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

December 8, 2006

EA-06-245

Richard M. Rosenblum
Senior Vice President and
Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
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SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000361;362/2006009

Dear Mr. Rosenblum:

On November 30, 2006, the US Nuclear Regulatory Commission (NRC) completed a component design bases inspection at your San Onofre Nuclear Generating Station. The enclosed report documents our inspection findings. The preliminary findings were discussed on July 20, 2006, with Mr. B. Katz and other members of your staff. After additional in-office inspection, a final telephonic exit meeting was conducted on November 30, 2006, with Mr. B. Katz and others of your staff.

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

Based on the results of this inspection, the NRC has identified five findings that were evaluated under the risk significance determination process. Violations were associated with all of the findings. All five of the findings were found to have very low safety significance (Green) and the violations associated with these findings are being treated as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. In addition, a licensee identified violation which was determined to be of very low safety significance is described in the report. If you contest any of the noncited violations, or the significance of the violations you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the San Onofre Nuclear Generating Station.

In accordance with Code of Federal Regulations, Title 10, Part 2.390 of the NRC's Rules of Practice, a copy of this letter and its enclosure 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 http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jeffrey A. Clark, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000361;362/2006009
w/Attachments: 1 - Supplemental Information
2 - Final Determination of Significance, Foreign Material in Condensate Storage Tank Enclosure

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From: Troy Pruett
To: GDR@nrc.gov
Date: 12/06/2006 9:45:33 PM
Subject: Re: REPORT

I concur on the report. This superseeds my earlier nonconcurrence.

Troy Pruett
817-860-8173
>>> George Replogle 12/06/06 4:40 PM >>>
Here's the report. I made the requested changes.

George

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-361, 50-362
License:	NPF-10, NPF-15
Report Nos.:	05000361/2006009 and 05000362/2006009
Licensee:	Southern California Edison Company (SCE)
Facility:	San Onofre Nuclear Generating Station, Units 2 and 3
Location:	5000 S. Pacific Coast Hwy. San Clemente, California
Dates:	June 26 through November 30, 2006
Team Leader:	G. Replogle, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	J. Drake, Operations Examiner, Operations Branch J. Reynoso, Reactor Inspector, Engineering Branch 1 M. Sitek, Resident Inspector, San Onofre Nuclear Generating Station
Accompanying Personnel:	L. Ellershaw, PE, Consultant G. Skinner, Electrical Engineer, Beckman and Associates W. Sherbin, Mechanical Engineer, Beckman and Associates
Others:	D. P. Loveless, Senior Reactor Analyst D. G. Passehl, Senior Reactor Analyst
Approved By:	J. Clark, PE, Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000361;362/2006009; June 26 through November 30, 2006; San Onofre Nuclear Generating Station: baseline inspection, NRC Inspection Procedure 71111.21, *Component Design Basis Inspection*.

The report covers an announced inspection by a team of three regional inspectors, three contractors and one resident inspector. Five findings were identified. All of the findings were of very low safety significance. The final significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 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 3, dated July 2000.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems; Barrier Integrity

- Green. The team identified a noncited violation of Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion V, *Procedures*, for the failure to follow procedural requirements and establish the Units 2 and 3 CST-120 condensate storage tank enclosures as foreign material exclusion areas. The team found several pieces of foreign material in each enclosure. Foreign materials in these areas could have caused auxiliary feedwater system operational problems following a seismic event. In addition, the licensee failed to properly address industry operating experience related to foreign materials in auxiliary feedwater system water sources. Finally, a related condensate storage tank sizing calculation failed to consider the potential for reactor vessel head void formation during the cooldown to shutdown cooling conditions. The licensee captured this finding in their corrective action program as Action Requests 060700471 and 0601000172.

  The failure to follow plant procedures was a performance deficiency. This finding is more than minor because it affected the mitigating system cornerstone objective (equipment performance attribute) of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The team determined that a Phase 3 significance determination was required because the finding screened as potentially risk significant due to a seismic initiating event. Region IV senior risk analysts performed a Phase 3 significance determination and determined that the issue represents a finding of very low safety significance (Section 1R21.b.1).

- Green. The team identified a noncited violation of Technical Specification 5.5.1.a for an inadequate emergency diesel generator ground fault alarm response procedure. Specifically, the procedure had operators check for grounds associated with the emergency diesel generator itself but did not specify actions to address the more likely ground locations, which included components on the 4.16kV bus. Since other plant procedures permit cross-tying the safety-related buses on the opposite unit in the event of a loss of an emergency diesel generator, the failure to properly consider grounds in other locations could result in additional equipment failures. The licensee captured this finding in their corrective action program as Action Request 060700753.

  The failure to provide an adequate alarm response procedure was a performance deficiency. This issue was more than minor because the procedure deficiency affected the mitigating system cornerstone objective (procedure quality attribute) of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Specifically, under certain circumstances, the emergency diesel generators may not have functioned following a seismic event. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the issue screened as having very low safety significance because the finding was not a design or qualification deficiency, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events (Section 1R21.b.2).

- Green. The team identified a noncited violation of Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion III, *Design Control*, for the failure to properly calculate control circuit voltages associated with the Unit 3 motor-driven auxiliary feedwater Pump 3P504 breaker. Correcting the error used approximately 1/3 of the available design margin. The licensee captured this finding in their corrective action program as Action Request 060700765.

  The failure to properly implement proper design controls was a performance deficiency. This issue was more than minor because, if left uncorrected, it could become a more significant safety concern. Specifically, the noted calculations are used for operability determinations and plant modifications. Uncorrected errors could mask equipment operability issues. This issue was similar to non-minor violation Example 3.j in NRC Inspection Manual Chapter 0612, Appendix E, *Examples of Minor Issues*, because there was a reasonable doubt of the operability of the pump breaker. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the issue screened as having very low safety significance because it was a design deficiency confirmed not to result in loss-of-operability in accordance with NRC Manual Chapter Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment (Section 1R21.b.3).

- Green. The team identified a noncited violation of the Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion XVI, *Corrective Actions*, for the failure to promptly identify a condition adverse to quality (trapped air in the safety injection suction lines). Each suction line contained approximately 11.5 cubic feet of trapped air, but the licensee's official design calculations assumed the lines were full of water. Additionally, industry operating experience notified the licensee that air in the safety injection system suction lines could cause operational problems (a condition adverse to quality) but the licensee failed to promptly identify the condition at San Onofre Nuclear Generating Station. The licensee's engineering evaluation erroneously determined that San Onofre Nuclear Generating Station was not vulnerable to the condition identified in the operating experience. The licensee captured this finding in their corrective action program as Action Request 060700747.

  The failure to promptly identify and correct a condition adverse to quality in response to applicable operating experience was a performance deficiency. This finding was more than minor because it affected the mitigating system cornerstone objective (equipment performance attribute) to ensure the reliability and capability of equipment needed to respond to initiating events. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the finding was of very low safety significance because it was a design deficiency confirmed not to result in loss-of-operability in accordance with NRC Manual Chapter Part 9900, Technical Guidance*, Operability Determination Process for Operability and Functional Assessment*. This finding has a cross-cutting aspect in the area of problem identification and resolution, in that the licensee failed to thoroughly evaluate applicable industry operating experience concerning air voids in recirculation piping suction lines (Section 1R21.b.4).

- Green. The team identified a Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion XVI, *Corrective Actions*, violation for the failure to promptly identify a condition adverse to quality (Train A emergency diesel generators lost seismic qualification). The licensee had identified that a ground fault on a nonsafety-related uninterruptible power supply could cause the emergency diesel generator to trip during a fire but failed to further determine that the same scenario could occur during a seismic event. The licensee captured this finding in their corrective action program as Action Request 060600500.

  The failure to promptly identify a condition adverse to quality was a performance deficiency. This finding is more than minor because it affected the mitigating system cornerstone objective (equipment performance attribute) of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, Train A emergency diesel generator operability was not assured for seismic events. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the internal events portion of the worksheet did not apply, because the finding only involved an external seismic event with a loss of offsite power. Additionally, for external events, the finding screened as have very low safety significance because it did not involve the loss or degradation of equipment or function specifically designed to mitigate an external event

(e.g., seismic snubbers, flooding barriers, tornado doors) and the safety function was not considered completely failed or unavailable, as the Train B emergency diesel generators were unaffected by the issue. This finding has a cross-cutting aspect in the area of problem identification and resolution, in that engineers failed to perform an appropriate extent of condition review and promptly identify the nonconforming emergency diesel generators, a condition adverse to quality (Section 1R21.b.5).

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 the applicable corrective actions are listed in Section 4OA7.

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

1R21 Component Design Bases Inspection (71111.21)

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than two or a Birnbaum value greater than 1E-6.

a. Inspection Scope

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components was consistent with the design bases and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For the review of operator actions, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 15-20 risk-significant and low design margin components, 3 to 5 relatively high-risk operator actions, and 4 to 6 operating experience issues. The sample selection for this inspection was 20 components, 5 operator actions, and 5 operating experience items.

The components selected for review were:

- Unit 2, Battery 2D1, 125Vdc safety-related control power battery
- Unit 2, (Center) component cooling water pump
- Unit 2, turbine driven auxiliary feedwater Pump P-140
- Unit 2, motor-driven auxiliary feedwater Pump P-141
- Unit 2, 4kV safety-related Bus 20A4
- Unit 2, Valve 2HV6212, Train A air-operated component cooling water system discharge valve to the non-critical loop
- Unit 2, Valve 2HV6497, Train A motor-operated saltwater cooling discharge valve
- Unit 2, Valve S21305MU448, auxiliary feedwater check valve to Steam Generator 2E088
- Unit 3, condensate storage Tank CST-120
- Unit 3, Train A, recirculation actuation signal circuitry
- Unit 3, Train A, component cooling water heat exchanger
- Unit 3, Train A, high pressure safety injection pump
- Unit 3, Train A, Valve 3HV6500, air operated shutdown cooling heat exchanger discharge valve
- Unit 3, Train B, emergency diesel generator
- Unit 3, Train B, charging pump
- Unit 3, auxiliary feedwater Pump P-141 throttle valve
- Unit 3, under-voltage relays
- Unit 3, Valve 3HV6500, Train A air operated component cooling water shutdown cooling heat exchanger discharge valve
- Unit 3, Valve 3HV9330, Train A motor-operated high pressure safety injection system valve to reactor coolant system Loop 2A
- Unit 3, Valve 3HV9304, Train B, motor-operated safety injection suction valve, inboard containment isolation valve

The selected operator actions were:

- Provide makeup to condensate storage Tank CST-121 from demineralizer water tanks.

- Provide makeup to condensate storage Tank CST-121 from condensate storage Tank CST-120.

- Isolate the faulted steam generator and cooldown the reactor coolant system.

- Locally control auxiliary feedwater flow control valves upon a loss-of-auxiliary feedwater.

- Depressurize steam generators and align the condensate system for cooling.

The operating experience issues were:

- NRC Information Notice 2004-01, *Auxiliary Feedwater Pump Recirculation Line Orifice Fouling - Potential Common Cause Failure*, January 21, 2004

- Nuclear Safety Advisory Letter NSAL-04-7, *Containment Sump Line Fluid Inventory*, dated November 15, 2004

- NRC Bulletin 88-04, *Potential Safety-Related Pump Loss*, May 5, 1988

- NRC Information Notice 97-07, *Problems Identified During Generic Letter 89-10 Closeout Inspections*, June 26, 1997

- NRC Information Notice 2001-19, *Improper Maintenance and Reassembly of Automatic Oil Bubblers*, August 12, 2002

b. Findings

b.1 Failure to Follow Foreign Material Exclusion Controls

Introduction. The team identified a Green noncited violation of the Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion V, *Procedures*, for the failure to follow procedural requirements and establish Units 2 and 3 CST-120 condensate storage tank enclosures as foreign material exclusion areas. Foreign materials in these areas could cause auxiliary feedwater system operational problems following a seismic event. In addition, the licensee failed to properly address industry operating experience related to foreign materials in auxiliary feedwater system water sources. Finally, a related condensate storage tank sizing calculation failed to address the potential for the formation of reactor vessel head voids during the cooldown to shutdown cooling conditions.
Description. Each unit has two safety-related condensate storage tanks. The primary tanks (CST-121 on each unit) contain at least 144,000 gallons while the secondary tanks (CST-120) contain at least 360,000 gallons. The water is dedicated for use by the auxiliary feedwater system and the steam generators for accident mitigation. The two tanks combined provide sufficient water to mitigate design basis accidents and events and allow operation under hot standby conditions for up to 24 hours following a plant trip.

The team noted that the CST-120 condensate storage tanks were not seismically qualified. Instead, a seismically qualified concrete enclosure surrounds each tank. The enclosures are open at the top. Following a seismic event, water from the CST-120 tanks is assumed to flow into the enclosures. Operators can gravity feed

water from the enclosures to the seismically qualified CST-121 condensate storage tanks. The team noted that any foreign materials left in the enclosures had the potential of being entrained in the auxiliary feedwater supply.

During plant walkdowns on July 11 and 12, 2006, the team identified several foreign material objects in the condensate storage Tank CST-120 enclosures for both units, as described below:

<u>Unit 2</u>

- Two approximately 15 by 18 inch plastic signs tie wrapped to metal posts
- Two cloth rags
- Several loose nylon tie wraps
- A small loose plastic bag

<u>Unit 3</u>

- Two approximately 15 by 18 inch plastic signs tie wrapped to metal posts
- A loose nylon tie wrap
- A small loose plastic bag

The team determined that the licensee had not followed procedures related to foreign material controls. Specifically, Procedure SO123-FO-1, *Site Foreign Material Exclusion Control Program*, Revision 3, states, in part:

> Foreign Material Exclusion Program provisions are to be applied to important-to-safety plant systems and components when exposed during . . . <u>operation</u> . . . Controls for foreign material exclusion SHALL be implemented to prevent the introduction of foreign material into mechanical or electrical equipment, components, or systems . . . foreign material exclusion controls SHALL provide for positive physical control and accountability of material, equipment, and tools <u>to prevent their introduction into mechanical</u> . . . <u>components or systems</u>. . . .

Contrary to the above, prior to July 12, 2006, no foreign material controls were implemented for the condensate storage Tank CST-120 enclosures for Units 2 and 3. The failure to provide controls would allow foreign materials to be left in the safety-related condensate storage tank enclosures and potentially block flow to and/or enter the auxiliary feedwater systems.

The team had two safety concerns:

- The plastic signs were of sufficient size to block the sump openings, potentially restricting the flow of water to the auxiliary feedwater pumps. The sump openings were approximately 19 inches square.

- The auxiliary feedwater pump minimum flow lines have very small orifices (11/64 inch diameter). Some of the material could get into these lines and block one or more of the openings. Pump operability could be challenged during periods when the discharge to the steam generators is isolated or the pumps are running at shutoff head.

In addition to the above, the team identified that the licensee had not adequately evaluated operating experience associated with these types of issues. NRC Information Notice 2004-01, *Auxiliary Feedwater Pump Recirculation Line Orifice Fouling - Potential Common Cause Failure*, dated January 21, 2004, provided operating experience information to licensees, including San Onofre Nuclear Generating Station, regarding the potential for foreign materials to foul auxiliary feedwater minimum flow line orifices. Although the licensee utilized minimum flow line orifices similar (small holes, but not exactly the same size) as those specified in the information notice, the licensee's evaluation (Action Request 040101393-1, dated March 23, 2004) of Information Notice 2004-01 inappropriately determined that the notice did not apply to San Onofre Nuclear Generating Station. The licensee reasoned that, because they used pure demineralized water in their condensate storage tanks, no foreign material could enter the system and foul the orifices. The licensee failed to properly consider all credited operating conditions, including those where water from the Tank CST-120 enclosures could be used for accident mitigation. Following a seismic event, foreign material in the San Onofre Nuclear Generating Station CST-120 enclosures could be transported to the auxiliary feedwater system and potentially foul the auxiliary feedwater minimum flow line orifices.

In response to the inspectors concerns, the licensee provided an evaluation to the NRC on August 27, 2006 that specified, in part, that Tank CST-120 was not necessary for accident mitigation because Tank CST-121 provided all the water necessary to respond to design basis accidents. In other words, the identification of foreign materials in the Tank CST-120 enclosure was not significant because operators could cool down the plant to shutdown cooling entry conditions without relying on the water in Tank CST-120. The licensee based this assertion on the Updated Final Safety Analysis Report, Section 9.2.6.1, which states, in part:

> The Seismic Category I Condensate Storage Tank CST-121 has sufficient storage capacity to maintain a hot standby condition for 2 hours, and to provide enough water to remove decay heat and to cooldown the reactor to 400° F, the temperature at which the shutdown cooling system can be used to remove decay heat.

Further, the licensee noted that Calculation M-0050-018, *Evaluation of T-121 Requirements*, dated February 4, 2000, concluded that Tank CST-121 could supply all the necessary water demands for design basis accidents, considering a 2-hour period of hot standby conditions followed by an immediate cooldown to 400° F. The calculation assumed a 75 °F per hour reactor coolant system cooldown rate.

The team identified that Calculation M-0050-018 did not demonstrate that Tank CST-121 could provide all necessary water following a design basis earthquake. Specifically, the calculation did not address reactor vessel head steam voids that will occur (under natural circulation conditions) if operators cool down and depressurize the reactor vessel at the specified cooldown rate. While, post accident, it was possible to initially cooldown at the 75° F per hour cooldown rate, initiation of shutdown cooling was also dependent on reactor coolant system pressure. Operators could not cooldown at the specified cooldown rate and initiate shutdown cooling as indicated in the calculation without violating plant procedures. For example, when operators attempted to cool down the plant at a sustained rate of 75° F per hour in the plant simulator (natural circulation conditions), reactor vessel steam voids formed. Operators were required by Emergency Operating

Instruction SO23-12-7, *Loss of Forced Circulation/Loss of Offsite Power,* Revision 19, Step 17.a to stop the cooldown and collapse the voids. This extended the simulator run time significantly, but the calculation did not account for this delay. The initial recommended cooldown rate, as specified in Emergency Operating Instruction Support Document SO23-14-11, Attachment 1, *Emergency Operating Instruction Supporting Attachments Bases and Deviation Justification,* Revision 1, was only 35 to 40° F per hour for natural circulation conditions. This document also recommended a maximum cooldown rate of 50° F per hour during natural circulation.

The team also noted that the calculation did not properly consider operating experience related to reactor vessel head voiding. Generic Letter 81-21, *Natural Circulation Cooldown,* dated May 5, 1981, addressed the potential for steam voids to form under natural circulation cooldown conditions. The generic letter recommended that licensee procedures contain specific actions to avoid the formation of reactor vessel steam voids and to mitigate steam voids if they did occur. The licensee's response to the generic letter, dated October 21, 1981, stipulated, in part, that plant procedures would include precautions delineating the maximum cooldown rate to avoid head voiding.

Calculation M-0050-018 concluded that operators would take 5.8 hours to achieve shutdown cooling conditions and that approximately 143,000 gallons of the 144,000 gallon Tank CST-121 capacity would be used. Since the calculation had virtually no margin for error, the inspectors concluded that the licensee did not have a reasonable basis to assume that Tank CST-121 could meet all design basis accident cooling needs.

In response to NRC concerns regarding the noted calculation and licensee's evaluation, the licensee revised their evaluation to specify that they could only achieve shutdown cooling initiation temperature (but not pressure) and wrote Action Request 0601000172 to evaluate the adequacy of Calculation M-0050-018.

Analysis. The failure to follow plant procedures was a performance deficiency. This finding is more than minor because it affected the mitigating system cornerstone objective (equipment performance attribute) of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The team determined that a Phase 3 significance determination was required because the finding screened as potentially risk significant due to a seismic initiating event. Region IV senior reactor analysts performed a Phase 3 significance determination, which is included as Attachment 2 to this report. The senior reactor analysts found the issue to be of very low safety significance.

Enforcement. Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion V, *Procedures*, requires, in part, that "Activities affecting quality shall be prescribed by documented . . . procedures . . . and shall be accomplished in accordance with these instructions." The exclusion of foreign materials from the condensate storage Tank CST-120 enclosures is an activity affecting quality. Procedure SO123-FO-1, requires, in part, "Foreign Material Exclusion Program provisions are to be applied to important-to-safety plant systems and components when exposed during . . . operation . . . Controls for foreign material exclusion SHALL be implemented to prevent the introduction of foreign material into mechanical or electrical equipment, components, or systems . . . foreign material exclusion controls SHALL provide for positive physical control and accountability of

material, equipment, and tools to prevent their introduction into mechanical . . . components or systems. . ." Contrary to the above, prior to July 12, 2006, the condensate storage Tank T-120 enclosures, on each unit, were exposed during plant operation to foreign materials, which could block flow to safety-related equipment and/or be transported into mechanical safety-related equipment and cause damage. The licensee failed to implement foreign material exclusion controls for the enclosures. The licensee captured this issue in their corrective action program as Action Request 060700471. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program (Action Request 060700765), it is considered a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000361;362/2006009-01, Failure to Follow Procedures Addressing Foreign Material Exclusion.

b.2 Inadequate Procedure for Emergency Diesel Generator Ground Alarm

Introduction. The team identified a Green noncited violation of Technical Specification 5.5.1.a for an inadequate emergency diesel generator ground fault alarm response procedure. Specifically, the procedure only had operators check for grounds associated with the emergency diesel generator itself and did not specify actions to address the more likely ground locations, which included components on the 4.16kV bus. Since other plant procedures permit cross-tying the safety-related buses on the opposite unit in the event of a loss-of-an emergency diesel generator, the failure to properly consider grounds in other locations could result in additional equipment failures.

Description. In accordance with the Updated Final Safety Analysis Report, one of the non-essential emergency diesel generator trips is an electrical ground fault trip. This trip is bypassed during a loss-of-coolant accident but is left intact for other events, including a loss-of-offsite power event.

On the safety-related 4.16kV buses, the licensee uses a high impedance grounding scheme. With this grounding configuration, a ground fault would not normally generate exceptionally high electrical currents but would, instead, increase the voltage across two phases of the emergency diesel generator by a factor of 1.73. The purpose of the ground fault trip is to limit generator exposure to this higher voltage, which could cause damage. Individual feeder breakers on the 4.16kV bus are not coordinated to isolate ground faults that might occur on individual components, so a ground fault anywhere on a 4.16kV train would result in tripping the associated emergency diesel generator.

The team identified that Alarm Response Instruction SO23-5-2.35.2, *Diesel Generator Local Annunciator Panel L161 Alarm Response*, Revision 5, was inadequate because it failed to provide appropriate instructions to operators in response to indication of a phase to ground fault on a 4.16kV bus. The procedural guidance was inappropriate because it only identified two causes for a ground fault: 1) generator phase to ground fault; and 2) generator leads phase to ground fault. Contrary to this guidance, when the diesel is supplying the 4.16kV bus, a ground anywhere on the supplied 4.16kV system can cause the emergency diesel to trip, not just on the generator itself. For example, a phase to ground fault associated with any of the emergency core cooling system pump motors, breakers and feeder cables could cause the associated emergency diesel generator to trip. Furthermore, these alternative ground locations were much more likely to occur than those specified in the alarm response instruction.

The team was further concerned because plant procedures permit cross-tying the loads on one unit (due to a lost power source) to an energy source on the opposite unit. For example, an operating bases earthquake could cause a loss-of-offsite power on both Units 2 and 3. If the Unit 2, Train A emergency diesel generator tripped on a ground fault, operators could power the Unit 2, Train A loads from the Unit 3, Train A bus. If the ground fault actually existed on a Unit 2, Train A load, versus on the diesel generator itself, this would cause the Unit 3, Train A emergency diesel generator to trip unnecessarily.

Technical Specifications 5.5.1.a requires, in part, procedures recommended by Regulatory Guide 1.33, Appendix A. Section 5 of this appendix recommends procedures for abnormal, offnormal, or alarm conditions. Additionally, the appendix specifies for these types of procedures that the procedure contain, in part, the immediate operator actions and the long-range actions. These operator actions, in this case, were inadequate.

Analysis. The failure to provide an adequate alarm response procedure was a performance deficiency. This issue was more than minor because the procedure deficiency affected the mitigating system cornerstone objective (procedure quality attribute) of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the issue screened as having very low safety significance because the finding was not a design or qualification deficiency, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events.

Enforcement. San Onofre Nuclear Generating Station, Units 2 and 3, Technical Specifications 5.5.1.a, requires, in part, procedures recommended by Regulatory Guide 1.33, Appendix A. Section 5 of this document specifies procedures for abnormal, offnormal, or alarm conditions. For these types of procedures, the appendix specifies that the procedure contain, in part, the immediate operator actions and the long-range actions. Contrary to the above, prior to July 20, 2006, Alarm Response Instruction SO23-5-2.35.2 was inadequate, in that, it failed to provide appropriate immediate and long-range operator actions. Specifically, the procedural instructions did not specify the most likely causes of the ground fault condition or provide suitable actions for responding to the condition. Since this finding was of very low safety significance and has been entered into the licensee's corrective action program (Action Request 060700753), it is considered a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000361/2006009-02, Inadequate Diesel Ground Alarm Procedure.

b.3 Failure to Follow Correct Methodology for 125Vdc Calculations

Introduction. The team identified a Green noncited violation of Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion III, *Design Control*, for the failure to properly calculate control circuit voltages associated with the Unit 3 motor-driven auxiliary Pump 3P504 breaker. The magnitude of the error used about 1/3 of the available design margin.

Description. Per Technical Specification Bases 3.8.4, *DC Sources-Operating*, the licensee sized the safety-related 125Vdc batteries in accordance with Institute of Electrical and Electronic Engineers (IEEE) Standard 485 - 1983, *Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications*. The results

from the sizing calculations are also used to determine available control circuit voltage. For circuits that use 125Vdc control power, including Class 1E 4kV circuit breakers, the licensee performed voltage drop calculations and verified that the voltage available to breaker controls circuits was greater than 90Vdc under worst case design basis accident conditions (Calculation E4C-131, *125VDC Control Circuit Analysis for Class 1E4KV and 480V Circuit Breaker Operation*, Revision 0).

The team identified: 1) that the licensee did not follow IEEE 485 when calculating the available battery voltage and 2) the calculated control circuit voltage for circuit Breaker 3A0612 (breaker to motor-driven auxiliary feedwater Pump 3P504) was adversely impacted by the error. IEEE Standard 485 described acceptable techniques for calculating the load during the battery duty cycle. One technique involved segregating the duty cycle into several discrete time intervals and determining the peak current within each interval. The peak current was then used to determine battery voltage for the interval. IEEE 485 specifies the minimum acceptable time interval as 1 minute. For the subject breaker, the licensee deviated from this method by using the non-peak battery current that occurred during the first minute of battery loading. This deviation resulted in increasing the calculated voltage. Based on the improper calculation method, the voltage available to the control circuit was 94.6Vdc. When corrected (including the removal of other conservatisms), the voltage was 93.1Vdc. While the ultimate change in voltage was not large, 1.5Vdc, the problem was considered more significant because of the small amount of margin available (94.6 - 90.0Vdc acceptance limit = 4.6Vdc margin). The magnitude of the error used about 1/3 of the available margin. Considering that these calculations would be used to support plant modifications and operability determinations, this issue could be more significant if left uncorrected.

Analysis. The failure to properly implement design controls was a performance deficiency. This issue was more than minor because, if left uncorrected, it could become a more significant safety concern. Specifically, the noted calculations are used for operability determinations and plant modifications. Uncorrected errors could mask equipment operability issues. This issue was similar to non-minor violation Example 3.j in NRC Inspection Manual Chapter 0612, Appendix E, *Examples of Minor Issues*, because there was a reasonable doubt of the operability of the pump breaker. Using the Manual Chapter 0609, Phase 1 screening worksheet, the issue screened as having very low safety significance because it was a design deficiency confirmed not to result in loss-of-operability in accordance with NRC Manual Chapter Part 9900, Technical Guidance*, Operability Determination Process for Operability and Functional Assessment*.

Enforcement. Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion III, *Design Control*, requires, in part, "Measures shall be established to assure that the applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application . . . are correctly translated into specifications, drawings, procedures and instructions. . . . The design control measures shall provide for verifying or checking the adequacy of design, such as . . . by the use of alternate or simplified calculational methods . . . ." The San Onofre Nuclear Generating Station Units 2 and 3 Technical Specifications Bases, Section B 3.8.4, specified IEEE 485-1983 as the applicable standard for battery sizing. Additionally, San Onofre Nuclear Generating Station Procedure E4C-017 states, in part, "the methodology used to analyze and validate the DC system as meeting its safety-related requirements is from IEEE Standard 485-1983." IEEE 485-1983 specified, in part, the minimum time interval assumed

for analysis purposes as being 1 minute. Further, the standard states: ". . . if a discrete sequence can be established, the load for the period should be assumed to be the maximum load at any instant." Contrary to the above, prior to July 20, 2006, in Calculation E4C-131, the licensee used the non-maximum load during the first minute of post-accident battery operation. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program (Action Request 060700765), it is considered a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000362/2006009-03, Incorrect Methodology for 125Vdc Calculations.

b.4 <u>Failure to Properly Address Industry Information Regarding Air in Safety Injection lines</u>

<u>Introduction</u>. The team identified a Green noncited violation of Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion XVI, *Corrective Actions*, for the failure to promptly identify a condition adverse to quality (trapped air in the safety injection suction lines). Each suction line contained approximately 11.5 cubic feet of trapped air but the licensee's official design calculations assumed the lines were full of water. Additionally, industry operating experience notified the licensee that air in the safety injection system suction lines could cause operational problems (a condition adverse to quality), but the licensee failed to promptly identify the condition at San Onofre Nuclear Generating Station. The licensee's engineering evaluation erroneously determined that San Onofre Nuclear Generating Station was not vulnerable to the condition identified in the operating experience.

<u>Description</u>. Westinghouse Service Advisor Letter NSAL-04-7*, Containment Sump Line Fluid Inventory*, dated November 15, 2004, notified the licensee of a potential generic problem with Combustion Engineering plants. The letter stated, in part:

> The containment emergency core cooling system suction lines between the sump isolation valve and the sump check valve at two Combustion Engineering sites were found to contain trapped air. Upon receipt of a recirculation actuation signal following a postulated loss-of-coolant accident, such trapped air would be drawn into the operating emergency core cooling system pumps . . . Ingestion of the trapped air by an operating emergency core cooling system pump may create hydraulic instabilities that could result in damage and possible loss of pump operability . . . Internal clearances are such that air ingestion into an operating high pressure (multi-stage) centrifugal pump could result in damage due to contact between rotating and stationary internal components . . . The rate of influx of water from the containment sump when the isolation valves open to initiate the recirculation/long term core cooling mode of operation may be sufficient to sweep any trapped air into the emergency core cooling system pumps.

The licensee evaluated the operating experience via Action Request 041101570, dated November 24, 2004. The response stated, in part:

> During normal operation the closed valves assure that the pipe will stay filled and not evaporate . . .
>
> With the containment sump filled above the suction strainers, the line would readily vent whenever the isolation valves were open.

> The issue raised by Westinghouse in NSAL-04-7 is applicable to San Onofre Nuclear Generating Station, but Units 2 and 3 are not vulnerable to the problem because the existing piping layout and operating procedures preclude the accumulation of significant quantities of air trapped in the containment sump lines. No further action is required regarding NSAL-04-7.

The team identified that the licensee's evaluation was inadequate for the following reasons:

- The piping was not full of water. Operating Instruction SO23-3-2.7.2, Attachment 7, *Filling the Containment Emergency Sump Suction Lines*, Revision 12, directs operators to establish level in the suction piping lines at a point where a portion of the piping would remain voided. In response to the team questions, the licensee determined that approximately 11.5 cubic feet of air would remain in each train.

- The licensee had no engineering evaluation or other justification that would support the position that, "with the containment sump filled above the suction strainers, the line would readily vent whenever the isolation valves were open." Contrary to this assertion, the Westinghouse advisory letter stated:

  > The rate of influx of water from the containment sump when the isolation valves open to initiate the recirculation/long term core cooling mode of operation may be sufficient to sweep any trapped air into the emergency core cooling system pumps.
  >
  > Inspector Note: When the containment sump isolation valves open, the pressure in containment is much higher than the pressure in the piping. When the valves open, flow would travel into the piping, versus outward.

- The existence of air in the safety injection piping was an unanalyzed condition for San Onofre Nuclear Generating Station. Calculation A-SG-FE-0090, *Post Loss of Coolant Accident Long-Term Cooling*, Revision 0, assumed the piping lines were full of water.

In response to the team's concerns, the licensee evaluated current operability. The licensee concluded that, based on engineering judgement, the air would settle in high portions of the piping system and would not likely reach the pumps. Therefore, the trains remained operable. As a long-term corrective measure, the licensee planned to perform an analysis to demonstrate acceptability of the suction piping air voids.

Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion XVI, requires the licensee, in part, to promptly identify and correct conditions adverse to quality. Air voids in the safety injection suction lines, a condition outside the system's design, was a condition adverse to quality. The licensee failed to promptly identify and correct the condition adverse to quality following the notification of a generic problem from a reputable vendor.

Analysis. The failure to promptly identify and correct a condition adverse to quality in response to applicable operating experience was a performance deficiency. This finding was more than minor because it affected the mitigating system cornerstone objective (equipment performance attribute) to ensure the reliability and capability of equipment needed to respond to initiating events. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the finding was of very low safety significance because it was a design deficiency confirmed not to result in loss-of-operability in accordance with Part 9900, Technical Guidance. This finding has a cross-cutting aspect in the area of problem identification and resolution, in that the licensee failed to thoroughly evaluate applicable industry operating experience concerning air voids in recirculation piping suction lines.

Enforcement. Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion XVI, requires, in part, that "measures shall be established to assure that conditions adverse to quality, such as . . . nonconformances are promptly identified and corrected." Trapped air in the safety injection suction lines was a nonconformance (condition adverse to quality) because the condition was outside the safety injection system design. The design assumed that the lines were full of water. Contrary to the above, as of July 20, 2006, the licensee was notified of the potential condition adverse to quality on approximately November 15, 2004, but failed to promptly identify and correct the problem. Because this violation was of very low safety significance and has been entered into the licensee's corrective action program as Action Request 060700747, this violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000361;362/2006009-04, Failure to Identify Air Voids in Safety Injection Suction Piping.

b.5 Failure to Promptly Identify Nonconforming Emergency Diesel Generators

Introduction. The team identified a Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion XVI, violation for the failure to promptly identify a condition adverse to quality (Train A emergency diesel generators were no longer seismically qualified). The licensee had identified that a ground fault on a nonsafety-related uninterruptible power supply circuit could cause the emergency diesel generator to trip during a fire but failed to further determine that the same scenario could occur during a seismic event (with a loss-of-offsite power).

Discussion. On January 28, 2005, as documented in Action Request 05101702, the licensee identified that a ground fault on a nonsafety-related uninterruptible power supply, powered from the Train A Class 1E 4.16kV bus, could cause the associated emergency diesel generator to trip during a fire scenario. The emergency diesel generators were equipped with a ground fault relay that would trip the diesels in response to a line to ground fault anywhere on the diesel supplied 4.16kV system. The trip signal is bypassed for a loss-of-coolant accident but is intact for a loss-of-offsite power event. This condition existed on both units since initial construction.

The licensee established a fire watch as a short-term measure and completed a circuit modification on March 2, 2005, on Unit 2 and May 25, 2005, on Unit 3. The modification established proper coordination between the breaker and the 4.16kV bus so that a ground fault on the uninterruptible power supply circuit would trip the associated breaker and not the emergency diesel generator.

The team identified that the same type of ground fault could trip the Train A emergency diesel generator during a seismic event. The licensee should have identified this problem as part of the extent of condition evaluation for the original concern. Further, the team concluded that, at the time, operability of the Train A emergency diesel generators on each unit was not assured. The Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 2, *Protection Against Natural Phenomena*, states, in part,

> Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes . . . without loss of capability to perform their safety functions.

The licensee was committed to Criterion 2 in Section 3.1.1.2.2 of their Updated Final Safety Analysis Report. The failure to maintain seismic qualification would, at minimum, be a nonconforming condition. In accordance with the NRC Manual Chapter Part 9900 Technical Guidance, the licensee should have identified the nonconforming condition and evaluated emergency diesel generator operability. Failure to meet the conditions required for operability would have driven the licensee to follow the applicable technical specification requirements. The failure to identify the condition adverse to quality bypassed this process and operability for seismic events was not evaluated. The team noted that because the licensee had corrected the condition by the time of the inspection, current emergency diesel generator operability was not a concern.

Analysis. The failure to promptly identify a condition adverse to quality was a performance deficiency. This finding is more than minor because it affected the mitigating system cornerstone objective (equipment performance attribute) of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, *Significance Determination Process,* Phase 1 screening worksheet, the internal events portion of the worksheet did not apply, because the finding only involved an external seismic event with a loss of offsite power. Specifically, Train A emergency diesel generator operability was not assured for seismic events. Additionally, for external events, the finding screened as have very low safety significance because it did not involve the loss or degradation of equipment or function specifically designed to mitigate an external event (e.g., seismic snubbers, flooding barriers, tornado doors) and the safety function was not considered completely failed or unavailable, as the Train B emergency diesel generators were unaffected by the issue. This finding has a cross-cutting aspect in the area of problem identification and resolution, in that engineers failed to perform an appropriate extent of condition review and promptly identify the nonconforming emergency diesel generators, a condition adverse to quality.

Enforcement. Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion XVI, requires, in part, that "measures shall be established to assure that conditions adverse to quality, such as . . . nonconformances are promptly identified and corrected." Contrary to the above, from January 28, 2005 until March 2, 2005 (Unit 2) and May 25, 2005 (Unit 1), the licensee failed to promptly identify a nonconforming condition, non-seismic qualification of Train A emergency diesel generators. Because this violation was of very low safety significance and has been entered into the licensee's corrective action program as Action Request 060600500, this violation is being treated as a noncited violation consistent with

Section VI.A.1 of the NRC Enforcement Policy: NCV 05000361;362/2006009-05, Failure to Identify Diesel Generator Seismic Nonconformance.

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

The team reviewed actions requests associated with the selected components, operator actions and operating experience notifications. In addition, this report contains the following issue that has problem identification cross-cutting aspects.

> Section 1R21.b.4 documents an issue where engineers failed to perform an adequate evaluation of recent operating experience. They had failed to address air voiding in the San Onofre Nuclear Generating Station safety injection suction lines.
>
> Section 1R21.b.5 documents a finding where engineers failed to perform an adequate extent of condition review in response to a fire protection related problem, where an emergency diesel generator could trip unexpectedly. Consequently, the engineers failed to promptly identify a related condition adverse to quality, in that the same problem also affected the emergency diesel generators during seismic events.

4OA3 Event Followup

(Closed) Licensee Event Report 05000361;362/2005-003-00. Technical Specification Violation for Inoperable Offsite Sources

On March 14, 2005, the licensee identified that the undervoltage relays on both units were set such that the offsite transmission network may not have provided the necessary voltage to the Class 1E system under certain circumstances. This issue is addressed in Section 4OA7 of this report.

4OA6 Meetings, Including Exit

On July 20, 2006, the team leader presented the preliminary inspection results to Mr. B. Katz, Vice President, Nuclear Oversight and Regulatory Affairs, and other members of the licensee’s staff. On November 30, 2006, the Engineering Branch 1 Chief conducted a telephonic final exit meeting with Mr. B. Katz and other members of the licensee's staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-16000, for being dispositioned as a licensee identified noncited violation.

Licensee Event Report 2005-003-00: Technical Specification 3.8.1, AC Sources - Operating, requires that two qualified circuits between the offsite transmission network and onsite Class 1E AC electrical power distribution system shall be operable in Modes 1 through 4. Contrary to this requirement, since 1995 until March 14, 2005, the licensee identified that the undervoltage relays on both units were set such that the offsite transmission network may not have provided the necessary voltage to the Class 1E system under certain circumstances. The undervoltage relays were set to trip at 218kV when they should have been set to 222.2kV to ensure operability. This issue was entered into the licensee's corrective action program as Action Requests 050301091 and 050500092.

Attachments: 1 - Supplemental Information
2 - Preliminary Determination of Significance, Foreign Material in Condensate Storage Tank Enclosure

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Breig, Station Manager
B. Katz, Vice President, Nuclear Oversight and Regulatory Affairs
M. Love, Manager, Maintenance
N. Quigley, Manager, Mechanical/Nuclear Maintenance Engineering
L. Pressey, Nuclear Regulatory Affairs
A. Scherer, Manager, Nuclear Regulatory Affairs
M. Short, Manager, Systems Engineering
T. Vogt, Manager, Operations
R. Waldo, Vice President, Nuclear Generation
D. Wilcockson, Manager, Plant Operations
C. Williams, Manager, Compliance
T. Yackle, Manager, Maintenance Engineering

NRC personnel

C. Osterholtz, Senior Resident Inspector
M. Sitek, Resident inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000361;362/2006009-01	NCV	Failure to Follow Procedures Addressing Foreign Material Exclusion (Section 1R21.b.1)
05000361;362/2006009-02	NCV	Inadequate Diesel Ground Alarm Procedure (Section 1R21.b.2)
05000362/2006009-03	NCV	Incorrect Methodology for 125VDC Calculations (Section 1R21.b.3)
05000361;362/2006009-04	NCV	Failure to Identify Air Voids in Safety Injection Suction Piping (Section 1R21.b.4)
05000361;362/2006009-05	NCV	Failure to Identify Diesel Generator Seismic Nonconformance (Section 1R21.b.5)

Closed

05000361;362/2005-003-00	LER	Inoperable Offsite Power Sources (Section 4OA7)

LIST OF DOCUMENTS REVIEWED

Action Requests

000201051	000300526	000501428	010801579	011001639

020400316	041001426	051000182	060700495	980300468
020700633	041101207	060005000	060700747	980300481
031100883	041101570	060200773	060700753	980902035
031100884	041201050	060201011	920400105	990200519
040500115	050301091	060201045	960100058	990200520
040500760	050401120	060400439	960900040	990601566
040700933	050500092	060501087	961100984	990900145
040900783	050800238	060700471	980101743	990900729

Calculations

A-SG-FE-0074, *Small Break Loss of Coolant*, Revision 0

A-SG-FE-0090, *Post Loss of Coolant Accident Long-Term Cooling*, Revision 0

C-259-05.02.02, Storage Tank Building-Concrete Design," Revision 2

C-259-5.02.05, *Watertight Reliability of Condensate Storage Tank and It's Concrete Enclosure Walls Under a Operating Basis Earthquake and Tornado Events*, Revision 0

C-259-5.02.06, *Tank Building Structural Steel Barriers and Restraints*, Revision 2

E4C-017, *125V Battery and DC System Sizing,* Revision 18

E4C-017.1, *Class 1E 125VDC System Data/Loading,* Revision 2

E4C-042, *8kV Power Cable Ampacity P504 Section 8.4.2 Bus 2A06,* May 15, 1998

E4C-051, *600V Power Cable Ampacity for 480V Load Center with Maintained Spacing,* Revision 15

E4C-065, *Adequacy Evaluation-Cables in Raceways with Regulatory Guide 1.75/Appendix R Barriers,* Revision 13

E4C-084, *Unit 2 Motor Control Center Control Circuit Voltage Analysis,* Revision 0

E4C-085, *Unit 3 Motor Control Center Control Circuit Voltage Analysis,* Revision 0

E4C-086, *San Onofre Nuclear Generating Station 2 & 3 Data Development and Documentation,* Revision 5

E4C-098, *4kV Switchgear Protective Relay Setting Calculation,* Revision 3

E4C-099, *Safety-related 480V Power Circuit Breaker Settings,* Revision 1

E4C-0112, *Class 1E 480V Motor Control Center Protection,* Revision 11

E4C-120, *600 Volt Power Cable Ampacity for 480V Motor Control Center & 120 VAC Panels,* Revision 2

E4C-130, *TLU Calculation for Undervoltage Relay Circuit at Class 1E 4KV Switchgear,* Revision1, ECN No. A42610

E4C-131, *125VDC Control Circuit Analysis for Class 1E 4kV and 480V Circuit Breaker Operation*, Revision 0

J-BHB-029, *Reactor Water Storage Tank Minimum Level to Maintain Safety Analysis Assumptions, Including Instrument Uncertainties,* Revision 0

MS-123-125, *Generic Letter 89-10 Setpoint Calculation Design Guide,* Revision 2

M-0012-01D, *Net Positive Suction Head of Emergency Safety Feature Pumps,* Revision 2

M-0012-027, *High Pressure Safety Injection & Low Pressure Safety Injection Inservice Test Minimum Performance Requirements,* Revision 0

M-0012-033, *High Pressure Safety Injection Pump Minimum Performance Requirements*, Revision 2

M-0012-036, *Postulated Transient Recirculation Flow from Reactor Water Storage Tanks*, Revision 1

M-0013-002, *Safety Injection and Recirc Total Head Loss-High Pressure Safety Injection and Hot Leg Injection*, Revision 1

M-0026-011, *Component Cooling Water Flow-Pressure Distribution Analysis,* Revision 1

M-0027-023, *Component Cooling Water/Salt Water Cooling Heat Exchanger Operability,* Revision 0

M-0027-029, *Component Cooling Water/Salt Water Cooling Heat Exchanger Performance Tests,* Revision 0

M-0041-086, *Auxiliary Feedwater Pump Motor Bearing Qualification,* Revision 2

M-0050-017, *BTP RSB 5-1 Condensate Inventory,* Revision 3

M-0050-018, *Evaluation of CST-121 Volume Requirements,* Revision 0

M-0056-018, *Auxiliary Feedwater Pump Net Positive Suction Head Requirements*

M-0056-036, *Auxiliary Feedwater Pumps System Performance Curves-Inservice Test Curves* Revision 3

M-0072-036, *Containment Emergency Cooler Performance Verification*, Revision 0

M-12.2, *Safeguard Pumps Net Positive Suction Head with Suction from Reactor water Tank*, Revision 0

M-12.12, *Emergency Sumps-Margin Against Vortexing and Air Binding*, Revision 1

M-89-10-SP-3HV9304, *Calculation Sump Suction Isolation Valve*, Revision 1

M-8910-1204-OB-001, *Generic Letter 89-10 Operational Basis Calculation for High Pressure Safety Injection Cold Leg and Hot Leg Injection Valves*, Revision 1

M-8910-1305-OB-001, *Motor Operated Valve 2HV-4706, Auxiliary Feedwater Pressure Differential For Generic Letter 89-10*, July 14, 1997

M-8910-SP-3HV9330, *Generic Letter 89-10 Setpoint For 3HV-9330*, Revision 2

NFM-2/3-TA-0010, *San Onofre Nuclear Generating Station 2 and 3 Loss of Normal Feedwater Flow*, Revision 1

N-0240-006, *Reactor Water Storage Tank Technical Specification Requirement*, Revision 0

N-4080-026, *Containment Pressure/Temperature Analysis for Design Basis Loss of Coolant Accident,* Revision 1

N-4090-008, *Spent Fuel Pool Decay Heat*, Revision 3

N-4090-009, *Units 2 and 3 Auxiliary Feedwater Pump Room and Doghouse Pressure-Temperature Analysis*, Revision 0

San Onofre Nuclear Generating Station Generic Letter 89-10, *Program Design Basis, Motor-Operated Valve-2HV4706 Design Basis Margin Assessment*, April 20, 2004

SO23-452-F, *Salt Water Cooling System Operating Characteristics*, Revision 0

SO23-454-B, *Main Steam Safety Relief Valves*, Revision 0

SO23-457-D, *Auxiliary Feed Water Pump Room Heat Load*, Revision 1

S-PEC-049, *Safeguards Pump Net Positive Suction Head Available*, Revision 0

S-PEC-086, *Safety Injection System Head Losses. Net Positive Suction Head Requirements of Safeguard Pumps*, Revision 0

T/H-2/3-MAAP04-002, *Small Loss of Coolant Accident (2 inches) Recirculation Actuation Signal Actuation Timing*, Revision 2

1370-ICE-36123, *Refueling Water Tank Level*, Revision 01

Completed Inservice Tests

SO23-3-3.60.4 for Saltwater Pump S21413MP112; dated September 29, 2005, March 21, 2006, and June 8, 2006.

SO23-3-3.60.6 for Auxiliary Feedwater Pump (Turbine Driven) S21305MP140**;** dated April 30, 2006, January 3, 2006, February 9, 2004.

SO23–3.60.6 for Auxiliary Feedwater Pump (Motor Driven) S21305MP141**;** dated April 5, 2006, October 19, 2005, July 27, 2005.

Design Basis Documents

DBD-SO23-TR-AA, *Accident Analysis Design Basis Document*, Revision 8

DBD-SO23-TR-EQ, *Environmental Qualification Topical Report*, Revision **7**

DBD-SO23-TR-IS3, *Inservice Testing Topical Report*, Revision 0

DBD-SO23-TR-0A, *Operator Action Design Basis Document Background*, Revision 0

DBD-SO23-390, *Chemical and Volume Control System*, Revision 8

DBD-SO23-400, *Component Cooling Water System,* Revision 9

DBD-SO23-410, *Saltwater Cooling System Design Basis Document*, Revision 6

DBD-SO23-740, *Safety Injection, Containment Spray, and Shutdown Cooling Systems*, Revision 8

DBD- SO23-780, *Auxiliary Feedwater*, Revision 7

Drawings

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30165 Sheet 4, *One Line Diagram Non-1E Uninterruptible Power Supply System*, Revision 15

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41066, *Component Cooling Water Pump -Tag No. 2P025, Inservice Test Curves*, Revision 0

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02060733000	03101035000	04042171000	05031448001	06070075300
02080142000	03110764000	04051972000	05080069000	06070076400
02110029500	04010139300	04060175300	05100054300	10003670200
02110156000	04011788000	04091067000	05120460000	30084007000
03030272000	04030129700	05010170200	06011550000	30089001000

30094007023	71004003000	80001021000	81010748200	96080062000
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L-HCAFWPB-FU/ Auxiliary Feedwater, Control Auxiliary Feedwater Pumps After Fire

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L-HCCRCNNCTU/Auxiliary Feedwater, Cross-connect Auxiliary Feedwater Pump P141(or P504) to Opposite Steam Generator

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L-HCCST120-DEP, Manually Provide Condensate Storage Tank Make-up from Hill Tanks

L-HCCST120-U/Auxiliary Feedwater, Manually Provide Condensate Storage Tank Make-up from Hill Tanks

L-HCDMWTMU-U/Auxiliary Feedwater, Manually Provide Condensate Storage Tank Make-up from Hill Tanks

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U2/3 Safety Evaluation Report-Supplement 3, dated September 16, 1981.

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17175-2, *Preliminary Qualification Plan for Diesel Generator System*, dated May 2, 1978

The team provided the following information request in writing to the licensee prior to the inspection.

Initial Information Request
Component Design Basis Inspection (71111.21)
San Onofre, Units 2 and 3

Please provide the following information in order to support the NRC's component design basis inspection effort at your facility. If there are problems obtaining any of this information, please call the Team Leader, George Replogle at (817) 860-8249 to discuss alternate arrangements. We would like to have the information ready when we arrive on site for the "bag-man" portion of the inspection on May 30, 2006.

We prefer, but it's not required, that the information be provided electronically and in a searchable format, such as Adobe, Word, Word Perfect, or Excel. Other licensee's have found that providing the information on a CD is effective and efficient.

1. The risk ranking of components from your site specific probabilistic safety analysis sorted by Risk Achievement Worth and by Birnbaum Importance.

2. A list of your top 500 cutsets from your probabilistic safety analysis.

3. Risk ranking of operator actions from you site specific probabilistic safety analysis sorted by Risk Achievement Worth. Provide copies of your human reliability worksheets for these items (you may limit this list to the 100 most risk significant actions).

4. If you have an external events or fire probabilistic safety analysis model, provide the information requested in Items 1 and 2 for external events and fire.

5. Any pre-existing evaluation or list of components and calculations with low design margins (i.e. pumps closest to the design limit for flow or pressure, diesel generators close to design required output, heat exchangers close to rated design heat removal etc.)

6. For the last two years, a list of operating experience evaluations, modifications and corrective actions sorted by component or system. A one line, or short, description is acceptable.

7. A list of any common-cause failures of components in the last 5 years at your facility.

8. A list of Maintenance Rule functions.

9. A list of your Maintenance Rule a(1) components.

10. A list of your current temporary modifications.

11. A current list of "operator work arounds."

12. Piping and instrument drawings for your emergency core cooling systems, emergency diesel generators and off-site power supplies. At this time, only the

mechanical piping drawings are needed for the emergency core cooling systems and the emergency diesel generators.

In addition to the above, if available electronically, please provide a copy of each of the following on CD.

1. Final/Updated Safety Analysis Reports

2. Technical Specifications

3. Design Bases Documents for the emergency core cooling systems (including auxiliary feedwater), emergency diesel generators and off-site power supplies

4. System descriptions or operator training manuals for the emergency core cooling systems, emergency diesel generators and off-site power supply systems

Thank you for your cooperation in these matters.

ATTACHMENT 2
Final Determination of Significance
San Onofre Nuclear Generating Station
Foreign Material in Condensate Storage Tank Enclosure

1. Statement of Performance Deficiency

The licensee failed to follow procedures related to foreign material controls. Specifically, the licensee failed to follow procedural requirements and establish the T-120 condensate storage tank (CST) enclosure as a foreign material exclusion area. In addition, the licensee failed to properly address industry operating experience related to foreign materials in auxiliary feedwater system water sources. This resulted in foreign material being left in the CST T-120 enclosure.

2. Phase 1 Screening Logic, Results and Assumptions

In accordance with NRC Inspection Manual Chapter 0612, Appendix B, *Issue Screening*, the inspectors determined that the failure to provide controls over foreign material, which could enter the safety-related CST and affect secondary cooling was a licensee performance deficiency. The issue was more than minor because the issue was related to the equipment performance attribute and affected the objective of the Reactor Safety/Mitigating Systems Cornerstone. Specifically, the ability to provide makeup water to CST Tank T-121 following a seismic event was impaired by the presence of foreign material in the CST T-120 enclosure.

The inspectors evaluated the issue using the SDP Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones provided in Manual Chapter 0609, Appendix A, *Significance Determination of Reactor Inspection Findings for At-Power Situations*. In the Mitigating Systems Cornerstone section the finding screened as potentially risk significant to a seismic external initiating event. Therefore, this Phase 1 screening transitions directly into a Phase 3 analysis.

3. Phase 3 Evaluation for External Events

The analysts determined that, for the subject performance deficiency to affect the core damage frequency, a seismic event must result in both a loss of offsite power (LOOP) and the failure of CST Tank T-120 resulting in the tank dumping its contents into the enclosure. Additionally, to quantify the increase in core damage frequency (ΔCDF) caused by the debris in the Tank T-120 enclosure, the analysts must estimate the probability that the auxiliary feedwater system would fail as a result of this performance deficiency, as well as the change in core damage probability assuming that the above postulated conditions occurred.

As such, the analysts evaluated the subject performance deficiency by determining each of the following parameters for any seismic event producing a given range of median acceleration "a" [SE(a)]:

a. The frequency of the seismic event SE(a) ($\lambda_{SE(a)}$) ;
b. The probability that a LOOP occurs during the event ($P_{LOOP\text{-}SE(a)}$);
c. The probability that Tank CST T-120 fails during the event ($P_{T120\text{-}SE(a)}$);

d. The probability that the AFW system fails ($P_{AFW-SE(a)}$);
e. The probability that operators fail to recover AFW ($P_{NR-SE(a)}$); and
f. The conditional change in core damage probability ($\Delta CCDP_{SE(a)}$)

The ΔCDF for the acceleration range in question ($\Delta CDF_{SE(a)}$) can then be quantified as follows:

$$\Delta CDF_{SE(a)} = \lambda_{SE(a)} * P_{LOOP-SE(a)} * P_{T120-SE(a)} * P_{AFW-SE(a)} * P_{NR-SE(a)} * \Delta CCDP_{SE(a)}$$

Assuming that each range "a" is selected by the analysts specifically to be independent of all other ranges, the total increase in risk, ΔCDF, can be quantified by summing the $\Delta CDF_{SE(a)}$ for each range evaluated as follows:

$$\Delta CDF = \sum_{a=.03g}^{6g} \Delta CDF_{SE(a)}$$

over the range of SE(a).

a. Frequency of the Seismic Event

NRC research data indicated that seismic events of 0.05g or less have little to no impact on internal plant equipment. As such, the analysts assumed that seismic events less than 0.03g do not directly affect the plant. The analysts assumed that seismic events greater than 6.0g lead to core damage. The analysts, therefore, examined seismic events in the range of 0.03g to 6.0g. The analysts divided that range of seismic events into segments (called "bins" hereafter).

In order to determine the frequency of a seismic event for a specific range of ground motion (g values), the analysts used the licensee's individual plant examination of external events seismic hazards vector converted to peak ground acceleration. The analysts obtained values for the frequency of the seismic event that generates a level of ground motion that exceeds the lower value in each of the bins and the frequency that exceeds the upper value of each bin. The analysts then calculated the difference in these "frequency of exceedance" values to obtain the frequency of seismic events within the binned seismic event ranges.

b. Probability of a Loss of Offsite Power

The analysts assumed that a seismic event severe enough to break the ceramic insulators on the transmission lines would cause an unrecoverable loss of offsite power. The analysts obtained data on switchyard components from the Risk Assessment of Operating Events Handbook; Volume 2, *External Events*, Revision 4, which referenced generic fragility values listed in:

- NUREG/CR-6544, *Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences*, April 1998; and

- NUREG/CR-4550, Volumes 3 and 4 part 3, *Analysis of Core Damage Frequency: Surry / Peach Bottom'* 1986

The references describe the mean failure probability for various equipment using the following equation:

$$P_{fail}(a) = \Phi\ [\ \ln(a/a_m) / (\beta_r^2 + \beta_u^2)^{½}]$$

Where Φ is the standard normal cumulative distribution function and

- a = median acceleration level of the seismic event;
- a_m= median of the component fragility;
- β_r= logarithmic standard deviation representing random uncertainty;
- β_u= logarithmic standard deviation representing systematic or modeling uncertainty.

In order to calculate the LOOP probability given a seismic event the analysts used the following generic seismic fragilities for switchyard insulators:

a_m = 0.3 g
β_r = 0.30
β_u = 0.45

Using the above normal cumulative distribution function equation the analysts determined the conditional probability of a LOOP given a seismic event. For each of the bins the calculation was performed substituting for the variable "a" (median acceleration level) the median of the peak ground acceleration levels obtained from the bins described above.

c. Probability That Tank CST T-120 Fails

In order to calculate the probability that CST T-120 ruptures given a seismic event, the analysts used the following generic seismic fragilities (assuming a large flat-bottom storage tank):

a_m = 1.1 g
β_r = 0.30
β_u = 0.35

These values were selected from NUREG/CR-6544, Table 6-1 and the analysts used them in conjunction with the seismic event modeling and seismic risk quantification methods described in Section 2.4 of the Risk Assessment of Operating Events Handbook, Volume 2: External Events, Revision 0. Using the standard normal cumulative distribution function equation shown in Section 2.b of this analysis, the analysts determined the conditional probability that CST T-120 would rupture given a seismic event in the acceleration range from each of the bins.

d. Probability of Auxiliary Feedwater System Failure

The analysts determined that the probability of AFW system failure ($P_{AFW\text{-}SE(a)}$) was equal to the probability of completely clogging the CST Tank T-120 enclosure sump (P_{SUMP}) or the probability that the sump doesn't clog ($/P_{SUMP}$) and foreign material would be ingested into one or more of the AFW pumps failing the pumps (P_{Ingest}) and random failures occur in the remainder of the AFW system ($P_{random\ AFW}$) rendering the entire system inoperable. The equation can be expressed as:

$$P_{AFW\text{-}SE(a)} = P_{SUMP} + (/P_{SUMP} * P_{INGEST} * P_{random\ AFW})$$

- Probability of Sump Clogging:

  The analysts used the debris characteristics provided by the inspectors and dimensions of the enclosure to estimate the probability of the sump clogging (P_{SUMP}). As a first-order approximation, the analysts used a ratio of the open area of the material (including the signs) found in the enclosure to the area of the enclosure itself. This random point estimate probability was then modified by evaluating the dimensions of each piece of debris, the dimensions of the sump itself, and the probability that the material could clog the sump grating.

  The analysts concluded that the signs themselves were large enough to block the sump. The best estimate probability that the signs would randomly block the sump was conservatively determined to be 9.80×10^{-3}. As such, the probability that the debris doesn't clog the sump was the complement of this value, 9.902×10^{-1}.

- Probability of Foreign Material Ingestion Failing AFW:

  The probability of foreign material that becomes ingested into one or more of the AFW pumps and thus failing that pump was conservatively assumed by the licensee to be 1.0. The licensee's PRA Report, "Foreign Material found inside the condensate tank CST T-120 enclosure," PRA -06-01, stated that ". . . any debris which is drawn into the pump suction will fail that pump."

  However, the analysts noted that the debris would still have to be transported to the sump, through the sump-to-tank line, into CST T-121 and from there into the suction of the pump. Therefore, the analysts calculated a bounding value for this probability by estimating the probability that any of the foreign debris would randomly distribute and land touching the sump grating. Given that there were 17 pieces of debris that were of concern, the relative shapes of these pieces, and the size of the sump versus free area of the enclosure, the analysts calculated a 29% probability that any piece of debris that fell to the enclosure floor would touch the sump grating.

  The analysts then assumed that this debris would be transferred through the crossover line and that it would be sucked into the

suction of the Pump P-141 because the discharge of the gravity feed line from the Tank CST T-120 enclosure enters Tank CST T-121 near and above the suction of Pump P-141.

The probability of random AFW failure was determined by using the SPAR Model and solving the AFW fault tree assuming the baseline seismic event which caused a nonrecoverable LOOP and failure of at least one of the AFW pumps. The resultant random failure probability of AFW was determined to be 1.97×10^{-3}.

Solving the equation for the best estimate probability of AFW system failure:

$$\begin{aligned} P_{AFW\text{-}SE(a)} &= P_{SUMP} + (/P_{SUMP} * P_{INGEST} * P_{random\ AFW}) \\ &= 9.80 \times 10^{-3} + (9.90 \times 10^{-1} * 0.29 * 1.97 \times 10^{-3}) \\ &= 1.03 \times 10^{-2} \end{aligned}$$

e. Evaluation of Recovery Actions

The analysts evaluated the following potential recovery actions:

- Recovery via Tank-to-Tank Cross Tie

  Using the SPAR-H method of estimating human error probabilities (HEPs) described in NUREG/CR-6883, the analysts conservatively determined that a best estimate probability that operators fail to establish a connection via the tank-to-tank cross tie from CST T-120 to CST T-121 is 0.36. This assumed the nominal diagnosis human error probability (NDHEP) of 0.01 and the nominal action human error probability (NAHEP) of 0.001 were each multiplied by the performance shaping factors (PSFs) documented below:

  Diagnosis:

(1)	Available Time:	Extra Time	(PSF = 0.1)
(2)	Stress:	High	(PSF = 2)
(3)	Procedures:	Incomplete	(PSF = 20)
(4)	Ergonomics:	Poor	(PSF = 10)

  Action:

(1)	Available Time:	> 5 Times Required	(PSF = 0.1)
(2)	Stress:	High	(PSF = 2)
(3)	Complexity:	Moderately Complex	(PSF = 2)
(4)	Procedures:	Incomplete	(PSF = 20)
(5)	Ergonomics:	Poor	(PSF = 10)

  The analysts noted, qualitatively, that there was also a probability that the failure mode of Tank CST T-120 would cause the cross connect to be unavailable. This corroborated the use of a large nonrecovery value. This nonrecovery probability of 0.36 was applied only to sump clogging sequences.

- Fire Water Aligned to AFW Discharge

  On October 4, 2006, the analysts discussed aligning fire water to the discharge of the AFW pumps with the licensee's PRA staff. Some of the licensee's engineers considered aligning fire water to allow cooling in a pressurized reactor via a depressurized steam generator was realistic, based on an in-house calculation. However, there was disagreement within the licensee's organization on whether the calculation was valid. The licensee's PRA staff had not determined that this method had a realistic probability of success. As such, the licensee has neither modeled this recovery in their PRA of record, nor attempted to show a success path for their evaluation of this specific performance deficiency.

- Other Sources of Water Makeup to Tank CST T-121

  The analysts discussed the other sources and/or methods of tank makeup with the PRA staff on October 4, 2006. The licensee stated that successful alignment of other makeup sources was hypothetical and developed after identification of this performance deficiency. None of the alignment methods were proceduralized nor had materials and equipment been staged and dedicated for the task. As such, these alignment methods do not meet the requirements of Regulatory Guide 1.200 for being incorporated into a PRA study. Therefore, the analysts determined that it was inappropriate to include these in the significance determination process.

f. Conditional Change in Core Damage Probability

The analysts evaluated the spectrum of seismic initiators to determine the resultant impact on the reliability and availability of mitigating systems affecting the subject performance deficiency. The analysts used the San Onofre 2 & 3 Revision 3P SPAR Model, Change 3.21, created October 2005, to perform the Phase 3 evaluation.

The analysts first created a baseline case by setting the initiating event probability for a LOOP to 1.0 and all other initiating event probabilities in the SPAR model to the house event FALSE, indicating that these initiators would not happen during the seismically induced LOOP. Offsite power was assumed to be non-recoverable following seismic events that break the ceramic insulators (low fragility components) on the transmission lines. Therefore, the analysts set the non-recovery probabilities for offsite power to 1.0. The SPAR model showed the resultant conditional core damage probability as $3.135\text{x } 10^{-4}$, which represented the baseline case that is used in the above equation.

The SPAR Model showed that loss of AFW during an unrecoverable LOOP leads directly to core damage. Therefore, the change in core damage probability is:

$$\Delta CCDP_{SE(a)} = 1.00 - 3.135 \times 10^{-4} = 9.997 \times 10^{-1} \approx 1.00$$

Phase 3 Results:

Considering the factors described above, the total increase in risk, ΔCDF, can be quantified by summing the $\Delta CDF_{SE(a)}$ for each bin as follows:

$$\Delta CDF = \sum_{a=.03g}^{6g} \Delta CDF_{SE(a)}$$

over the range of SE(a).

The total increase in core damage frequency was estimated to be about 2.7×10^{-7} over a 1-year period, for seismic events ranging from 0.03g to 6.0g. Based on the nature of this performance deficiency, the ΔCDF was caused solely by the effects of the deficiency on the plant response to seismic events. As described above, the total ΔCDF indicated that the finding was of very low safety significance (Green).

3. <u>Risk Contribution from Large Early Release Frequency (LERF)</u>

Using Inspection Manual Chapter 0609 Appendix H, the analysts determined that this was a Type A finding (i.e., LERF contributor) for a large dry containment. For PWR plants with large dry containments, only findings related to accident categories ISLOCA and SGTR have the potential to impact LERF. In addition, an important insight from the Individual Plant Evaluation program and other PRAs is that the conditional probability of early containment failure is less than 0.1 for postulated core damage scenarios that leave the RCS at high pressure (>250 psi) at the time of reactor vessel breach. The risk of this finding is not driven by inter-system loss of coolant accidents or steam generator tube ruptures, and the postulated core damage scenarios for this finding would leave the reactor coolant system at high pressure and, thus, early containment failure would be unlikely. Therefore, the analysts concluded that LERF is not a significant contributor to the risk associated with this finding.