reportability of the failure of the isolation valve for condensate storage tank T-120. (Section 40A2.5d)

- San Onofre Nuclear Generating Station staff performed an inadequate operability determination of the steam admission valves to the turbine-driven auxiliary feedwater pumps after the inspectors raised concerns about lack of stem lubrication. (Section 40A2.5a)

- San Onofre Nuclear Generating Station staff performed an inadequate extent of condition evaluation involving potentially degraded Potter & Brumfield motor driven rotary relays. (Section 40A2.5j)

- An operability determination was inadequate evaluating a loose electrical connection in high pressure safety injection motor cubicle 2A0608. Specifically, the method used to evaluate the circuit continuity did not properly take into account the circuit operation. The licensee initiated Nuclear Notification 200871532 on April 9, 2010, to evaluate the inspector's concern.

- The inspectors reviewed a root cause, three apparent causes, and one common cause evaluation dealing with operators failing to properly make correct operability determinations. In one example, operators failed to declare an atmospheric dump valve inoperable based on testing results and failed to write a nuclear notification when the degraded conditions changed. Additionally, the inspectors identified that the licensee had also failed to implement all its corrective actions associated with this example.

iii. <u>Assessment – Effectiveness of Corrective Action Program</u>

The inspectors concluded that actions to correct conditions adverse to quality were generally adequate; however, there were notable examples where the licensee had not implemented effective corrective actions. Some examples included:

- Licensee actions to correct substantive crosscutting issues have not been effective. Despite actions to reverse the trend, San Onofre Nuclear Generating Station has experienced five consecutive assessment cycles with an increasing number of substantive crosscutting issues.

- San Onofre Nuclear Generating Station actions have not been effective in responding to two previously issued NCVs dealing with prioritizing the large backlog of procedure change requests. During this inspection, the inspectors found that procedure changes were not implemented following modifications to the instrument air system. The inspectors concluded that San Onofre Nuclear Generator Station corrective action to two previously issued noncited violations for the same issue were not fully effective. This violation is being cited as a Notice of Violation. (Section 40A2.5h)

- The licensee's actions to improve the conduct of operations in the control room have not been effective based on control room observations, which identified ineffective use of place keeping, use of 3-way communications, announcing alarms to the control room supervisor, and review of prerequisites prior to procedures being implemented. (Section 40A2.5e)

.2 Assessment of the Use of Operating Experience

a. Inspection Scope

The inspectors examined the licensee's program for reviewing industry operating experience, including reviewing the governing procedure and self-assessments. A sample of operating experience notification documents that had been issued during the assessment period were reviewed to assess whether the licensee had appropriately evaluated the notification for relevance to the facility. The inspectors also examined whether the licensee had entered those items into their corrective action program and assigned actions to address the issues. The inspectors reviewed a sample of root cause evaluations and significant condition reports to verify if the licensee had appropriately included industry operating experience.

b. Assessment

Overall, the inspectors determined that the licensee had appropriately evaluated industry operating experience for relevance to the facility, and had entered applicable items in the corrective action program. The inspectors noted that operating experience was considered in cause evaluations. The licensee failed to incorporate two of the four operating experience evaluation results into plant operating procedures and design documents. This is documented as a violation of 10 CFR Part 50, Appendix B, Criterion III. (Section 40A2.5l)

.3 Assessment of Self-Assessments and Audits

a. Inspection Scope

The inspectors reviewed a sample of licensee self-assessments and audits to assess whether the licensee was regularly identifying performance trends and effectively addressing them. The inspectors also reviewed audit reports to assess the effectiveness of assessments in specific areas. The specific self-assessment documents and audits reviewed are listed in the attachment.

b. Assessment

The inspectors concluded that the licensee had an effective self-assessment process. Licensee management was involved in developing the topics and objectives of self-assessments. Attention was given to assigning inspectors members with the proper skills and experience to do an effective self-assessment and to include people from outside organizations. Audits were self-critical and identified deficiencies in various programs such as the corrective action program and the equipment reliability program.

.4 Assessment of Safety-Conscious Work Environment

a. Inspection Scope

From February 1-10, 2010, a inspectors conducted 40 focus group sessions consisting of approximately 8-10 individuals each. The focus groups were conducted to assess the safety-conscious work environment at the San Onofre Nuclear Generating Station. The results of the focus groups were documented in NRC Inspection Report 05000361;05000362/2009009 dated March 2, 2010, and in the NRC's Chilling Effect Letter issued to San Onofre dated March 2, 2010.

b. Assessment

As documented in the NRC's March 2, 2010, Chilling Effect Letter, the NRC concluded that some employees in multiple workgroups at San Onofre Nuclear Generating Station have the perception that they are not free to raise safety concerns using all available avenues, and that management has not been effective in encouraging employees to use all available avenues without fear of retaliation. This conclusion resulted from numerous observations, including: (1) employees expressing difficulty or inability to use the corrective action program; (2) a lack of knowledge or mistrust of the Nuclear Safety Concerns Program (NSCP); (3) a substantiated case of a supervisor creating a chilled work environment in his/her work group; and (4) a perceived fear of retaliation for raising safety concerns. The licensee replied by letter dated March 31, 2010. Further actions by the NRC are discussed in the March 2 letter.

.5 Specific Issues Identified During This Inspection

a. Inadequate Operability Determination for Turbine-Driven Auxiliary Feedwater Pump Steam Admission Valves

Introduction. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified involving the failure to perform an adequate operability determination as required by procedure. Specifically, the licensee's operability evaluation for a degraded turbine driven auxiliary feedwater pump steam admission valve failed to address all the specified safety functions of the affected component as described in the final safety analysis report and design basis documents.

<u>Description</u>. On April 7, 2010, inspectors noted what appeared to be unlubricated valve stems on the Unit 3 steam admission valves to the turbine-driven auxiliary feedwater pump, which are designated as 3HV8200 and 3HV8201. These valves are normally held open by spring pressure and are normally closed with nonsafety-related instrument air. The design basis requires that for certain accident sequences in which the nonsafety-related instrument air system is unavailable, these valves must be manually closed within 30 minutes. The valves are manually closed by turning their respective hand wheel about 25 rotations. The valves are provided with manual gagging (locking) devices to force the valves closed without instrument air and to lock the valves closed, such that they won't inadvertently re-open. The inspectors were concerned that increased friction from an unlubricated valve stem would make turning the hand wheel against the spring force more difficult during manual operation.

The inspectors identified the issue to the licensee and noted that design basis documents required the valves be manually closed and "gagged" or locked in the following accident scenarios: (1) a high energy line break in the auxiliary feedwater pump room; (2) a steam generator tube rupture; and (3) a fire in the auxiliary feedwater pump room. The inspectors also discussed the design bases with the licensee. As a result of the inspectors' concern, the licensee initiated Nuclear Notification 200869281, and on April 8, 2010, commenced an operability determination. On April 10, 2010, San Onofre personnel completed the operability determination and concluded that the unlubricated gagging devices were operable. However, the inspectors found that the operability determination was inadequate.

The operability determination concluded that the valves were used in the postfire safe shutdown analysis which was addressed by the notification, but did not address the impact on technical specification operability. The operability determination stated that the valves could be manually closed and gagged but it provided no technical basis for the statement. Inspectors reviewed San Onofre Procedure SO123-XV-52, "Functionality Assessments and Operability Determinations," Revision 15. Step 6.5.1 required that the immediate operability determination identify the specified safety function of the affected system, structure or component. Step 6.5.1.3.2 stated that the operability determination must identify the performance parameter used to determine operability.

Inspectors found that the April 10, 2010, operability determination was inadequate because it failed to identify the performance parameters used to determine operability; specifically, the design basis for these valves. Inadequacies included:

i. The determination incorrectly stated, "Manual closure of 2HV8200 and 2HV8201 is not a credited safety function of these valves for emergency operating events." This was contrary to Final Safety Analysis Report Table 10.4-7, which described use of the valves during a high energy line break without the use of instrument air. Final Safety Analysis Report Section 15.6.3, described valve closure and release termination within 30 minutes of a steam generator tube rupture. Design Basis Document SD-SO23-780 also described the manual action to gag the valves closed.

ii. The determination incorrectly assumed that nonsafety-related instrument air would always be available to stroke the valves closed from the control room.

iii. The determination incorrectly assumed the valves are closed against zero opposing force to prevent them from re-opening on an auxiliary feedwater start signal.

iv. The determination cited a procedure that only stated to close the valve, but the procedure did not state during which events the valves should be closed. This instruction was apparently only used during maintenance of the valves or terry turbine.

Based on interviews, operations and engineering were not specifically aware that the valves needed to be manually gagged closed even though the inspectors discussed the design basis with other licensee personnel. After additional inspectors' questioning and re-review of the design basis and the gag operation with San Onofre Nuclear Generating Station personnel, San Onofre Nuclear Generating Station re-performed the operability determination under Nuclear Notification 20088760 and completed it on April 21, 2010. Based on the second operability determination, which included contacting the vendor, licensee personnel informed the inspectors on April 22, 2010, that the valves were declared inoperable and that the licensee was taking interim compensatory corrective actions. Thus, the licensee's initial operability determination on April 10, 2010, had been inadequate even after the inspectors had discussed the design basis with licensee personnel prior to the licensee's evaluation.

The licensee documented this violation in Nuclear Notification 20088760, and its short term corrective actions included required training and the staging of a leveraging device in the vicinity of the valves to assist operators in closing and/or gagging the valves, as required.

Analysis. Inspectors found that the failure to perform an adequate operability determination and to identify the degraded condition was a performance deficiency. The deficiency was more than minor because it impacted the Mitigating Systems Cornerstones and its objective to ensure the availability and reliability of equipment that responds to initiating events. Using Inspection Manual Chapter 0609, the issue screened to Phase 3 because it represented a loss of safety function for approximately two weeks and it screened to greater than Green using the Phase 2 pre-solved worksheet. The inspectors determined that the finding was Green based on the bounding analyses discussed in the analysis section of 4OA2.5b. Specifically, this vulnerability existed for approximately two weeks (the time between the inadequate evaluation and the correct evaluation), which is considerably less than the one year vulnerability discussed in the analysis section of 40A2.5b. The inspectors determined that the cause of the finding has a crosscutting aspect in the area of human performance associated with decision making. Specifically, San Onofre Nuclear Generating Station utilized unsupportable assumptions in its evaluation that were not consistent with the Final Safety Analysis Report or the valve vendor manual. [H.1.b]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality be prescribed by procedures and be accomplished in accordance with those procedures. San Onofre Procedure SO123-XV-52, "Functionality Assessments and Operability Determinations," Revision 15, Step 6.5.1 requires, in part, that the immediate operability determination identify the specified safety function of the affected system, structure or component. San Onofre Procedure SO123-XV-52 Step 6.5.3.1 requires, in part, that the operability determination must identify the performance parameter used to determine operability. Contrary to the above, from April 10 to April 22, 2010, San Onofre Nuclear Generating Station performed an inadequate operability determination required by San Onofre Procedure SO123-XV-52. Specifically, San Onofre Nuclear Generating Station failed to identify the design basis parameters for the steam admission valves for the turbine-driven auxiliary feedwater pumps as described in the Final Safety Analysis Report and design basis documents. In accordance with the NRC's Enforcement Policy, because the violation was of very low safety significance, and was entered into the corrective action program as Nuclear Notification 20088760, this violation is being treated as noncited violation, consistent with the NRC Enforcement Policy VI.A: NCV 05000361/2010006-01, "Inadequate Operability Determination for turbine-driven auxiliary feedwater pump steam admission valves."

b. Failure to Translate Design Basis Information for Closure of Turbine-Driven Auxiliary Steam Admission Valves

Introduction. On April 7, 2010, inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for steam admission valves to the turbine-driven auxiliary feedwater pumps that could not be closed within 30 minutes per the design basis.

Description. As discussed in the previous section, on April 7, 2010, inspectors found apparently unlubricated valve stems on the Unit 3 steam admission valves to the turbine-driven auxiliary feedwater pump, which are designated as 3HV8200 and 3HV8201. The inspectors identified a Green noncited violation related to the inadequate operability determination that licensee personnel performed on April 10, 2010.

On April 22, 2010, after performing a second operability determination, the licensee representatives informed the inspectors that they had contacted the vendor and found the valves were inoperable because a person would not be able to manually close and gag the valves under ideal lubrication conditions. The licensee's standard was that a person would be able to apply a force of up to 100 pounds. However, ideally lubricated valves would require 132 pounds of force on the hand wheel, and the hand wheel would have to be turned approximately 25 times in order to close and gag the valve. The increased friction from lack of lubrication and disc-in-seat forces could exceed 200 pounds of force on the hand wheel. As a result of this information, the licensee began taking corrective actions by posting leveraging devices for operators to use in the event manual closure of the valves was needed. On April 22, 2010, required reading on valve operation was implemented to train all licensed and nonlicensed operators on the valve operation.

For postfire safe shutdown procedures, damage to the 2(3)HY8200 and 2(3)HY8201 solenoid valves associated circuit cables routed to auxiliary relay cabinet L071 could cause a loss of the ability to close the air-operated 2(3)HV8200 and 2(3)HV8201 from the control room. San Onofre Procedure SO23-13-21 "Fire" provided instructions for operators to mitigate the effects of fire damage to safe shutdown equipment in plant areas. The steam admission valves are required to be closed within 30 minutes of a fire by the postfire safe shutdown analysis. Based on the April 22, 2010, operability determination, the licensee added steps to San Onofre Procedure SO23-13-21 to use a crescent wrench and leverage device. These tools were locally staged to back off the hand wheel stem nut and then use the leverage device on the hand wheel to force the gag to shut the valve against its opening spring. The inspectors concluded that prior to April 22, 2010; manual actions could not have been taken within the 30-minute period because of the lack of tools and the operator's lack of familiarity with San Onofre Procedure SO23-13-21 which identified key manual actions needed.

The inspectors noted the following prior opportunities the licensee had to identify this deficiency:

i. In 2004, Action Request 040700869 erroneously stated that the safety function to close the 8200 valves was not required in the design basis document.

ii. In 2005, Action Request 050700659 was written to request that design engineering delete the manual closure of the valves from the ASME O&M Code in-service testing based on an incorrect evaluation which stated that the valves were not part of the accident analysis. The action request also erroneously stated the valves would not impact other programs such as fire protection.

iii. On November 19, 2009, the licensee failed to identify this issue during its review of Operating Experience 30062, "Locally Operated Time Critical

Valves May be Difficult to Operate Under Accident Conditions" which dealt with the possibility that the expected differential pressure across locally operated valves must be considered when evaluating the ability of operators to change valve positions in accident conditions. The operating experience stated that this evaluation should be similar to the review required by Generic Letter 89-10 for valves locally operated under high differential pressure.

On April 22, 2010, San Onofre Nuclear Generating Station's corrective action was to post leveraging devices and to schedule lubrication of the valves for August 2010. The NRC considered immediate lubrication to be an important corrective action that the licensee had not adequately addressed while the inspectors were onsite. In addition, because a dry lubricant was used on the valve (in accordance with the manufacturer's recommendations) and the valve was exposed to the weather, the inspectors also questioned the 10 year frequency for lubrication. Based on further questioning from the inspectors, on May 25, 2010, the licensee wrote Nuclear Notification 200937258 to address the inspectors' concern about the adequacy of lubrication of the valve stem as well as the frequency of lubrication.

The inspectors concluded that prior to April 22, 2010; the 8200 series valves had been inoperable because the licensee had not translated the design basis into procedures. The licensee did not translate into its procedures the design bases requirements to manually close the valves within 30 minutes of the required accident scenarios and did not consider the force needed to manually close and gag the valves. Inspectors also found that the licensee was not meeting Licensee Controlled Specification Surveillance Requirement 3.7.113.1.12 to manually stroke the valve every 24 months to ensure compliance with the fire protection program. In addition, simulated operator actions during a walkthrough of San Onofre Procedure SO23-13-21, "Fire," could not be performed in the time specified in engineering calculations, nor were all appropriate steps specified. The licensee was also evaluating necessary actions for a permanent corrective action to this issue.

Analysis. The inspectors found that the failure to translate design basis information regarding the 2(3)HV8200 and 2(3)HV8201 valves into procedures was a performance deficiency. The deficiency was more than minor because it impacted the Mitigating Systems Cornerstones and its objective to ensure the availability and reliability of equipment that responds to initiating events. The inspectors screened the issue to more than one cornerstone due to its affect on early release (steam generator tube rupture), fire protection, and mitigating systems (high energy line break).

The inspectors screened the issue to Phase 2, Inspection Manual Chapter 0609 Appendix H, because the finding represents an actual open pathway in the physical integrity of reactor containment during a steam generator tube rupture accident scenario. In Inspection Manual Chapter 0609 Appendix H, Step 4.1, the inspectors screened this as a Type B finding (affects large early release fraction but not core damage frequency) needing a Phase 2 evaluation. Inspectors used Table 4.1 and found that the finding involved a large release path from the reactor coolant system to the environment. Using Table 6.2, inspectors screened the Phase 2 to greater than Green because the condition existed for greater than one year and the volume of steam released would be larger than the free volume of containment.

The inspectors screened the issue to Phase 2 for at-power inspection findings using Inspection Manual Chapter 0609 because the turbine-driven auxiliary feedwater valves could not be closed within 30 minutes after a high energy line break to prevent failure of the two remaining auxiliary feedwater pumps. This represented the potential loss of a safety function.

Inspectors screened the issue to Phase 2 for Appendix F of Inspection Manual Chapter 0609 because the valves could not be closed for a fire in the auxiliary feedwater room.

The senior reactor analyst performed a Phase 3 analysis to determine the risk significance of the degraded turbine-driven auxiliary feedwater steam admission valve. The analysis considered the effects of a high energy line break in the pump room, a steam generator tube rupture, and fires in the pump room and auxiliary feedwater pipe tunnel. The inspectors determined that the combined significance of these scenarios was a delta-core damage frequency of 1.3E-7/yr and a delta-large early release frequency of 4.2E-8/yr. Therefore, the violation was determined to be of very low significance.

The violation has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program. Specifically, the licensee had multiple opportunities to evaluate this problem but failed to do so. [P.1(a)]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that the design basis for systems, structures, and components be correctly translated into specifications, drawings, and procedures. Contrary to the above, prior to April 22, 2010, the licensee failed to translate the following design basis information into procedures: (1) the requirements to manually close and gag within 30 minutes the steam admission valves for the turbine driven auxiliary feedwater pump in response to high energy line breaks or steam generator tube rupture; and (2) the failure to determine the forces required to manually close the valves. Because the violation was of very low safety significance (Green), and was entered into the corrective action program as Nuclear Notification 200870861 this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy section VI.A: NCV 05000361/2010006-02, "Failure to translate design basis information for turbine-driven auxiliary feedwater pump Steam Admission Valves."

c. <u>Lack of Preventive Maintenance Results in Valve Failure of Condensate Storage Tank</u>

<u>Introduction</u>. A Green noncited violation of Technical Specification 3.7.6 was identified which requires, in part, that condensate storage tank T-120 be operable. Specifically, the tank isolation valve 2HV5715 had been inoperable for a period greater than the allowed outage time of seven days while Unit 2 was in Modes 1, 2, and 3. The valve isolates nonseismic piping from the tank and is required to be manually closed within 90 minutes following a seismic event. The licensee had not performed preventive maintenance on the valves resulting in the valves failing to close during an in-service test on January 26, 2010.

<u>Description</u>. On January 26, 2010, the hand wheel on the Unit 2 condensate storage tank manual valve 2HV5715 broke while licensee personnel attempted to perform an in-service test. This valve isolates nonseismic from seismic piping supporting condensate storage tank T-120. The design basis for the valve is to be closed within 90 minutes of an operating basis earthquake in order to preserve the water inventory in condensate storage tank T-120. The water inventory in that tank is needed to provide a water source for the auxiliary feedwater pumps to remove heat from the reactor. A line break in the nonseismic portion of the condensate system could drain tank T-120 of its water inventory, which is required to support plant cooldown from Mode 1 to Mode 5. Final Safety Analysis Report 10.4.9.2.3.4, "Emergency Operation," states that tank T-121 is the primary source of auxiliary feedwater condensate with tank T-120 required for backup.

The licensee employee performing the in-service test attempted to cycle the valve but was not able to rotate the hand wheel. So, in accordance with procedures, the licensee contacted the control room and obtained permission to use a leveraging device to turn the valve. When the licensee employee applied the leveraging device to the hand wheel, it sheared the pin connecting the hand wheel to the valve manual actuator stem. The valve was repaired the next day. During the subsequent diagnostics, the actuator stem was found to be heavily rusted and without lubrication. The licensee employee determined that the valve had been inadvertently removed from the preventive maintenance program several years prior.

At the time the valve failed, Unit 2 was in an outage and the valve was not required to be operable. Nuclear Notification 200765235 stated that the valve was inoperable and could not fulfill its safety function to preserve the water inventory in condensate storage tank T-120. However, in determining past operability, emails were attached to Nuclear Notification 200765235 that stated that corrective maintenance could be performed to open the valve; specifically, that a mechanic could have been called upon to disassemble the valve actuator and manually close the valve. Thus, the licensee concluded the valve was operable prior to January 26, 2010, and that the failure was not reportable.

The inspectors challenged the licensee in its determination that the valve had been operable prior to the hand wheel breaking and that the failure was not reportable to the NRC. The inspectors' position was that it was inappropriate to consider corrective maintenance in the reportability determinations. The licensee originally maintained its position asserting that the valve was operable and that the issue was not reportable, again basing its decision on corrective maintenance. After the inspectors referred the licensee to appropriate NRC guidance in NUREG-1022, the licensee determined that the broken valve had not been operable prior to the event and that the event was reportable.

The inspectors also challenged the use of leveraging devices on isolation valve 2HV5715 as well as other manually-operated valves. Several other manual valve hand wheels in the area had markings indicative of extensive use of leveraging devices. Inspectors were informed that nuclear notifications were not being written each time leveraging devices were used on manual valves, which was required in accordance with Procedure SO123-0-A6, "Routine Equipment Operations," Revision 8. San Onofre Nuclear Generating Station is re-examining its in-service testing periodicity and preventive maintenance practices in Nuclear Notification 200952866. The inspectors also noted that a nuclear notification had not been written, as required, when the leveraging device was used on isolation valve 2HV5715 on January 26, 2010.

In order to determine whether the licensee could reasonably close the valve within 90 minutes of an operating basis earthquake, inspectors performed a walkdown of the actions licensee staff would take following a seismic event. The inspectors interviewed licensee staff who had not been informed of the inspectors question prior to the walk down. The inspectors proposed a scenario to the shift manager that the plant experienced an operating basis earthquake and assessed the time it would have taken before she/he would have contacted the maintenance general foreman to fix isolation valve 2HV5715. The inspectors then interviewed the maintenance general foreman in order to understand what she/he would do for this situation. The inspectors also interviewed three mechanics and gave them the scenario conditions. Reviewing the time line starting 90 minutes after the earthquake, the inspectors determined the total time to close the valve was approximately 105 minutes. Based on this data, the inspectors raised the concern to the licensee that its staff would be unable to meet its design basis for closing this valve following an operating basis earthquake.

Subsequent to the inspection, the licensee ran this scenario in the simulator (without announcing it to the crew in advance). The results were that it took the crew an estimated 134 minutes to have a mechanic manually turn the valve. Therefore, the licensee determined that its staff could not complete manually closing the valve within the 90 minute time frame required by the design basis, and began taking actions to review its licensing basis and its procedures, and conducted additional training to its staff.

The isolation valve was last stroked in March 2008, and the licensee could not determine the exact date the valve became inoperable. Given the failure mode, the inspectors concluded that the valve had been inoperable for greater than seven days when the licensee was last in Mode 1, 2 or 3, when the valve was required to be operable.

The licensee documented this deficiency in Nuclear Notification 200765235, and repaired the valve and placed it into the preventive maintenance program.

Analysis. The inspectors determined that the failure to perform preventive maintenance, including lubricating the valve actuator's components necessary to manually close valve 2HV5715, was a performance deficiency. This issue is more than minor because it impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the broken valve impacted the protection against external events attribute for seismic protection. The inspectors used Inspection Manual Chapter 0609, "Initial Screening and Characterization of Findings," to analyze the significance of this finding. The inspectors screened the finding to Phase 2 because the condensate storage tank T-120 was inoperable for a significant period greater than that allowed in technical specifications (using the time over two methodologies, the tank was inoperable for approximately a year). This screened the finding out of Phase 2 to Phase 3 because the closest surrogate for this deficiency was failure of one of the auxiliary feedwater pumps for one year which screened to red. A Phase 3 analysis was performed by the senior reactor analyst. Using San Onofre Nuclear Generating Station's seismic information and fragility data associated with the piping that could not be isolated because of the failed condition of valve 2HV5715, the frequency of a seismic event that would cause a pipe break and drain tank T-120 was estimated to be 2.7E-5/yr. Given a seismic event that causes a loss of offsite power (nearly 100 percent of seismic events that rupture the piping would also cause a loss of offsite power), operators are compelled by procedure to cool down and initiate shutdown cooling. The amount of water that is protected with valve 2HV5715 failed open, which includes inventory from tank T-121 and water below the break line in tank T-120, given that operators close the working manual isolation valve within 30 minutes is more than what is needed to get to shutdown cooling in natural circulation with only one of two steam generator atmospheric dump valves in operation, even if there is a 4-hour hold time at hot standby. The analyst estimated that the failure probability of operators to cool down and initiate shutdown cooling is 1.0E-2. Therefore, assuming a zero base case, the estimated delta-core damage frequency of the finding is 2.7E-5/yr. (1.0E)=2.7E-7/yr.

The inspectors also determined that the cause of the finding has a crosscutting aspect in the area of human performance associated with resources in that San Onofre Nuclear Generating Station did not ensure that equipment was available and adequate to assure nuclear safety by minimization of long-standing equipment issues in that the valve was not being maintained through a preventive maintenance program. [H.2(a)]

Enforcement. Technical Specification 3.7.6 requires, in part, that tank T-120 to be operable. Valve 2HV5715 is required for tank operability because it must be closed after an earthquake to preserve tank inventory. Condition C provides for a completion time of seven days. Contrary to the above, prior to January 26, 2010, valve 2HV5715 could not be closed for greater than its completion time of seven days. The valve was failed in the open position. Because this violation was of very low safety significance and was entered into the licensee's corrective action program under Nuclear Notifications 200765235. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000361/2010006-03, "Lack of preventive maintenance results in valve failure and inoperable condensate storage tank."

d. Failure to Submit a Licensee Event Report Within 60 Days

Introduction. On April 22, 2010, inspectors identified a Severity Level IV violation of 10 CFR 50.73, "Licensee Event Report System," in which the licensee failed to submit a licensee event report within 60 days following failure of condensate storage tank isolation valve 2HV5715.

Description. As previously discussed, on January 26, 2010, condensate storage tank T-120 manual isolation valve 2HV5715 failed its in-service stroke test after a leveraging device was used to turn the hand wheel, at which time it sheared off. The valve operator stem was heavily rusted and did not move resulting in the failure. This valve must be closed per San Onofre Procedure AOI SO23-13-3, "Earthquake," Revision 13, Attachment 1, Step 2.3.3 within 90 minutes of an operating basis earthquake in order to prevent the loss of water inventory from condensate storage tank T-120 from a line break in the nonseismic portion of the condensate system.

In determining reportability, an email was attached to Nuclear Notification 200765235 that stated a mechanic could disassemble the valve actuator and manually turn the valve. Thus, the licensee concluded the valve was operable prior to January 26, 2010, and that the failure was not reportable.

The inspectors challenged the use of corrective maintenance to determine that the valve was previously operable. Originally, licensee representatives informed the inspectors that its mechanics could pry the position indicator off and close the valve against the frozen operator using a wrench. The inspectors questioned the licensee on this position because this action would require turning the stem against the frozen operator thus potentially damaging the operator.

Following discussions licensee personnel provided a more reasonable position by stating that the valve operator could be unbolted and removed, and the butterfly disc stem could then be closed with a wrench. The inspectors determined this method was plausible, but still required corrective maintenance. The inspectors noted that the use of corrective maintenance did not meet

NUREG 1022, "Events Reporting Guidelines 10 CFR 50.72 and 50.73," guidance which states that operability must be ensured and that corrective maintenance is not an appropriate basis for operability. Tanks T-121 and T-120 are required to be operable per Technical Specification 3.7.6 to supply 24 hours of demineralized water to the auxiliary feedwater system. Without the ability to close valve 2HV5715, tank T-120 was not operable. Because the tank was not operable, it met the conditions of 10 CFR 50.73(a)(2)(v) as an event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, and mitigate the consequences of an accident. As such, the event was reportable under 10 CFR 50.73(a)(1).

The licensee maintained its position that based on corrective maintenance the condition was not reportable until the inspectors pointed out the section in NUREG 1022, at which time, the licensee determined the condition was reportable.

The licensee documented this violation in Nuclear Notification 200888616, and the licensee took actions to issue a licensee event report.

Analysis. The failure to submit a licensee event report as required was a performance deficiency. The inspectors reviewed this issue in accordance with Inspection Manual Chapter 0612 and the NRC Enforcement Policy. The inspectors determined that traditional enforcement was applicable to this issue because the NRC's regulatory process was impacted. Specifically, the NRC relies on the licensee to identify and report conditions or events meeting the criteria specified in regulations in order for the NRC to perform its regulatory function, and when this is not done, the regulatory function is impacted. The inspectors determined that this finding was not suitable for evaluation using the significance determination process, and as such, was evaluated in accordance with the NRC Enforcement Policy. The finding was reviewed by NRC management, and the significance of the violation was classified at Severity Level IV and treated as a noncited violation consistent with the NRC Enforcement Policy. This finding was determined to have a crosscutting aspect in the area of human performance in the decision-making component in that the licensee did not make safety-significant decision using a systematic process, especially when faced with uncertainty. [H.1(a)]

Enforcement. Title 10 CFR 50.73(a)(1) requires, in part, that licensees shall submit a licensee event report for any event of the type described in this paragraph within 60 days after the discovery of the event. Title 10 CFR 50.73(a)(2)(v) identifies a reportable event as, in part, an event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, or mitigate the consequences of an accident. Contrary to the above, prior to March 27, 2010, San Onofre Nuclear Generating Station failed to submit a licensee event report within 60 days for the failure of valve 2HV5715 which could have prevented the fulfillment of the safety functions and was a condition prohibited by Technical Specification 3.7.6. Technical Specification 3.7.6 requires that tank T-120 be operable in Modes 1, 2, and 3 in order to supply 24 hours of demineralized water to the auxiliary feedwater system. Without the ability to close valve 2HV5715, tank T-120 was not operable. As a result, valve 2HV5715 is a component that is needed to remove residual heat and mitigate the consequences of an accident. All three trains of auxiliary feedwater could not perform their design function because there would be insufficient condensate inventory after an earthquake. In accordance with the NRC's Enforcement Policy, the finding was reviewed by NRC management and because the violation was of very low safety significance, and was entered into the corrective action program as Nuclear Notification 200888616, this violation is being treated as a Severity Level IV noncited violation, consistent with the NRC Enforcement Policy: NCV 05000362/2010006-04, "Failure to report conditions that could have prevented fulfillment of a safety function."

e. Failure by Control Room Operators to Follow Conduct of Operations Procedure

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.5.1.1.a, "Scope" for control room operators' failure to adhere to conduct of operations procedural requirements.

Description. On April 7, 2010, inspectors performed a detailed observation of control room activities for Units 2 and 3 at San Onofre Nuclear Generation Station. Unit 2 was performing a startup from a refueling outage and Unit 3 was operating at 50 percent rated thermal power. The inspectors observed a shift turnover from night shift to day shift and attended all turnover meetings. The inspectors watched Unit 2 operators perform startup activities that included a dilution to within 200 parts per million estimated critical boron concentration. Additionally, the inspectors observed Unit 2 operators withdraw control rods from shutdown bank 'B' and partial length control rods and viewed Unit 3 operators perform a dilution with primary water to maintain reactor power at 50 percent. The inspectors also monitored various routine control room activities such as acknowledging alarms, refilling the Unit 2 closed cooling water surge tank, and controlling pressure in the Unit 2 steam generators. The inspectors observed the control room operators interacting with other departments such as maintenance, health physics, engineering, and chemistry.

The inspectors compared actions in the control room with San Onofre Procedure SO123-0-A1, "Conduct of Operations," Revision 26, and observed numerous deficiencies. When Unit 2 alarms were received in the control room, the inspectors observed the following:

- Place keeping was not implemented on any unexpected alarms received in the Unit 2 control room per Section 6.4.3.3 and Guideline 5 of Section 6.4.3. Alarm response procedures were referred to by the reactor operators and read but no place keeping occurred.

- The operator announcing the alarm did not always report it to the control room supervisor as required by Section 6.4.3.3.

- When an alarm annunciated, or an alarm condition clears, all conversations in the control room did not stop until the alarm had been acknowledged or reset, as required by Section 6.4.3.1.

The following alarms were received in the Unit 2 control room during the inspectors' observations:

- Reactor Coolant Pump 4 seal pressure HI/LO

- Generator potential transformer fuse blown

- Channel 4 startup rate high

- Control Element Assembly Group Deviation (the senior reactor operator in charge of reactivity instructed the reactor operator to mark steps in alarm response procedure)

- Other alarms were received during the observation but marking of alarm response procedures were not normally performed

The inspectors observed a control room supervisor conduct a pre-job briefing at the beginning of shift. During the briefing, numerous questions were asked of the control room supervisor by on-shift operators on how the supervisor wanted the operators to control steam generator pressure. The questions or answers were not acknowledged using three-way communications to ensure full understanding took place as required by Section 6.6.4.7 of San Onofre Procedure SO123-0-A1. Throughout the inspector's observations, additional examples of missed three way communications were observed.

In addition, when an operator was performing the filling of the closed cooling water surge tank, the operator did not verify written instruction prerequisites before using the procedure as required by San Onofre Procedure SO123-0-A1.

During the pre-job brief for pulling shutdown group 'B' and partial length control rod groups, the reactivity senior reactor operator did not verbalize the five summarize, anticipate, foresee, evaluate and review questions as part of the brief. Only the operating experience question was discussed. Other items not reviewed as required included: (1) four questions dealing with critical steps, (2) error-likely situations, (3) how bad can it get, and (4) what defenses are in place and are they adequate. These were required by San Onofre Procedure SO123-0-A1.

During a Unit 3 reactivity change, the inspectors observed the control room supervisor performing oversight of the activity take a phone call while the evolution was in progress. The control room supervisor first engaged in conversation before informing the person that he would have to call them back later. This was contrary to Section 6.5.2, Step 6.5.2.8 which states, “All reactivity changes in the control room require direct senior reactor operator oversight. Senior reactor operator oversight requires the senior reactor operator be cognitive of, present for, and approve the reactivity change.”

The licensee documented these procedural deficiencies in Nuclear Notification 200871332 and its short term corrective actions included operation’s management reviewing the observations with the inspectors and then establishing a recovery plan to improve operator performance.

Analysis. The failure of control room operators to adhere to conduct of operations procedural requirements is a performance deficiency. The finding was more than minor because, uncorrected, the failure to follow these procedural requirements could lead to a significant safety concern due to the potential of operators making errors while operating safety-related systems. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined the finding had a very low safety significance because the finding did not result in a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. As a result, the issue was of very low safety significance (Green). The finding has a crosscutting aspect in the area of human performance associated with the work practices because the licensee did not ensure supervisory and management oversight of work activities. [H.4(c)]

Enforcement. Technical Specification 5.5.1.1.a, “Scope” requires, in part, that written procedures be established, implemented, and maintained covering the 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. Specifically Regulatory Guide 1.33 Section 1.d “Procedure Adherence,” requires operators to follow their procedures. San Onofre Procedure SO123-0-A1, “Conduct of Operations,” Revision 26, Sections 6.4.3.1, 6.6.4, and 6.5.2 require, in part, the following: implement alarm response procedure place keeping; announce alarms to the control room supervisor; stop conversations in the control room when an alarm annunciates; perform 3-way communications during pre-job briefing; review the five questions, summarize, anticipate, foresee, evaluate and review, during a pre-job brief; and review the prerequisites prior to each use of a procedure; and requires that a senior reactor operator remain cognitive of the reactivity change evolution.

Contrary to this, on April 7, 2010, control room operators failed to follow San Onofre Procedure SO123-0-A1, “Conduct of Operations,” Revision 26, requirements in numerous instances including failures to: implement alarm response procedure place keeping; announce alarms to the control room supervisor; stop conversations in the control room when an alarm annunciated;

perform 3-way communications during a pre-job briefing; review the five questions, summarize, anticipate, foresee, evaluate and review, during a pre-job brief; review the prerequisites prior to each use of a procedure; and remain cognitive of the reactivity change evolution by a control room supervisor. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification 200871332, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: 05000362/2010006-05; "Control Room Operators' Failure to Adhere to Conduct of Operations Procedural Requirements."

f. Failure to Provide Adequate Procedures for Boron Dilution Activities

Introduction. The inspectors reviewed a self-revealing Green noncited violation of Technical Specification 5.5.1.1.a, "Scope" for the failure of boron saturation procedure to have adequate direction to prevent an unplanned power transient.

Description. On December 25, 2009, the chemistry department requested operators to perform a reactor coolant system delithiation using ion exchanger 3ME074 for Unit 3. This ion exchanger was not boron saturated so the evolution would require diverting to radiological waste while performing a manual blended makeup.

The operating crew performed a pre-job brief prior to commencing the evolution where they discussed the procedures, expected plant response, and compensatory actions for power increase. The crew reviewed the logs and found the last blended makeup to be light in boron concentration which could result in a slight power increase. The crew was aware of Unit 3 having a feedwater heater leak that was identified on the previous shift. The leak was scheduled to be repaired later that day and would require a slight down power to remove the feedwater heater from service. The crew was concerned with exceeding the licensed power limit and therefore set an upper power limit of plus 0.5 percent. Due to a down power scheduled later that day the crew did not set a lower power limit nor did they believe it was required.

San Onofre Procedure SO23-3-2.4,controlling the evolution, "RCS Purification and De-borating Ion Exchanger Operation," Revision 21, provided a guideline to stop the evolution at 10 minutes for the crew to monitor plant response to determine if it was as expected. The crew believed that the blended makeup would be light and plant response was known.

The crew commenced the evolution to boron-saturate the ion exchanger 3ME074. However, the crew did not stop the evolution at 10 minutes because they did not believe it to be a requirement. As a result, the crew over-borated the reactor and caused an unplanned down power of 0.74 percent.

Operation management conducted an investigation of the event and initiated a Nuclear Notification 200721702. The crew members involved in the event were coached about expected performance during reactivity manipulations.

Operations issued a priority 2 notification to the operations department describing the event and management's expectations for reactivity activities. San Onofre Procedure SO23-3-2.4 was revised to place procedure requirements in place to prevent events such as this from occurring again.

Analysis. The failure to have adequate procedural direction to control plant power changes is a performance deficiency. The finding was more than minor because it was associated with the initiating events cornerstone attribute of human performance, and it affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and that challenge critical safety functions during shutdown, as well as during power operations. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that the transient initiator did not contribute to both the likelihood of a reactor trip and to the likelihood that mitigation equipment or functions would not be available. As a result, the issue was of very low safety significance (Green). The finding has a crosscutting aspect in the area of human performance associated with the work practices because licensee supervisory personnel did not ensure activities associated with reactivity control were performed in a controlled manner such that nuclear safety was assured. [H.4(c)]

Enforcement. Technical Specification 5.5.1.1.a requires, in part, that written procedures be established, implemented, and maintained covering the 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. Specifically Regulatory Guide 1.33 section 3.n "Chemical and Volume Control System," shall have instructions for controlling power changes. Contrary to this, as of December 25, 2009, San Onofre Procedure SO23-3-2.4, "RCS Purification and De-borating Ion Exchanger Operation," Revision 21, was inadequate in that it only provided guidelines, not requirements, to control the borating of ion exchangers. As a result, an operations crew performed the evolution and did not adhere to guidelines (because they were not required) and over-borated the reactor, which in turn caused an unplanned down power transient of 0.74 percent. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification 200721702, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: 05000362/2010006-06, "Failure to provide adequate procedure for boron dilution activities."

g. Inadequate Procedures for Radiation Monitoring of Component Cooling Water

Introduction. The inspectors identified a noncited violation of Technical Specification 5.5.1.1.a, "Scope," involving the failure to establish procedures for component cooling water system alignments such that leakage of radionuclides to the environment would be monitored during all operational alignments of component cooling water. Specifically, radiation monitors could be aligned to only one train of component cooling water at a time and the licensees' procedures had no provision for monitoring the second train when both trains were in-service.

Description. On April 5, 2010, inspectors walked down the component cooling water system during which San Onofre Nuclear Generating Station personnel discussed heat exchanger tube leakage in the Unit 2 train B heat exchanger below the operability limit of 18 gallons per minute. The surge tank level was decreasing and component cooling water inventory was being lost to the salt water system. The salt water system is the ultimate heat sink for safety equipment and it operates at a lower pressure than component cooling water. Inspectors reviewed the current operability evaluation contained in Nuclear Notification 200823240 as well as system piping and instrumentation drawings, and learned that radiation monitor 7819 (Unit 2 and Unit 3) can only be aligned to one train of component cooling water at a time. That is because it is connected to the non-critical loop. Noncritical loop loads include the radioactive waste building and containment loads such as control rod drive mechanism cooling and reactor coolant pump cooling. Leakage from the shutdown cooling heat exchangers would be captured by the component cooling water system but the radioactivity may not be measured depending on which train of component cooling water is aligned to radiation monitor 7819.

Inspectors reviewed Final Safety Analysis Report, Section 9.2.2.1 and found that the component cooling water system is designed to be an intermediate barrier between salt water and contaminated heat loads during non-accident scenarios. Final Safety Analysis Report Section 11.5.2.1.3.1 describes radiation monitor 7819 on the non-critical loop: "The component cooling water monitor samples component cooling water from a noncritical component cooling water line that may be isolated from the rest of the component cooling water for certain engineered safety features actuation system conditions. Whenever the noncritical loop of component cooling water is isolated, the system is not monitored and in-leakage to the component cooling water from a higher activity system will not be detected." The Final Safety Analysis Report states that component cooling water is operated at a higher pressure than salt water. This also causes a potential release path.

The alignment of the noncritical loop radiation monitor described in Section 9.2.2.2.1 of the Final Safety Analysis Report was not in accordance with procedures. San Onofre Procedure SO23-2-17, "Component Cooing Water System Operation," Revision 32, Step 6.7 and Attachment 9 Step 6.2 did not direct operators to align the letdown heat exchanger to the component cooling water loop being monitored by radiation monitor 7819 or direct compensatory radiation monitoring by other means. The steps leave this part of system alignment to the discretion of the operator. In addition, plant operators did not question the procedure's adequacy when both trains of component cooling water were in service.

The inspectors concluded that the licensing basis was not correctly implemented with this procedure. San Onofre Procedure AOI SO23-13-7, "Loss of Component

Cooling Water (CCW)/Saltwater Cooling (SWC)," Revision 14 (EC 14-1), Step 13.e, directed operators to check that the trend on radiation monitor 7819 was normal when system leakage is detected. Inspectors found that this was not in accordance with Final Safety Analysis Report, Section 9.2.2.3.2. The steps contained instructions to check the radiation monitor trend but not to ensure that it was aligned to the train that was suspected of leakage.

The inspectors found that San Onofre Nuclear Generating Station did not translate the component cooling water system design into procedures that ensured that radionuclide releases would not occur without monitoring in all operational alignments. Radiation monitor 7819 is not in the Offsite Dose Calculation Manual as a release point. Final Safety Analysis Report Section 11.5.1.2, Effluent Monitoring Systems, does not describe the component cooling water system as a monitored release point or radiation monitor 7819 as an effluent radiation monitor. Plant procedures and sections of the Final Safety Analysis Report support general design criterion 64 for monitoring of radioactive releases.

Inspectors concluded that San Onofre Nuclear Generating Station did not consider the component cooling water system heat exchangers as release paths in several alignments such as shutdown cooling, emergency core cooling system sump recirculation, normal chemical and volume control system letdown, and spent fuel cooling. Plant procedures contained no consideration that component cooling water radiation monitors 7819 (Unit 2 and Unit 3) could only be aligned to one train of component cooling water at a time but that in-leakage could potentially occur in the opposite component cooling water train and be released to the salt water system. Although monthly grab sample monitor component cooling water, this frequency is not sufficient to monitor for radionuclides which could be released into Salt Water Cooling. As a result, existing procedures to monitor component cooling water leakage while at power were inadequate to ensure grab sampling of the component cooling water train not aligned to radiation monitor 7819 (Unit 2 and Unit 3).

The licensee entered this issue into the corrective action program as Nuclear Notification 200871387, and instituted compensatory actions to routinely sample the component cooling water train that is not aligned to the radiation monitor and to perform sampling when the radiation monitor is not in service. These compensatory measures are to remain in place until San Onofre Nuclear Generating Station completes its evaluation of the issue.

<u>Analysis.</u> The failure to translate the design bases into procedures that ensure the radiation monitoring of the safety-related component cooling water system in all operational alignments is a performance deficiency. The inspectors determined that this finding was more than minor because this issue impacted the Public Radiation Protection Cornerstone and its objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Specifically, the component cooling water radiation monitors were not

sufficient to ensure adequate release measurements. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609.04 and determined that the finding screened to Inspection Manual Chapter 0609, Appendix D, Public Radiation Safety Significance Determination Process. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix D, and determined that the finding was of very low safety significance because dose did not exceed Appendix I criteria. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that plant operators did not have a low threshold for identifying deficiencies in procedures. [P.1(c)]

Enforcement. Technical Specification 5.5.1.1.a. requires, in part, that written procedures be established, implemented, and maintained covering the 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; Section 7.g requires procedures for radiation monitoring operation. Contrary to the above, prior to April 22, 2010, the licensee failed to establish procedures for component cooling water system alignments that would prevent unmonitored leakage to the environment through leakage into the Salt Water Cooling system. Because the violation was of very low safety significance and was entered into the corrective action program as Nuclear Notification 200871387, this violation is being treated as noncited violation, consistent with the NRC Enforcement Policy VI.A: NCV 05000361/2010006-07, "Failure to Establish Component Cooling Water Radiation Monitoring Procedures."

h. Failure to Revise Procedures with Known Technical Errors

Introduction. The inspectors identified a cited violation of Technical Specification 5.5.1.1a for the failure to maintain written procedures covered in Regulatory Guide 1.33. Specifically, as of April 2010, the licensee failed to properly control procedure changes associated with plant modifications resulting in procedures with known technical deficiencies being used at the facility.

Description. On April 8, 2010, the inspectors reviewed corrective actions from two previous noncited violations for the licensee's failure to maintain procedures. The first noncited violation was 05000361:05000362/2009003-02 and was associated with the licensee's failure to implement controls over its backlog of procedure change requests such that procedures with known technical deficiencies were in use in the field (before being revised). The second noncited violation was 05000361:05000362/2009009-02 and also involved the licensee's failure to implement controls over its backlog of procedure change requests such that procedures with known technical deficiencies were in use in the field.

During this inspection, the inspectors identified that the backlog of procedure change requests had increased to 3,389. The inspectors identified that most of these procedure changes were appropriately classified according to the "TEAM" method in accordance with San Onofre Procedure SO23-XV-109.1, "Procedure Action Request Committee Process," Revision 1. The inspectors approach classifies procedure changes as technical, enhancement, administrative correction, or modification. Technical changes were defined for plant impacting procedures or procedures that must be issued the next business day as changes that could place a structure system or component in an unevaluated condition; could cause a plant trip; could cause a loss of megawatts; could degrade nuclear safety; could cause unexpected reactivity changes; or could cause an immediate personnel safety issue. However, for procedure changes related to plant modifications the inspectors identified that there was no procedural direction to ensure technical procedure changes were incorporated for operating the equipment following modifications. Additionally, the inspectors identified at least one procedure change request that had been inappropriately classified as a plant modification when it was, in fact, a technical procedure change that was unrelated to a plant modification. (see Section 40A2.5l)

The inspectors requested that the licensee review the backlog of modification-related procedure changes to determine if any were related to modifications that had already been installed in the plant. Of the 212 modification-related procedure changes in the backlog, the licensee identified 60 procedure changes associated with plant modifications that were either installed or partially installed. These 60 pending changes included changes to 10 procedures, including one alarm response procedure, associated with a modification to the instrument air system which had been installed during Unit 2 refueling outage R2C16; these procedures did not reflect the current plant configuration. Following the inspectors' identification of these unincorporated technical changes, the licensee initiated a full review of plant modifications classified procedure change requests. The licensee identified a total of 18 procedures which required technical changes as a result of plant modifications. The licensee agreed that these procedure changes should have been made prior to the associated plant modifications being turned over to operations. The result had been that procedures with known technical deficiencies as a result of plant modifications had been in use in the field.

The inspectors further identified that the process for ensuring modification-related procedure changes were incorporated prior to the modifications being turned over to operations was informal and was not controlled by procedure. The determination of which procedure changes were important and which could be deferred was left up to the procedure writer; there was no procedural guidance for making this determination. This finding was entered into the licensee's corrective action program as Nuclear Notification 200888919, and the licensee took actions to suspend use of the affected procedures until they could be revised.

Analysis. The failure to maintain San Onofre Nuclear Generator Station procedures covered by Regulatory Guide 1.33 is a performance deficiency. The finding is of more than minor significance because, if left uncorrected, it would have the potential to lead to a more significant safety concern by having technically inaccurate procedures being used on important plant systems. Using Inspection Manual Chapter 0609.04, Phase 1 "Initial Screening and Characterization of Findings," the finding was determined to have a very low safety significance because the finding did not result in a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because problems were not thoroughly evaluated such that the resolutions addressed the causes and extents of condition. [P.1(c)]

Enforcement. Technical Specification 5.5.1.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," Appendix A, recommends procedures for the operation of certain plant systems. Contrary to the above, as of April 2009, the licensee failed to maintain written procedures as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Specifically, the licensee did not ensure that following equipment modifications made to the instrument air system, procedures requiring technical changes were suspended, put on administrative hold, or otherwise restricted from use until the required changes were made. As a result, several procedures with known technical deficiencies were available for operator use.

This performance deficiency was previously identified by the NRC on two occasions and were documented as noncited violations 05000361: 05000362/2009003-09 and 05000361; 05000362/2009009-02. The inspectors determined that the licensee had failed to restore compliance within a reasonable time following issuance of these noncited violations. Therefore, this violation is being cited in a Notice of Violation consistent with Section VI.A of the NRC Enforcement Policy: VIO 05000361; 05000362/2010006-08, "Failure to Maintain Written Procedures Covered in Regulatory Guide 1.33."

i. Failure to Set Goals In Accordance With the Maintenance Rule

Introduction. The inspectors identified two examples of a Green noncited violation of 10 CFR 50.65(a)(2) for failure to monitor the performance of auxiliary feedwater system components against established goals in a manner to provide reasonable assurance that the system was capable of fulfilling designated auxiliary feedwater maintenance rule functions.

Description. Under the maintenance rule, San Onofre Nuclear Generating Station defines three separate functions for monitoring the auxiliary feedwater system. Function 1 has a stated purpose for motor-driven Train A to supply feedwater from the condensate feedwater tanks to steam generator 88 for plant cool down when main feedwater is unavailable. Function 2 is identical except that it tracks motor-driven Train B auxiliary feedwater. Function 3 covers the turbine-driven auxiliary feedwater pump to supply both steam generators. All three functions stated that they include the water supply piping and valves from condensate storage Tanks T-120 and T-121. The auxiliary feedwater system has unavailability goals of 1.2 percent per 12 month period for functions 1 and 2 and 1.1 percent per 12 month period for function 3. This equates to approximately 79.3 hours of unavailability for function 3.

On December 9, 2008, San Onofre Nuclear Generating Station performed flawed maintenance that bent the fuse holder contacts such that there was a loose electrical connection. On December 19, 2008, the control room received an annunciator indicating a problem with the Unit 3 turbine driven auxiliary feedwater pump. The licensee identified the loose electrical connection caused the alarm and repaired the connection early on December 20, 2008. In Nuclear Notification 200253911, the licensee counted 9.6 hours of unavailability because that was the time from control room annunciation of a problem to the completion of repairs. The maintenance rule evaluation did not elaborate as to why this amount of time was used. The inspectors noted that the loose connection existed for approximately 10 days prior and for approximately 64 hours while Unit 3 was in Mode 1.

When questioned by the inspectors, San Onofre Nuclear Generating Station personnel stated the basis for using 9.6 hours was that the pump was functional because only its earthquake qualification was in question. Inspectors found that the evaluation of the loose connection did not consider resistance heating of the loose connection or that an earthquake was not required for the failure to be annunciated in the control room. The counting of additional unavailability hours would have caused the Unit 3 turbine driven auxiliary feedwater pump to exceed its 10 CFR 50.65 (a)(2) goal and be placed into (a)(1) status. However, since the pump had been previously placed in (a)(1) status in April 2009 due to functional failures, the approximately 64 hours additional hours of unavailability would have prevented the system from transitioning back to (a)(2) status within 6 months. When questioned, plant personnel informed the inspectors that it utilized NRC performance indicator guidance from NEI 99-02 as its evaluation of unavailability. Combined with other system unavailability, function 3 would exceed its approximately 80 hour unavailability monitoring goal and (a)(1) monitoring would have been significantly extended. Because the licensee performed an inadequate evaluation of unavailability time, the system was returned to (a)(2) status on August 20, 2009.

The second example of inadequate evaluation of unavailability time involved the licensee's maintenance rule evaluation of the failure of auxiliary feedwater condensate isolation valve 2HV5715. On January 26, 2010, valve 2HV5715 failed its in-service stroke test (as described in section 40A2.5c). The hand wheel stem snapped when a leveraging device was used to attempt to open the valve. The valve operator stem was heavily rusted because it had been removed from the preventive maintenance regimen program. This valve must be closed per procedure within 90 minutes of an Operating Basis Earthquake to prevent the loss of water inventory from condensate storage tank T-120 from a line break in the non-seismic portion of the condensate system. Nuclear Notification 200765235 was written to evaluate the broken valve.

Inspectors found that the maintenance rule evaluation counted a functional failure for the valve, but utilized Mitigating Systems Performance Index guidance from NEI 99-02 for unavailability. Inspectors found that use of this guidance was inappropriate to the circumstances and that the evaluation was inadequate. The licensee also utilized Appendix C to NRC Inspection Procedure 71111.13 for evaluating unavailability time, but only considered limited portions of the guidance. The licensee's program procedure described availability but did not provide sufficient guidance for this situation.

The maintenance rule evaluation in Nuclear Notification 200765235 also stated that since the valve failed its stroke test in Mode 6, that there was no unavailability impact. The evaluation stated: "The timing of when this valve would no longer close is unknown and may have been during the required Mode 1 thru 3." Inspectors found that the licensee had not attempted to perform an engineering evaluation to determine when the valve failed due to the rust. Given the as-found condition, the number of unavailability hours was most likely significantly higher than the 79 hour monitoring threshold. This long-standing deficiency was significant because no preventive maintenance had been performed on the valve resulting in its degradation. This deficiency impacted all three maintenance rule functions for the auxiliary feedwater system.

Step 6.5.1.7 of San Onofre Procedure SO123-XV-5.3, "Maintenance Rule Program," required the monitoring of unavailability for these trains. Due to inadequate tracking and accounting, the licensee failed to identify that it exceeded the auxiliary feedwater trains' monitoring goal. San Onofre Procedure SO123-XV-5.3, Step 6.5.1.7, required review of functional unavailability information from all sources as necessary to ascertain performance relative to established criteria. Lastly, San Onofre Procedure SO123-XV-5.3, Step 6.5.1.7, required that when a trend of performance indicates a performance criterion has been exceeded, the train will be evaluated for goal setting. This did not occur. As a result, the plant engineering department took action to evaluate the issues identified by the inspectors and is reviewing existing guidance. These actions were documented in Nuclear Notification 201001922.

Analysis. Failure to adequately account for unavailability time in the licensee's maintenance rule evaluation of the auxiliary feedwater system is a performance deficiency. This finding is more than minor because it affects the equipment performance attribute of the Mitigating Systems Cornerstone per Inspection Manual Chapter 612, Appendix B. Specifically, San Onofre Nuclear Generator Station failed to appropriately account for system unavailability hours which would have resulted in the moving the system to (a)(1), requiring goals and monitoring the performance against those goals for the three auxiliary feedwater functions. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04, Phase 1 "Initial Screening and

Characterization of Findings," and determined that this finding is of very low safety significance, Green. Specifically, the maintenance rule is an administrative activity that could not result in the loss of a system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of the finding was determined to have a crosscutting aspect in the area of human performance in the decision-making component because the licensee did not use a systematic process when faced with the unexpected unavailability for the latent equipment deficiencies. [H.1.a]

Enforcement. Title 10 CFR 50.65 requires, in part, when performance of systems, structures, or components cannot be demonstrated per paragraph (a)(2), that performance goals and corrective action shall be established under paragraph a(1). San Onofre Nuclear Generating Station's monitors auxiliary feedwater maintenance rule functions 1, 2, and 3 with an unavailability goal of approximately 80 hours per rolling 12 month period. Contrary to the above, on January 26, 2009, and December 9, 2008, San Onofre Nuclear Generator Station's auxiliary feedwater maintenance rule functions 1, 2, and 3 exceeded their (a)(2) monitoring goals and San Onofre Nuclear Generator Station failed to evaluate and establish (a)(1) goals. Specifically, the evaluations discounted significant unavailability hours from long maintenance induced failures that would have cause the 80 hour goals to be exceeded. Because this violation was of very low safety significance and was entered into the licensee's corrective action program under Nuclear Notification 201001922, this violation is being treated as a noncited violation in accordance with the NRC Enforcement policy: NCV 05000361/05000362/2010006-09, "Failure to Establish Goals and Monitor for a(1) auxiliary feedwater trains."

j. Failure to Identify and Correct the Use of Degraded Relays in Safety-Related Equipment

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to promptly identify and correct conditions adverse to quality. Specifically, the licensee performed an inadequate extent of condition review and failed to identify that deficient motor driven rotary relays were installed in various safety-related applications.

Description. On August 5, 2007, the Unit 3 emergency diesel generator 3G002 was taken out of service for preventive maintenance. On August 9, 2007, the licensee performed preventive maintenance in the emergency diesel generator cabinet 3L160. The maintenance activity instructs personnel to perform continuity checks for all associated contacts in the electrical cabinet to ensure they are in the correct position, and then perform relay checks to ensure the relays and associated contacts perform as expected when energized or de-energized. During performance of the maintenance activity, maintenance personnel reported (Action Request 070800466) that normally de-energized relay 3L160-2-K52, a

Potter & Brumfield motor driven relay, was sluggish and would not rotate completely. The 2008 problem identification and resolution team documented the deficiency in NCV 05000362/2008012-02, "Failure to Properly Implement Operability Determination Process" because the licensee did not perform an operability determination of the sluggish relay. The licensee entered the issue into the corrective action program as Nuclear Notification 200146292.

The licensee evaluated the motor driven relays in Direct Cause Evaluation 8001654561, and determined that the cause of the failure was an oversized coil manufacturing deficiency. The licensee stated that this was a "well documented failure mechanism for Potter & Brumfield motor driven relays manufactured between 1989 and 1992." The licensee also stated that "there are a large number of normally de-energized motor driven relays in the plant from the manufacturing lots with the oversize coils." The licensee replaced the relays, whose failure could impact the operability of the emergency diesel generators, with new relays that were manufactured with a retaining ring around the coil to prevent oversized coil failures. The licensee generated Nuclear Notification 200188863 to address the extent of condition. However, the extent of condition only focused on the motor driven relays installed in the four emergency diesel generators.

The inspectors asked if the deficient motor driven relays, which remained installed in the plant and were not covered in the scope of the extent of condition review, were installed in safety-related applications. The licensee found 62 normally de-energized relays whose failure "could impact the performance of a specified safety function." The licensee generated Nuclear Notification 200887995 and created maintenance orders to replace the degraded relays at the next available opportunity.

Analysis. The failure to perform an adequate extent of condition evaluation and identify and correct a condition adverse to quality was a performance deficiency. This finding was more than minor because it impacted the equipment performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609.04, Phase 1,"Initial Screening and Characterization of Findings," the inspectors determined the finding to be of very low safety significance (Green) because it did not represent the loss of a system safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with the decision-making component in that the licensee did not use conservative assumptions in making decisions about the extent of condition. [H.1(b)]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, from October 2008, to April 2010, the licensee did not promptly identify and correct the use of deficient motor driven relays in safety-related systems and components. Because the finding is of very low safety significance and has been entered into the corrective action program as Nuclear Notification 200146292, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000361; 05000362/2010006-10, "Failure to Identify and Correct Use of Deficient Motor Driven Relays."

k. Failure to Secure Loose Items in the Switchyard

Introduction: A Green noncited violation of Technical Specification 5.5.1.1.a was identified involving the failure to follow San Onofre Procedure SO123-XX-11, "Switchyard Work Performance" Revision 2. Specifically, the inspectors identified the licensee's failure to adequately control loose material within the switchyard.

Description: On April 7, 2010, inspectors performed a walkdown of the 230kV switchyard. During the walkdown, inspectors identified several pieces of temporary moveable equipment that were not tethered in the switchyard. Inspectors determined that loose material in the switchyard could be hazardous to electrical equipment that could affect the loss of offsite power in the event of seismic activity, tornados, high winds, or hurricanes. The licensee entered a Nuclear Notification 200870138 in their corrective action program to evaluate the condition. The licensee's San Onofre Procedure SO123-XX-11 "Switchyard Work Performance" Revision 2, under Section 6.12, "Temporary Equipment", Step 6.12.1, specifically states, "All unattended temporary movable equipment left in the Switchyard or Relay House SHALL be restrained in such a manner so as to prevent damage to any installed equipment during a seismic event."

The inspectors interviewed plant personnel and determined that personnel failed to remove the materials from the switchyard subsequent to completing assigned work activities in the switchyard. The licensee verified that three of the loose items found by inspectors had been in the switchyard unrestrained since the first week of October 2009, three other items had been unrestrained in the switchyard since March 2, 2010, and two more items had been unrestrained in the switchyard for the life of the plant. The licensee failed to provide effective oversight to ensure the loose material was tied down throughout the duration of work being performed in the switchyard as well as the removal of material following completion of the respective jobs.

The licensee documented this violation in Nuclear Notification 200870138, and its short term corrective actions included removing or securing loose items, evaluating materials in the switchyard for high winds and seismic concerns, and ensuring operator rounds that included checking for loose material.

Analysis. The failure to control loose material near risk-significant equipment is a performance deficiency. This finding is more than minor because it impacts the protection against the external factors attribute of the Initiating Events

Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and power operations. Using the Significance Determination Process Phase 1 worksheets from Inspection Manual Chapter 0609, the inspectors determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This finding also has a human performance crosscutting aspect associated with the work control component in that personnel failed to appropriately plan work activities involving job site conditions which may impact plant structures, systems and components. [H.3(a)]

Enforcement. Technical Specification 5.5.1.1.a, in part, requires that procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33, Appendix A, requires in part, written procedures for Acts of Nature (e.g. tornado, flood, dam failure, earthquakes). Contrary to the above, the licensee failed to follow procedure as required by Regulatory Guide 1.33, Appendix A. Specifically, the licensee failed to adequately control loose material in the switchyard as required by San Onofre Procedure SO123-XX-11, "Switchyard Work Performance," Revision 2. The licensee entered a notification in their corrective action program as Nuclear Notification 200870138. This violation is being treated as a noncited violation, consistent with Section Vela of the Enforcement Policy:
NCV 05000361; 05000362/2010006-11, "Failure to control loose items in the electrical switchyard."

I. Failure to Translate Design Basis Information into Affected Calculations and Procedures

Introduction. The inspectors identified two examples of a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure of the licensee to ensure that new information affecting the plant design bases was incorporated into affected procedures, calculations, and drawings. Specifically, the inspectors identified two instances where the licensee determined, based on a review of information provided by vendors, that design margins or instructions in safety-related calculations or procedures were adversely impacted but failed to revise the calculations or procedures to reflect these non-conservative assumptions.

Description. On April 9, 2009, the licensee initiated Nuclear Notification 200385686 to evaluate Westinghouse Technical Bulletin TB-09-4. This technical bulletin identified that auxiliary feedwater pump heat was not explicitly considered in the sizing calculation for the condensate storage tank; addition of this heat could have an effect of approximately 3000 gallons on the required condensate storage tank volume. On September 25, 2009, the licensee completed its evaluation of this technical bulletin. The licensee concluded that while Technical Bulletin TB-09-4 was applicable to San Onofre Nuclear Generator Station, there was sufficient margin in the existing calculation for the system to perform according to design requirements; no further action was necessary. The affected calculation was not updated. The inspectors verified the licensee's determination that the non-conservatism addressed in the technical bulletin was bounded by other assumptions in the condensate storage tank sizing calculation. However, the inspectors determined that the failure of the licensee to note the neo-conservatism in the calculation could result in the loss of margin should the bounding assumptions be changed in the future. The licensee initiated Nuclear Notification 200886265 to address this deficiency.

On November 5, 2009, the licensee initiated Nuclear Notification 200656309 to evaluate Westinghouse Nuclear Safety Advisory Letter NSAL-09-8. This letter identified the potential for the presence of vapor in emergency core cooling and residual heat removal systems during certain modes of operation. The letter identified the potential that if the residual heat removal system is operated in the shutdown cooling mode above 200°F, initiation of safety injection following a loss of coolant accident could result in the injection water flashing to steam, binding the low pressure safety injection pumps. On January 14, 2010, the licensee completed its evaluation of this nuclear safety advisory letter and determined that while it was applicable to San Onofre Nuclear Generator Station, the concerns noted in the letter had already been addressed in San Onofre Nuclear Generator Station procedures or instructions which contained cautions against operation of shutdown cooling above 200°F. A task was generated under Nuclear Notification 200656309 to modify San Onofre Procedure SO23-5-1.3, "Plant Startup from Cold Shutdown to Hot Standby," Revision 35, to note flashing of injection water as a reason shutdown cooling operation should be secured in Mode 5 prior to entering Mode 4. This task was improperly characterized as an plant modifications or modification-related, procedure and assigned a due date of June 30, 2010. During a review of all plant modification procedure change requests requested by the inspectors, the licensee determined that the plant modification classification was inappropriate and changed it to an "E," or enhancement. The inspectors determined that this procedure change should have properly been classified as a "T," or technical change, and been implemented prior to the next use of the procedure during reactor startup.

On March 26, 2010, during reactor startup following Unit 2 outage R2C16, the licensee was operating in Mode 4 at approximately 270°F while attempting to restore one train of auxiliary feedwater. When this restoration was delayed, the licensee's risk analysis group advised the operators to place shutdown cooling in standby to provide an alternate source of core cooling should the single operable train of auxiliary feedwater be lost. Because the procedure only noted that reactor coolant temperature "should" be maintained below 200°F while shutdown cooling is in operation and did not reference the conclusions drawn from the licensee's analysis of NSAL-09-8, operations personnel failed to recognize the vulnerability of the system to flashing and vapor binding the pump on initiation of low pressure safety injection. Referencing a note contained in the limitations section of the procedure (Attachment 12, Step 15.1) which states, "When shutdown cooling is in-service, then [reactor coolant system] temperature...shall not exceed 340°F...," operations personnel began taking steps to place shutdown cooling in standby with reactor coolant temperature at approximately

270°F. After this course of action was questioned by the NRC resident inspectors and station management, the licensee identified the operating experience evaluation performed under Nuclear Notification 200656309 and determined that placing shutdown cooling in standby at 270°F was inappropriate with current procedures. The licensee initiated Nuclear Notification 200855352 to identify why this course of action was considered.

The licensee's review of NSAL-09-8 under Nuclear Notification 200656309 also identified that while cooling down, San Onofre Procedure SO23-3-2.6, "Shutdown Cooling System Operation," Revision 26, contains procedural steps to isolate the suctions of the low pressure safety injection pumps prior to the initiation of shutdown cooling. However, the inspectors noted that, similar to the startup situation, there is no procedural step or limitation to indicate that these valves must be shut above 200°F to prevent flashing of the fluid should a safety injection signal be received. Further, the limitations and specifications section of the procedure (Attachment 16, Step 1.1) only restricts operation to at or below 340°F. In its evaluation of NSAL-09-8, the licensee did not initiate a procedure change request to address this vulnerability in this procedure. Because procedural restrictions in both San Onofre Procedures SO23-5-1.3 and SO23-3-2.6 permit shutdown cooling operation up to 340°F and because Section 5.4.7 of the Final Safety Analysis Report specifies that shutdown cooling is put into service once reactor coolant system temperature has been reduced below 350°F, the inspectors determined that site procedures and design basis documentation are inadequate to ensure that operators do not place shutdown cooling in service above 200°F.

Analysis. The failure of the licensee to maintain plant design basis specifications, drawings, procedures, and instructions up-to-date is a performance deficiency. The finding is of more than minor significance because it adversely affects the design control attribute of the Mitigating Systems Cornerstone objective. Using Inspection Manual Chapter 0609.04, Phase 1, "Initial Screening and Characterization of Findings," the finding was determined to have a very low safety significance because the finding did not result in a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee failed to implement and institutionalize operating experience information, including vendor recommendations, through changes to plant processes, procedures, equipment, and training programs. [P.2(b)]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that the design basis for safety-related structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this requirement, on June 27, 2009, September 25, 2009, and January 14, 2010, the licensee failed to assure that the design basis for safety-related structures, systems, and components was correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee identified nonconservative errors in calculations and procedures but failed to incorporate this new information into the affected calculations and procedures. Because this finding was of very low safety significance, was not repetitive or willful, and was entered into the corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000361;05000362/2010006-12, "Failure to Maintain Design Basis Information."

m. Failure to Meet Action Plan for Substantive Crosscutting Issues

Introduction. The inspectors identified a Green finding involving examples of the licensee's failure to meet its action plan as described in letters to the NRC documenting actions San Onofre Nuclear Generator Station would take to correct the third and fourth consecutive assessment cycles of substantive crosscutting issues in the areas of human performance and problem identification and resolution.

Description. The NRC's annual assessment letter dated March 4, 2009, was the third cycle where substantive crosscutting issues were identified in human performance and problem identification and resolution. San Onofre Nuclear Generator Station responded to the open substantive crosscutting issues in a letter titled, "Response to Annual Assessment Letter Inspection Report 05000361/2009001, 05000362/2009001," dated April 21, 2009, with the status of corrective actions planned to address the human performance and problem identification and resolution crosscutting issues, including schedules, milestones, and performance monitoring metrics. San Onofre Nuclear Generator Station committed to completing six initiatives to improve its human performance and eight initiatives to improve its process for problem identification and resolution. The licensee committed to completing specific actions to improve performance in these areas.

The status of these commitments was provided to the NRC in an October 30, 2009, letter. As of that date:

- Of the 48 commitments made to improve performance in the human performance area, 28 were complete. Of the 20 remaining open, 4 (20 percent) were past due.
- Of the 36 commitments made to improve performance in the problem identification and resolution area, 21 were complete. Of the 15 remaining open, 3 (20 percent) were past due.

Several of the specific actions to which the licensee committed were not completed by their specified due dates and/or were not completed as specified, as evidenced by the following examples:

i. The licensee committed to establishing response inspectors training and providing this training to selected personnel by December 31, 2009. As of March 31, 2010, this training had not been completed.

ii. The licensee committed that divisions that were not meeting apparent cause evaluation timeliness goals would develop action plans to improve apparent cause evaluation timeliness to less than or equal to 40 days by December 10, 2009. As of March 31, 2010, these divisions had not developed action plans. In a letter dated March 31, 2010, the licensee revised the language of this commitment to reflect actions taken.

iii. The licensee committed to establishing a specific work down curve and/or schedule for backlog of actions requiring closure review boards by February 20, 2010, so that by March 2010, closure review boards were normally completed within 30 days of action completion. As of March 31, 2010, the licensee had failed to establish work down curves or schedules and closure review boards were not being completed in a timely fashion.

On October 29, 2009, after completing an independent safety culture survey in which it noted several areas requiring improvement, the licensee sent another letter to the NRC committing to 56 specific actions to resolve these issues. Five areas requiring action to preserve and improve safety culture were identified. The licensee committed to completing specific actions in each of these areas by specified due dates. Several of the specific actions to which the licensee committed were not completed by their specified due dates and/or were not completed as specified, as evidenced by the following examples:

i. The licensee committed to establishing a specific work down curve and/or schedule for backlog of actions requiring closure review boards and implement by February 20, 2010, so that by March 2010, closure review boards were normally completed within 30 days of action completion. As of March 31, 2010, the licensee had failed to establish work down curves or schedules and closure review boards were not being completed in a timely fashion.

ii. The licensee committed to establishing a project plan and schedule for resolving SAP issues by February 15, 2010, that included: mechanisms for employee input on problems and solutions; definition of end-state desired performance; implementation of improvements; and evaluation of effectiveness. As of March 31, 2010, the licensee had failed to establish a project plan as specified.

On March 25, 2010, the licensee initiated Nuclear Notification 200848923, noting that "a number of the commitments to the NRC made in the October 29, 2009, and October 30, 2009, letters...were completed with inadequate initial quality or were completed late." On March 31, 2010, the licensee submitted a letter to the

NRC modifying some of these commitments. Specifically, the licensee changed the wording for 19 committed actions to reflect the actions taken, which did not align with the actions initially committed, including due date changes for nine past-due commitments for which the licensee changed the due date to a future date. As a result, the licensee failed to satisfy several commitments and due dates made to the NRC related to corrective actions to correct the substantive crosscutting areas. In addition, the licensee modified commitments without first discussing the changes with the Nuclear Regulatory Commission.

Also, the licensee's letters dated April 21, 2009, identified the metrics by which the licensee would assess the state of its corrective action program. The inspectors reviewed the metrics and identified several questions regarding the data the licensee was evaluating for its metrics. Examples included:[1]

- The metric for measuring the time to perform root cause evaluations has been relatively flat over the monitoring period; the metric for measuring the time to perform apparent cause evaluations has exhibited a downward (improving) trend. However, the inspectors found that these metrics are tracked from the assignment date to the "Evaluation Complete Date." As discussed in Nuclear Notification 200886035, the assignment date can be weeks or months after the issue/event was discovered before San Onofre Nuclear Generator Station begins counting time against the metric; and San Onofre Nuclear Generator Station stops counting time against the metric after the divisional corrective action program coordinator review which can be weeks or months before final approval by the corrective action review board. Thus, the data for the time to perform cause evaluations does not reflect the true time it takes the licensee to assign and complete the cause evaluation until the time the corrective actions are identified and approved.

  For example:

  - At the corrective action review board on April 19, 2010, an apparent cause evaluation charter was approved for a notification that was originally written on December 19, 2009. As of April 20, 2010, the apparent cause evaluation had not been assigned; therefore, the clock had not started to track the metric. Thus, the metric for this evaluation did not account for about four months of time.

  - At the corrective action review board on April 19, 2010, an apparent cause evaluation was approved that had been started on November 15, 2009, for a notification generated on November 13, 2009. San Onofre Nuclear Generator Station stopped counting time for purposes of the metric when the

[1] Unless otherwise mentioned, all examples cover metrics tracked from July 2009 through February 2010.

divisional corrective action program coordinator approved the corrective action on January 22, 2010; yet final approval did not actually occur until the corrective action review board on April 19, 2010. Thus, the metric for this evaluation did not account for about three months of time.

- Licensee management had explained to NRC inspectors that their upward trend in the number of nuclear notifications written demonstrates an improvement in the corrective action program in that more people are using it. However, this data only goes back through July 2009. While there was a marked increase in the number of nuclear notifications generated over the first few months of the period, the number has since been constant.[2] The overall increase in nuclear notifications did not account for the expected increase in nuclear notifications from a larger number of personnel on site and the larger workload during the recent extended outage.

- Similarly, licensee management has cited the declining average age of open actions as an indicator of improvement. However, while the average age of corrective actions related to cause evaluation has been trending steadily downward, this appears to be largely due to a concerted effort by the licensee to work off the oldest corrective actions rather than a true overall reduction in the age of corrective actions. Further, this metric does not track the average age of corrective actions to prevent recurrence, which has been trending sharply upward.

- The number of nuclear notifications open has demonstrated a significant upward trend since November 2009. In its April 21, 2009, letter to the NRC, the licensee committed to reducing the number of open nuclear notifications, in part by developing actions to reduce backlog for each division not meeting its divisional metric. On April 14, 2010, the closure review board package related to this commitment was closed, with the statement that the metric had been met for 2009. However, this commitment was modified by the licensee's March 31, 2010, letter which stated that the 2009 goals had been met; that the licensee was now focusing on 2010 goals. The commitment was closed as having been accomplished; however, this metric has been red and trending upward since January 2010.

Analysis. The inspectors determined that the licensee's failure to perform actions as documented in its plan to the NRC was more than minor because if left uncorrected could result in a more significant safety concern. Using Inspection Manual Chapter 0609, Appendix M, this finding was reviewed by NRC management and was determined to be of very low safety significance (Green).

---

[2] Significance level 5 and lower exhibit a slight upward trend over the past three months; significance level 1-4 nuclear notification generation has been trending slightly downward over the same period.

The finding has a crosscutting aspect in the area of human performance associated with the work practices because the licensee did not ensure management oversight of work activities. [H.4(c)]

Enforcement. The finding does not involve an enforcement action because no violation of regulatory requirements was identified. Because the finding does not involve a violation and it has very low safety significance, it is identified as FIN 05000361;05000362/2010006-13, "Failure to meet action plan for substantive crosscutting issues."

4OA6 Meetings

Exit Meeting Summary

On April 23, 2010, the inspectors conducted a briefing of the status of potential findings before concluding the onsite portion of the inspection. This briefing was presented to Mr. R. Ridenoure, Senior Vice President and Chief Nuclear Officer, and other members of the licensee staff. At the conclusion of the inspection on June 17, 2010, the inspectors conducted an exit briefing with Mr. Ridenoure 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.

4OA6 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 VI of the NRC Enforcement Policy, or being dispositioned as NCVs.

Green. The inspectors reviewed a licensee-identified finding involving the failure to follow San Onofre Procedure SO123-I-1.3, "Work Activity Guidelines," Revision 26, Step 6.18.2.2, which allows the licensee to skip a preventive maintenance work order step as long as an evaluation is documented that identifies why it is okay not to perform the step. Specifically, in March 2006, the licensee identified that cubicle for breaker 2A0807 had not been cleaned prior to Cycle 7 due to an energized reserve auxiliary transformer and generated Action Request 060300521 to document the deficiency. The action request further indicated that this issue was applicable to all of San Onofre Nuclear Generator Station 4.16 kV switchgear. All of the switchgears (A03, A04, A05, A06, A07, A08, and A09 for both units) have feeders from both the reserve auxiliary and unit auxiliary transformers (the GDC 17 off-site source of power). The licensee stated that it was not possible to clean every cubicle in a given bus within a single work window. The manufacturer recommended a cleaning frequency of five years of 1000 cycles of operation; however, cubicle for breaker 2A0807 had not been cleaned in over 14 years without an evaluation documenting a basis for postponing the preventive maintenance. Licensee personnel entered this issue into their corrective action program as Nuclear Notifications 200876216 and 200880374.

This finding was more than minor because it impacted the human performance attribute of the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using Inspection Manual Chapter 0609.04, Phase 1, "Initial Screening and Characterization of Findings," the inspectors determined the finding to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This finding was determined not to have a crosscutting aspect because it is a latent condition.

KEY POINTS OF CONTACT

Licensee Personnel

C. Amundson, Maintenance Engineer
V. Barone, Design Engineer
R. Battish, System Engineer
G. Becker, Operations Procedures
S. Chun, Maintenance Engineering Manager
S. Gardner, Electrical Supervisor, System Engineering
J. Jay, Site Procedures Manager
J. Madigan, Health Physics Manager
A. Martinez, Corrective Action Program Manager
A. Matheny, System Engineer
M. McBrearty, Licensing Engineer
C. Mitchell, Operations Procedures
J. Osborne, Project Manager
T. Remick, Engineer, Nuclear Fuel Management
R. Sandstrom, Manager, CAP Project
A. Shean, Nuclear Oversight Manager

NRC personnel

R. Caniano, Director, Division of Reactor Safety
M. Hay, Chief, Technical Support Branch
M. Shannon, Chief, Plant Support Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000361/2010006-01	NCV	Inadequate Operability Determination for Turbine-Driven Auxiliary Feedwater Pump Steam Admission Valves (Section 4OA2.5a)
05000361/2010006-02	NCV	Failure to Translate Design Basis Information for Turbine-Driven Auxiliary Feedwater Pump Steam Admission Valves (Section 4OA2.5b)
05000361/2010006-03	NCV	Lack of Preventive Maintenance Results in Valve Failure and Inoperable Condensate Storage Tank (Section 4OA2.5c)
05000362/2010006-04	NCV	Failure to Report Conditions That Could of Prevented Fulfillment of Safety Function (Section 4OA2.5d)

Opened and Closed

05000362/2010006-05	NCV	Control Room Operators' Failure to Adhere to Conduct of Operations Procedural Requirements (Section 4OA2.5e)
05000361/2010006-06	NCV	Failure to Provide Adequate Procedure for Boron Dilution Activities (Section 4OA2.5f)
05000361/2010006-07 05000362/2010006-07	NCV	Failure to Establish Component Cooling Water Radiation Monitoring Procedures (Section 4OA2.5g)
05000361/2010006-08 05000362/2010006-08	NOV	Failure to Maintain Written Procedures Covered In Regulatory Guide 1.33 (Section 4OA2.5h)
05000361/2010006-09 05000362/2010006-09	NCV	Failure to Establish Goals And Monitor for A(A) Auxiliary Feedwater Trains (Section 4OA2.5i)
05000361/2010006-10 05000362/2010006-10	NCV	Failure to Identify and Correct Use of Deficient Relays (Section 4OA2.5j)
05000361/2010006-11 05000362/2010006-11	NCV	Failure to Secure Loose Items in the Electrical Switchyard (Section 4OA2.5k)
05000361/2010006-12 05000362/2010006-12	NCV	Failure to Maintain Design Basis Information (Section 4OA2.5l)
05000361/2010006-13 05000362/2010006-13	FIN	Failure to Meet Action Plan for Substantive Crosscutting Issues (Section 4OA2.5m)

Discussed

None

LIST OF DOCUMENTS REVIEWED

NUCLEAR NOTIFICATIONS

051001450	200000500	200002210	200002831	200003235
200005532	200005669	200006247	200006366	200006369
200006446	200038227	200047962	200047966	200051692
200052533	200057409	200057494	200057495	200059017
200059581	200060319	200062659	200063244	200080798
200081823	200085457	200095432	200096864	200105838
200112302	200114904	200145364	200146292	200149442
200161642	200166828	200173442	200177549	200177574
200179975	200182897	200184754	200184925	200185228
200187140	200187174	200187386	200188818	200188819
200188863	200189008	200191575	200191643	200191644

NUCLEAR NOTIFICATIONS

200191645	200193463	200194565	200196248	200198876
200199177	200199779	200199803	200200494	200200611
200202392	200202393	200204501	200207687	200209764
200210468	200214923	200216417	200216513	200216785
200217658	200220855	200220901	200224995	200226676
200226851	200229880	200231408	200232002	200237510
200240476	200243930	200244824	200244829	200245222
200249395	200253140	200253911	200253923	200256206
200256262	200258836	200262707	200273137	200281150
200283647	200289984	200301597	200304171	200305694
200309516	200310250	200318226	200319240	200321468
200323460	200323662	200327156	200329766	200337121
200339686	200346192	200347912	200348622	200348676
200350707	200351309	200353559	200353830	200354725
200356209	200357930	200358255	200360012	200362207
200362248	200366460	200375226	200375263	200375271
200375476	200378003	200378783	200383586	200383717
200385686	200385833	200388215	200388299	200389219
200389465	200389602	200391307	200396072	200396074
200396078	200396106	200397538	200402044	200402733
200403327	200403903	200403904	200403907	200403931
200403942	200404016	200407083	200407263	200407581
200408677	200408745	200411720	200413389	200413417
200414063	200414385	200416902	200417206	200420952
200423048	200424908	200425771	200427466	200427700
200439005	200442871	200445728	200449046	200450694
200453351	200454549	200454708	200454875	200456738
200457151	200458808	200461737	200462842	200463613
200469510	200476904	200481911	200493704	200495283
200496192	200498067	200498776	200501125	200505402
200507991	200509834	200511477	200514597	200518579
200545007	200545500	200550606	200550985	200553431
200554449	200554503	200554762	200556120	200559128
200564587	200569111	200572704	200581670	200585309
200591743	200596242	200596804	200599691	200599743
200600926	200604461	200607694	200611851	200613666
200613716	200614081	200625389	200628825	200631222
200631367	200635119	200636471	200636549	200638562
200638824	200640096	200647126	200656309	200657895
200663614	200663620	200663692	200664434	200666345
200666345	200666778	200667666	200668488	200670338
200683591	200684138	200685073	200688490	200688648
200689102	200689282	200689526	200689551	200689650
200690408	200690878	200690900	200690971	200691209
200691226	200691370	200691516	200692319	200692334
200692335	200692347	200692815	200692819	200694409

NUCLEAR NOTIFICATIONS

200698869	200699499	200703718	200703793	200704636
200704875	200710313	200711245	200711324	200711339
200711991	200712412	200715724	200718801	200721702
200722117	200727789	200728270	200728441	200737719
200743785	200743785	200745033	200746950	200752137
200760309	200761459	200769308	200778595	200778598
200780929	200781022	200791845	200792682	200801929
200803364	200804931	200805827	200809842	200814132
200832315	200834923	200835619	200836042	200841643
200847163	200848923	200853352	200858260	200866485
200866488	200866490	200867104	200870138	200871526
200871527	200874078	200876130	200876216	200877698
200877796	200877799	200877834	200880374	200882433
200886035	200887746	200887995	200888919	

ORDERS

800011270	800049251	800073513	800073728	800076896
800076907	800081649	800164561	800183273	800185541
800192268	800216674	800216676	800216677	800216678
800269843	800275473	800289258	800314547	800354225

CONDITION REPORTS/OTHER

AR 020201440	AR 020201440	AR 020201440	AR 020801305	AR 020801305
AR 020801305	AR 020801305	AR 030100348	AR 030100348	AR 030100348
AR 030100348	AR 030401460	AR 030401460	AR 030401460	AR 041200133
AR 041200133	AR 041200133	AR 050401537	AR 050401537	AR 050401537
AR 060300521	AR 060300521	AR 060300521	AR 060301666	AR 060301666
AR 060301666	AR 061200817	AR 061200817	AR 061200817	AR 070500851
AR 070500851	AR 070500851	AR 070700345	AR 070700345	AR 070700345
AR 070700366	AR 070700366	AR 070700366	AR 070800283	AR 070800283
AR 070800283	AR 070800284	AR 070800284	AR 070800284	AR 070800285
AR 070800285	AR 070800285	AR 070800286	AR 070800286	AR 070800286
AR 070800287	AR 070800287	AR 070800287	AR 070800288	AR 070800288
AR 070800288	AR 070800289	AR 070800289	AR 070800289	AR 070800993
AR 070800993	AR 070800993	AR 071000901	AR 071000901	AR 071000901
AR 071200416	AR 071200416	AR 071200416	AR 071201393	AR 071201393
AR 071201393	AR 071201417	AR 071201417	AR 071201417	AR 080101417
AR 080101417	AR 080101417	AR 080200546	AR 080200546	AR 080200546
AR 080300666	AR 080300666	AR 080300666	AR 080301122	AR 080301122
AR 080301122	AR 080301404	AR 080301404	AR 080301404	AR 080400545
AR 080400545	AR 080400545	AR 080401137	AR 080401137	AR 080401137
AR 080401137	AR 080401144	AR 080401144	AR 080401144	AR 080401144
AR 080500972	AR 080500972	AR 080500972	AR 080600104	AR 080600104
AR 080600104	AR 080600212	AR 080600212	AR 080600212	RCE 93-004

ENGINEERING DOCUMENTS

NECP 800071431
NECP 800071869
NECP 800129634
NECP 800071494
NECP 800074314
NECP 800071495
NECP 800074316
NECP 800071764
NECP 800074486

MAINTENANCE ORDER

06101428 0412153600 05101896000

PROCEDURES

NUMBER	TITLE	REVISION / DATE
	Replacement of Foxboro CVCS Boric Acid Makeup System Controls with Ovation Distributed Control System (DCS)	0
	Shutdown Nuclear Safety	24
2-10-010	Operating Instruction Attachment 7 Boron Saturating 2(3)ME-074, CVCS Ion Exchanger	January 26, 2010
A610	Operation of Manual (Gearbox) Butterfly Valves Attachment 29	21
LCS 3.3.108	Vibration and Loose-Parts Monitoring System	
M-0050-017	BTB RSB 5-1 Condensate Inventory Calculation	4
N/A	SONGS System Health Report, 4KVS	4th Quarter, 2009
SCES-004-08	Corrective Action Program Audit	May 16, 2008
SCES-012-09	Equipment Reliability Audit	October 17, 2009
SCES-014-09	Corrective Action and Self-Assessment Program Audit	March 5, 2010
SD-SO23-110	220 kV Switchyard System	19
SD-SO23-120	6.9 kV, 4.16 kV, and 480 V Electrical Distribution Systems	19
SO123-0-A6	Operations Division Procedure (Precautions)	8
SO123-I-1.28.1	Electric Distribution Grounding Guide	4

PROCEDURES

NUMBER	TITLE	REVISION / DATE
SO123-0-A6	Operations Division Procedure (Precautions)	8
SO123-I-1.28.1	Electric Distribution Grounding Guide	4
SO123-I-1.3	Work Activity Guidelines	26
SO123-I-1.3	Work Activity Guidelines	26
SO123-I-1.34	Scaffolding Erection	27
SO123-I-4.13	Megger Testing	6
SO123-I-9.9	Square "D" and Westinghouse Type DS Circuit Breakers Inspection and Testing	4
SO123-II-9.48	Magnetrol and Other Miscellaneous Liquid Level Switches Calibration	6
SO123-MA-1	Maintenance and Construction Services Division	7
SO123-MA-1	Maintenance and Construction Services Division	7
SO123-OR-1	Operating Experience Program	9
SO123-RX-1	Reactivity Management Program	4
SO123-VI-1	Review/Approval Process for Orders, Procedures, and Instructions	28
SO123-XV-1.20	Seismic Controls	0
SO123-XV-109	Procedure and Instruction Format and Content	1
SO123-XV-109.1	Procedure Action Request Committee (PARC) Process	1
SO123-XV-3.3	NRC Reporting Requirements and Assessments	15 EC 15-1
SO123-XV-303	Closure Review Process	0
SO123-XV-50	Corrective Action Program	15
SO123-XV-50	Corrective Action Program	16
SO123-XV-52	Functionality Assessments and Operability Determinations	14

PROCEDURES

NUMBER	TITLE	REVISION / DATE
SO123-XV-91	Reactivity Management Implementation	4
SO123-XV-HU-1	Human Performance Program	6
SO123-XV-HU-4	Human Performance Roles and Responsibilities	1
SO123-XX-11	Switchyard Work Performance	2
SO123-XX-11	Switchyard Work Performance	2
SO123-XX-6	Operator Work Around Program	7
SO123-XXIV-5.1	Engineering & Technical Services Software Quality Assurance	6
SO123-XXX-3.5	Evaluation and Reporting of Problems to the NRC Pursuant to 10 CFR Part 21	3
SO123-XXX-3.5	Evaluation and Reporting of Problems to the NRC Pursuant to 10 CFR Part 21	3
SO123-XXXVI-1	Nuclear Fuel Management (NFM) Quality Program	6
SO23-10-9	Turbine Lube Oil System Operation	19
SO23-13-8	Severe Weather	8
SO23-13-8	Severe Weather	8
SO23-15-53.A	CIRC Water Box Cathodic Protection Sys Trouble	20
SO23-15-53B	Condensate Pump P050 Flow Lo	18
SO23-15-63.D	Annunciator Panel 63D, Switchyard/Penetration Switchgear	12
SO23-15-63.E	Annunciator Panel 63E, Switchyard	8
SO23-3-2.4	Operating Instruction Attachment 7 Boron Saturating 2(3)ME-074, CVCS Ion Exchanger	21
SO23-3-2.4	Operating Instruction Attachment 7 Boron Saturating 2(3)ME-074, CVCS Ion Exchanger	22
SO23-3-2.6	Shutdown Cooling System Operation	26

PROCEDURES

NUMBER	TITLE	REVISION / DATE
SO23-3-3.30.4	Main Steam System Online Valve Test	12
SO23-3-3.8	Safety Injection Monthly Tests	25
SO23-5-1.3	Plant Startup from Cold Shutdown to Hot Standby	35
SO23-6-25	Generator Stator Cooling Water System Operation	23
SO23-6-30	230kV Switchyard Rounds and Inspections	26
SO23-6-30	Switchyard Inspection and Operation	26
SO23-6-30	Switchyard Inspection and Operation	27
SO23-6-5	Main and Auxiliary Transformer Operation	20
SO23-6-6	Reserve Auxiliary Transformer Operation	15
SO23-IV-6.3.2	Security Intrusion Detection System Probability Testing	7
SO23-V-16	Emergency Core Cooling System (ECCS) Piping Gas Void Calculation	0
SO23-V-2.14	Thermal Inspection of Plant Components	9
SO23-V-2.14	Thermal Inspection of Plant Components	9
SO23-V-4.40	Electrical Equipment Monitoring Program	4
SO23-V-4.40	Electrical Equipment Monitoring Program	3 TCN 3-1
SO23-V-4.40	Electrical Equipment Monitoring Program	3
SO23-XV-2	Troubleshooting Plant Equipment and Systems	6
SO23-XV-50.CAP-1	Writing Nuclear Notification for Problem Identification and Resolution	3
SO23-XV-50.CAP-2	SONGS Nuclear Notification Screening	5
SO23-XV-50.CAP-4	Implementing Corrective Actions	3
SO23-XV-85	Boric Acid Corrosion Control Program (BACCP) Top Risk Significant Systems	5

PROCEDURES

NUMBER	TITLE	REVISION / DATE
SO23-XX-10	Maintenance Rule Risk Management Program	3
SO23-XX-10	Maintenance Rule Risk Management Program Implementation	3
SO23-XX-28	On-Line Work Management Process	3
SO23-XX-29.1	Seasonal Readiness	0
SO23-XX-30	Nuclear Maintenance Order (NMO) Generation, Screening, and Classification	1
SO23-XX-30	Nuclear Maintenance Order (NMO) Generation, Screening and Classification	2
SO23-XX-8	Integrated Risk Management	5
SO23-XXXVI-1.4	Documentation of Reload Fuel Cycle Analysis	6
SOS-025-09	Surveillance Report: Corrective Action Program Implementation	May 26, 2009
SOS-040-09	Surveillance Report: Station Integrated Business Plan Closure Review Process	October 09, 2009
SY-SO23-G-2	Systems Engineering Handbook	4
SY-SO23-G-2	Systems Engineering Handbook	3
SY-SO23-G-2	Walkdown Standard-Stainless Steel Schedule 10S Pipe	4
TM-2791A	SONGS Air Management From RWST and CES	July 2008

Licensee Meetings Attended

Production/Ops Focus Meeting (2)
Closure Review Board (CRB) (3)
Corrective Action Review Board (CARB) (2)
SONGS Switchyard Oversight Committee (SSOC) (1)

OTHER MISCELLANEOUS DOCUMENTS

TITLE	DATE
Leadership Engagement Trending System Engagement Summary Report	November 2009 to March 2010
Unit 2/3 Operations Leadership Observation RAA-Plant Monitor Reactivity Affecting Activity LOP 14	1st Quarter, 2010
Fundamental LOP 14 RAA-Plant Monitor Reactivity Affecting Activity From 10/01/2009-10/31/2009	April 6, 2010
Monthly Meeting Reactivity Oversight Group (ROG)	March 30, 2010
ROG Report-Meeting 03/30/2010	March 30, 2010
List of Operator Workarounds	July 2009 to November 2009
List of Operator Burdens	December 2009 to January 2010
List of Control Room Issues	July 2009 to January 2010
List of Temp Mods	October 2009 to March 2010
List of Control Room Deficiencies	April 2009 to February 2010
SONGS Operational Focus Index	April 6, 2010
Operations Division Corrective Action Burndown Plan	November 6, 2009
Leadership Engagement Trending System 10.14 RAA-PLANT Monitor Reactivity Affecting Activity (LOP 14)	February 2010
Impaired Alarm Record Unit 2	
Priority 1 HPSI Pump Control Circuits (Implementation of ECP that corrects problem)	June 5, 2009
Top Ten Equipment Issues at SONGS	
Leadership Engagement Trending System 10.19 Verification Practices (LOP 19)	1st Quarter, 2010

OTHER MISCELLANEOUS DOCUMENTS

TITLE	DATE
Site Plan Status Control Misposition Events	
Mission Times for Operability Determinations	Rev A February 24, 2010
Equipment Leaks Summary	September 2009 to March 2010
Gas Void Trend in 8" LPSI Header (U2 Loop 2A)	March 2009 to September 2009

Root Cause Directed Maintenance Evaluation dated September 1995 "RCDM 95-02 "Design Life of Normally Energized Agastat E7000/7000 Series Time Delay Relays" IAW NATS No. 9509010". (This study found that the relays have a design life much longer than the 10 years specified by the manufacturer and that the actual design life was greater than 40 years.)

CALCULATIONS

NUMBER	SUBJECT
M-0013-005	Safety Injection Tank Fluid Nitrogen Evolution

ACTION REQUESTS

050100895	050101779	050200370	050200417	050201676
050300921	050600035	050600474	050800143	050800896
051101380	051200838	051200895	060100358	060100673
060101480	060101481	060200732	060300521	060401277
060501595	060600040	060700177	060700430	060701103
060701128	060800908	060800909	060800962	060900185
060900824	060900981	061000969	061001517	061001528
061100693	061101379	061101448	061101460	061101464
061101478	061101632	070300300	070300991	070301057
070500395	070500760	070801108	071100540	071100697
071101406	071101426	071101427	071101428	071101429
071101431	071101432	071200061	071200215	071200331
071200546	071200614	071200621	071200927	071201038
071201039	071201169	071201203	071201205	071201245
071201468	071201558	071201814	080100008	080100688
080100844	080200815	080200903	080201125	080300231
080300994	080400273	080400955	080500248	080500381
080501351	080600108	080600397		

DRAWINGS

NUMBER	TITLE	REVISION
30342	Elementary Diagram Diesel Generator 2G002 Control DC System	11
30344	Elementary Diagram Diesel Generator 2G002 Excitation	14