UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD.
ARLINGTON, TX 76011-4511

February 4, 2016

Mr. Edward D. Halpin
Senior Vice President
and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT – NRC INTEGRATED INSPECTION REPORT 05000275/2015004 and 05000323/2015004

Dear Mr. Halpin:

On December 31, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Diablo Canyon Power Plant Units 1 and 2. On January 11, 2016, the NRC inspectors discussed the results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented three findings of very low safety significance (Green) in this report. These findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Diablo Canyon Power Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at the Diablo Canyon Power Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public

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Sincerely,

/RA/

Jeremy R. Groom, Chief
Project Branch A
Division of Reactor Projects

Docket Nos. 05000275, 05000323
License Nos. DPR-80, DPR-82

Enclosure:
Inspection Report 05000275/2015004
and 05000323/2015004
w/ Attachments: Supplemental Information
RFI for Inservice Inspection
RFI for Occupational Radiation Safety Inspection

cc w/ enclosure: Electronic Distribution

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Letter to Edward D. Halpin from Jeremy R. Groom dated February 4, 2016

SUBJECT: DIABLO CANYON POWER PLANT – NRC INTEGRATED INSPECTION REPORT 05000275/2015004 and 05000323/2015004

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000275; 05000323
License:	DPR-80; DPR-82
Report:	05000275/2015004; 05000323/2015004
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, CA
Dates:	October 1 through December 31, 2015
Inspectors:	T. Hipschman, Senior Resident Inspector J. Reynoso, Acting Senior Resident Inspector M. Stafford, Project Engineer R. Alexander, Senior Project Engineer J. Drake, Senior Reactor Inspector N. Greene, PhD, Health Physicist J. O'Donnell, CHP, Health Physicist
Approved By:	Jeremy Groom, Chief Project Branch A Division of Reactor Projects

SUMMARY

IR 05000275/2015004, 05000323/2015004; 10/01/2015 – 12/31/2015; Diablo Canyon Power Plant; Fire Protection, Inservice Inspection Activities, Follow-up of Events and Notices of Enforcement Discretion

The inspection activities described in this report were performed between October 1 and December 31, 2015, by the resident inspectors at Diablo Canyon Power Plant and inspectors from the NRC's Region IV office. Three findings of very low safety significance (Green) are documented in this report. All three of these findings involved violations of NRC requirements. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process."

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of Technical Specification 5.4.1.d, "Procedures," for the failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis. Specifically, from August 31 to September 2, 2015, the licensee failed to evaluate the aggregate impact of having three fire doors simultaneously blocked open in adjacent Unit 1 vital battery charger rooms. The licensee implemented immediate corrective actions by assigning a continuous fire watch to the area and documented the issue in the corrective action program as Notification 50826793.

  The failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, the failure to evaluate the aggregate impact of multiple fire system impairments affected the licensee ability to limit the impact of a potential fire. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1–Initial Screening and Characterization of Findings." Because the finding involved fire protection, the inspectors transitioned to IMC 0609, Appendix F "Fire Protection Significance Determination Process." The inspectors characterized the finding using IMC 0609, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," dated September 20, 2013. The finding screened as very low safety significance (Green), per Attachment 1, Question 1.4.3-A since the fire finding category was determined to be fire confinement, due to the fire doors being propped open, and the combustion loading on both sides of the door was determined to be a duration of 30 minutes as documented in licensee calculation M-824, "Controlled Combustion Loading Tracking." In addition, the inspectors determined this finding had a cross-cutting aspect in human performance associated with the teamwork component because the licensee's work groups did not properly communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, the work planners did not properly communicate to the fire protection department that all three fire doors would be open at the same time during battery charger load testing. [H.4] (Section 1R05)

Cornerstone: Mitigating Systems

- Green. The inspectors reviewed a self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Action," for the failure to identify the cause and take corrective action to prevent recurrence of a significant condition adverse to quality impacting both trains of the Unit 1 safety-related residual heat removal (RHR) system. Specifically, the licensee failed to identify a definitive cause and implement corrective actions to prevent recurrent failures of the socket weld for relief valve RHR-1-RV-8708 for both trains of the RHR system. As immediate corrective actions, the licensee installed additional piping supports to mitigate the vibrations at the socket weld and documented this issue in the corrective action program as Notification 50680750.

  The failure to identify the cause of the RHR vibration-induced problems and to take adequate corrective actions to prevent recurrence of the weld failures was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because if left uncorrected, it could lead to a more significant safety concern. Specifically, no additional supports were installed and no actions were taken to reduce or eliminate the vibrations to prevent recurring weld failures, which could affect the availability of the RHR system. The lack of corrective actions to prevent recurrence could leave RHR components and other components physically connected to the system susceptible to future failures. Using Inspection Manual Chapter 0609, Appendix A, the inspectors determined the issue to have very low safety significance (Green) because the performance deficiency, which affected the mitigating systems cornerstone, did not result in a loss of safety function and did not result in an actual loss of function for greater than the technical specification allowed outage time. The licensee entered this into their corrective action program as Notification 50680750. In addition, this finding has a cross-cutting aspect in the human performance area associated with conservative bias decision making component because individuals failed to use decision making practices that emphasize prudent choices over those that are simply allowable. Specifically, the licensee chose to only install a fatigue resistance weld rather than install additional pipe supports as were in the Unit 2 system [H.14]. (Section 1R08.5)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III "Design Control," for the failure to implement design control measures to verify the adequacy of the Unit 1 emergency diesel generators (EDGs) cooling system design to ensure operation of the EDGs under worst-case environmental conditions. Specifically, since initial licensed operations began in 1984, the licensee failed to ensure the Unit 1 EDGs were designed and built to operate under worst-case high wind and temperature conditions. As a result, sustained high winds from specific directions could have impacted EDG radiator performance resulting in the unavailability of the Unit 1 EDGs. Immediate corrective actions included issuing shift orders to the reactor operators to monitor for specific weather conditions (high air temperature, high wind speed and direction) and provide additional room cooling using established procedures, as necessary. The licensee documented the issue in the corrective action program as Notification 50599190.

  The failure to implement design control measures to ensure the emergency diesel generators could perform their design basis function was a performance deficiency. The performance deficiency was more than minor, and is therefore a finding, because it was associated with the design control attribute of the mitigating system cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the performance deficiency resulted in a condition where sustained high winds from specific directions could have impacted EDG radiator performance resulting in the unavailability of the Unit 1 EDGs. The inspectors evaluated the finding using Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process for Findings At Power," dated June 19, 2012. The inspectors determined that a detailed risk evaluation by an NRC senior reactor analyst was required since the finding was associated with a loss of EDG function. The regional senior reactor analyst performed a Phase 3 SDP analysis for the finding. The results of analysis established the incremental conditional core damage probability (ICCDP) was 2.74E-07, less than 1 x 10-6, and therefore the analyst determined that the subject finding was of very low safety significance (Green).

A cross-cutting aspect was not assigned to the finding since the finding did not represent current licensee performance. The condition existed since original construction of the plant. (Section 4OA3)

PLANT STATUS

Units 1 and 2 began the inspection period at full power.

On October 4, 2015, Unit 1, was shut down for a planned refueling outage. On November 7, 2015, Unit 1 returned to operation and began a controlled power ascension; the unit attained full power on November 13, 2015.

On December 11, 2015, Units 1 and 2 experienced heavy influx of debris from ocean swells and carryover from the circulating water system screens which resulted in fouling of the main condensers. In response, both units 1 and 2 reduced power to 25 percent to perform maintenance to removed debris from the circulating water system and condensers. Following maintenance Unit 2 returned to full power on December 14, 2015, and Unit 1 returned to full power on December 15, 2015.

Units 1 and 2 operated at or near full power for the duration of this inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

On December 11, 2015, the inspectors completed an inspection of the station's readiness for seasonal extreme weather conditions. The inspectors reviewed the licensee's adverse weather procedures for high ocean swells and circulating water intake management during the storm season and evaluated the licensee's implementation of these procedures. The inspectors verified that prior to the onset of the storm season, the licensee had corrected weather-related equipment deficiencies identified during the previous storm season.

The inspectors reviewed the licensee's procedures and design information to ensure that the circulating water systems would remain functional when challenged by debris loading due to high ocean swells. The inspectors verified that operator actions described in the licensee's procedures were adequate to maintain readiness of these systems. The inspectors walked down portions of these systems to verify the physical condition of the circulating water system.

These activities constituted one sample of readiness for seasonal adverse weather, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- October 5-9, 2015, Unit 1, reactor vessel refueling level indication system alignment
- December 22-23, 2015, Unit 1, emergency diesel generator air start and turbo air partial alignment

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems were correctly aligned for the existing plant configuration.

These activities constituted two partial system walk-down samples as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

On November 14, 2015, the inspectors performed a complete system walk-down inspection of the Unit 2 emergency diesel generator 2-2. The inspectors reviewed the licensee's procedures and system design information to determine the correct system lineup for the existing plant configuration. The inspectors also reviewed outstanding work orders, open condition reports, in-process design changes, temporary modifications, and other open items tracked by the licensee's operations and engineering departments. The inspectors then visually verified that the system was correctly aligned for the existing plant configuration.

Between November 23 and 24, 2015, the inspectors performed a complete system alignment inspection of the Unit 2 emergency diesel generator 2-2, lube oil, air, and jacket water systems to verify the functional capability of the system.

These activities constituted one complete system walk-down sample, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on four plant areas important to safety:

- October 5, 2015, Unit 1 containment 91 foot elevation
- October 20, 2015, Unit 1, turbine building 104 foot elevation, Area 1-TB-104
- November 24, 2015, Unit 2, emergency diesel generator and radiator fire areas located in turbine building 85 foot and 119 foot elevations
- December 18, 2015, Unit 1 and 2, turbine building 140 foot elevation

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted four quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of Technical Specification 5.4.1.d, "Procedures," for the failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis. Specifically, from August 31 to September 2, 2015, the licensee failed to evaluate the aggregate impact of having three fire doors simultaneously blocked open in adjacent Unit 1 vital battery charger rooms.

Description. On August 31, 2015, the licensee implemented Work Order 64092332 and blocked open three fire barriers (doors) to route test cables for planned load testing of all three Unit 1 vital battery chargers. The testing setup involved running energized electrical cables through three blocked open vital fire doors. The blocked open fire doors affected several fire zones/areas and all three of the Unit 1 vital battery charger rooms. Since the work required fire doors to be blocked open, the licensee issued a fire impairment for each door. Each fire impairment required a one-hour roving fire watch as a compensatory measure.

On September 1, 2015, during a tour of the plant, the inspector observed multiple fire doors blocked open and reviewed the fire impairments to determine if this configuration was adequate. Specifically, the inspectors questioned if the aggregate impact of multiple fire doors open was acceptable with respect to the fire protection system design and safe shutdown analysis. The licensee documented the inspector's concern in Notification 50803608. Because of the inspector questions, the licensee re-evaluated the aggregate impact of multiple blocked open fire doors. The licensee determined that while it was acceptable to have multiple fire doors open, additional compensatory measures were necessary. Instead of a one-hour roving fire watch, the licensee was required to implement a continuous fire watch.

Diablo Canyon Procedure OM8.ID2, "Fire System Impairment," Revision 19, Step 4.8, states, in part, that the fire protection engineer is responsible for reviewing the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis. The inspectors determined the fire protection engineer did not assess the aggregate impact of multiple fire impairments, and a significant contributor to this issue was that work planners did not properly communicate to the fire protection department that all three fire doors would be open at the same time during battery charger load testing.

Analysis. The failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, the failure to evaluate the aggregate impact of multiple fire system impairments affected the licensee ability to limit the impact of a potential fire. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1–Initial Screening and Characterization of Findings." Because the finding involved fire protection, the inspectors transitioned to IMC 0609, Appendix F "Fire Protection Significance Determination Process." The inspectors characterized the finding using IMC 0609, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," dated September 20, 2013. The finding screened as very low safety significance (Green), per Attachment 1, Question 1.4.3-A since the fire finding category was determined to be fire confinement, due to the fire doors being propped open, and the combustion loading on both sides of the door was determined to be a duration of 30 minutes as documented in licensee calculation M-824, "Controlled Combustion Loading Tracking." In addition, the inspectors determined this finding had a cross-cutting aspect in human performance associated with the teamwork component because the licensee's work groups did not properly communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, the work planners did not properly communicate to the fire protection department that all three fire doors would be open at the same time during battery charger load testing. [H.4]

Enforcement. Technical Specification 5.4.1.d, "Procedures," requires, in part, that the licensee establish, implement, and maintain applicable written procedures covering fire protection program implementation. The licensee established Procedure OM8.ID2, "Fire System Impairment," Revision 19, to implement, in part, the requirement of Technical Specification 5.4.1.d. Procedure OM8.ID2, Step 4.8, states that the "Fire protection engineer is responsible for reviewing the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis." Contrary to this requirement, from August 31 to September 2, 2015, the fire protection engineer failed to review the fire impairment list to assess the aggregate impact on the fire protection design and safe shutdown analysis. Specifically, the fire protection engineer failed to evaluate the aggregate impact of multiple fire system impairments involving three simultaneously blocked open fire doors in adjacent Unit 1 vital battery charger rooms. The licensee implemented immediate corrective actions by assigning a continuous fire watch to the area. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as Notification 50826793. (NCV 05000275/2015004-01, "Failure to Properly Evaluate for Aggregate Impact of Fire Impairments)

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On October 16, 2015, the inspectors completed an inspection of underground bunkers susceptible to flooding. The inspectors selected two underground vaults that contained risk-significant or multiple-train cables whose failure could disable risk-significant equipment:

- October 6, 2015, Unit 1, inspection of reactor pressure vessel lower head vault below the 91 foot elevation

- October 15-16, 2015, Unit 1, auxiliary saltwater system underground vault / pull boxes inspection BP 034 and 35

The inspectors observed the material condition of the cables and splices contained in the vaults and looked for evidence of cable degradation due to water intrusion. The inspectors verified that the cables and vaults met design requirements.

These activities constitute completion of two bunker/manhole samples, as defined in Inspection Procedure 71111.06.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

On October 30, 2015, the inspectors completed an inspection of the readiness and availability of the Unit 1 and 2 spent fuel pool risk-significant heat exchangers. The inspectors reviewed the data from a performance test for the heat exchanger. Additionally, the inspectors walked down the heat exchanger to observe its performance and material condition.

These activities constitute completion of one heat sink performance annual review sample, as defined in Inspection Procedure 71111.07.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

The activities described in subsections 1 through 5 below constitute completion of one inservice inspection sample, as defined in Inspection Procedure 71111.08.

.1 Non-destructive Examination (NDE) Activities and Welding Activities

a. Inspection Scope

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION/TYPE
Pressurizer	WIB-378 (Spray line))	Ultrasonic Phased Array
Pressurizer	WIB-378 (Spray line))	Ultrasonic
Pressurizer	WIB-378 (Spray line))	Liquid Penetrant
Auxiliary Feedwater	FW-1-680 FW-1	Liquid Penetrant
Reactor Vessel	Hot Leg Nozzle	Visual (VT-1)
Reactor Vessel	Bottom Bare Metal Visual	Visual (VT-1)

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION/TYPE
CVCS	CVCS-1-9, weld 2,3	Liquid Penetrant
Auxiliary Feedwater	FW-1-680 FW-2,3,4	Liquid Penetrant
Reactor Vessel	Upper Bare Metal Visual	Visual (VT-1)

During the review and observation of each examination, the inspectors observed whether activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Code requirements and applicable procedures. The inspectors also reviewed the qualifications of all nondestructive examination technicians performing the inspections to determine whether they were current.

The inspectors observed the following welding activities:

SYSTEM	WELD IDENTIFICATION	TYPE

Chemical and Volume Control System	CVCS-1-9 FW-1	Gas Tungsten Arc Welding
Auxiliary Feedwater	FW-1-680 FW-1	Gas Tungsten Arc Welding

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	TYPE
Chemical and Volume Control System	CVCS-1-9 FW-2,3	Gas Tungsten Arc Welding
Auxiliary Feedwater	FW-1-680 FW-2,3,4	Gas Tungsten Arc Welding

The inspectors reviewed whether the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code Section IX requirements. The inspectors also determined whether essential variables were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspectors reviewed the results of the licensee's bare metal visual inspection of the Reactor Vessel Upper Head Penetrations to determine whether the licensee identified any evidence of boric acid challenging the structural integrity of the reactor head components and attachments. The inspectors also verified that the required inspection coverage was achieved and limitations were properly recorded. The inspectors reviewed whether the personnel performing the inspection were certified examiners to their respective nondestructive examination method.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's implementation of its boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walk-down as specified in Procedure ER1.ID2, "Boric Acid Corrosion Control Program," Revision 7. The inspectors reviewed whether the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components, and whether engineering evaluations used corrosion rates applicable to the affected components and properly assessed the effects of corrosion-induced wastage on structural or pressure boundary integrity. The inspectors observed whether corrective actions taken were consistent with the ASME Code, and 10 CFR Part 50, Appendix B requirements.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the steam generator tube eddy current (ECT) examination scope and expansion criteria to determine whether these criteria met technical specification requirements, Electric Power Research Institute (EPRI) guidelines, and commitments made to the NRC. The inspectors also reviewed whether the ECT inspection scope included areas of degradations that were known to represent potential eddy current test challenges such as the top of tube sheet, tube support plates, and U-bends. The inspectors confirmed that no repairs were required at the time of the inspection. The scope of the licensee's ECT examinations included:

- 100 percent of the in-service tubes inspected full length with bobbin probes
- All potential AVB (anti vibration bar) wear sites inspected by bobbin probe

Plus Point inspections were conducted on the following:

- 100 percent of abnormal conditions detected by bobbin probe
- 100 percent of dents and dings $\geq$ 1 volt detected by bobbin probe and not Plus Point probe inspected in 1R16
- 100 percent of proximity indications

The following tube degradation mechanisms were identified:

- AVB tube wear

The inspectors observed portions of the eddy current testing being performed to determine whether: (1) the appropriate probes were used for identifying the expected types of degradation, (2) calibration requirements were followed, and (3) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of the site-specific qualifications for the techniques being used and reviewed whether eddy current test data analyses were adequately performed per EPRI and site specific guidelines.

Sludge lancing and foreign object search and retrieval (FOSAR) activities were conducted during Refueling Outage 1R19. Sludge lancing activities removed a total of approximately 8 pounds of sludge for all four steam generators. FOSAR examination included an in-bundle inspection of the center 10 columns of the hot leg and cold leg top-of-tube support region, and columns 20, 40, 60, 80, and 100 in both hot and cold legs. The examination also included 100 percent of the trough and outer periphery tubes.

Finally, the inspectors reviewed selected eddy current test data to verify that the analytical techniques used were adequate.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems (71111.08)

a. Inspection scope

The inspectors reviewed 18 notifications (condition reports) which dealt with inservice inspection activities and found the corrective actions were appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee has an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry operating experience. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

Introduction. The inspectors reviewed a self-revealing Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," associated with the licensee's failure to identify the cause and take corrective action to prevent recurrence of a significant condition adverse to quality impacting both safety-related residual heat removal (RHR) systems in Unit 1. Specifically, the licensee failed to identify a definitive cause and implement corrective actions to prevent recurrent failures of the socket weld for relief valve RHR-1-RV-8708 for both trains of the RHR system.

Description. On June 25, 2013, during a walk down of the Unit 1 Containment, maintenance personnel noted an accumulation of boric acid on the inlet pipe to valve RHR-1-RV-8708. The problem was reported in Notification 50570623. Subsequent clean-up of the boric acid accumulation revealed an active leak of 3 drops per minute (DPM) from a circumferential crack on the socket weld. The active boric acid leak was located on the common header from the RHR pump discharge to RCS Hot Legs 1 and 2. The active boric acid leak could not be isolated from the rest of the system. Both trains of the RHR system were declared inoperable and Technical Specification 3.0.3 was entered on June 25, 2013, at 9:58 p.m. The technical specification required a shutdown of Unit 1 within 7 hours. The condition also required a 4-hour notification to the NRC. The licensee did not classify this initial failure of the weld as a significant condition adverse to quality.

Procedure OM4.1014, “Notification Review Team,” Revision 25, Section 3.21 defined a significant condition adverse to quality as a condition that, if left uncorrected, could seriously affect the ability to operate the plant in a safe manner or will require a major effort to restore capability to perform specific functions. The failure of this weld resulted in the licensee declaring both trains of residual heat removal inoperable and required the plant to be shutdown in order to repair the weld; therefore, the failure of the weld should have been classified as a significant condition adverse to quality. Title 10 CFR Part 50 Appendix B, Criterion XVI states in part, “In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.”

As part of the apparent cause evaluation, the licensee performed an inspection of the weld using the inservice inspection group and a metallurgist from Pacific Gas and Electric Applied Technology Services (ATS). The inspection identified that the source of the leak was a circumferential crack, approximately 110 degrees in length, in the socket weld at the half-coupling (sockolet) on line 1-S1-3/4” (inlet side of RV-8708). The observed crack was circumferentially oriented and predominately in the middle of the weld face, with one end of the crack tapering to the socket-side toe of the weld. Based on previous experience, the licensee determined the crack was indicative of a high cycle fatigue failure, but the weld was not cutout and failure analysis was not performed to identify the definitive cause of the failed weld.

Two snubbers (22-59SL & 22-60SL) had been removed from the piping in 1988, during the Snubber Reduction Program. The licensee determined that snubber removal probably contributed to this event. The equivalent piping on Unit 2 is generally similar in geometry, however it has two rigid restraints (one lateral and one vertical), and one lateral snubber on the discharge side of the relief valve.

The licensee assume that the only potential source of cyclical loading on the weld was from vibrations cause by components within the RHR system, but did not take additional vibrational data with the plant in normal operating conditions. The socket weld was replaced on June 27, 2013. The new weld used a 2:1 leg ratio fillet weld, which has improved fatigue resistance over the original equal-leg socket weld. This weld configuration is recommended by EPRI, and is used by the industry and Diablo Canyon Power Plant where enhanced fatigue resistance is required.

Based on a comparison with the Unit 2 piping system, the licensee determined that new rigid lateral support(s) on Line 1184-1 were necessary to reduce piping vibration. However, the licensee decided to defer installation of the supports until refueling outage 1R19. The licensee assumed this was acceptable based on the new fatigue-resistant weld profile and the fact that it took approximately 25 years to fail the first time.

Because the licensee elected to perform an apparent cause evaluation rather than a root cause evaluation, no definitive root cause for the weld failure was identified, and no corrective actions to prevent recurrence (as defined in the licensee’s corrective action program) were taken. Upon the second failure of the relief valve RHR-1-RV-8708 weld (in December 2014), the licensee initiated Notification 50680750 to evaluate the failure and initiated a root cause evaluation to identify the cause of the residual heat removal system vibrations.

Analysis. The failure to identify the cause of the failed weld for Relief Valve RHR-1-RV-8708 and to take corrective actions to prevent recurrence of a significant condition adverse to quality is a performance deficiency. The performance deficiency is more than minor because if left uncorrected, it could lead to a more significant safety concern. Specifically, although individual actions were taken to address the failure caused by vibrations, the cause of the failure was not identified nor were adequate actions taken to prevent recurrence of the weld failure. The lack of corrective actions to prevent recurrence could leave RHR components and other components physically connected to the system susceptible to future failures. Using Inspection Manual Chapter 0609, Appendix A, the inspectors determined the issue to have very low safety significance (Green) because the finding, which affected the mitigating systems cornerstone, did not result in an actual loss of system function for greater than its technical specification allowed outage time.

This finding has a cross-cutting aspect in the human performance area of Conservative Bias decision making because individuals failed to use decision making-practices that emphasize prudent choices over those that are simply allowable. Specifically, the licensee chose to only install a fatigue resistance weld rather than install additional pipe supports as were in the Unit 2 system. [H.14].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that for a significant condition adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, between June 2013 and October 2015, the licensee failed to assure that the cause of a significant condition adverse to quality was determined and failed to implement corrective actions to preclude repetition. Specifically, the licensee failed to determine that the vibration-induced failure of valve RHR-1-RV-8708 was significant and failed to identify a definitive cause for the failure and take corrective actions to prevent recurrence of the failure. After the second failure of valve RHR-1-RV-8708 in December 2014, the licensee installed additional piping supports to mitigate the vibrations at the weld. The licensee has planned additional actions to determine the cause of the vibrations and mitigate them, but vibration levels remain elevated. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s corrective action program as Notification 50680750. (NCV 05000275/2015004-02, “Failure to Identify a Cause and Implement Actions to Prevent Recurrence of a Significant Condition Adverse to Quality)

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On December 16, 2015, the inspectors observed simulator training for an operating crew. The inspectors assessed the performance of the operators and the evaluators’ critique of their performance. The inspectors also assessed the modeling and performance of the simulator during the requalification activities.

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

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

The inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity or risk due to startup activities. The inspectors observed the operators' performance of the following activities:

- October 3-6, 2015, Unit 1, control room observations in preparations for refueling outage including plant shutdown, cooldown, reactor coolant drain down for core offloading

- October 11-12, 2015, Unit 1, reduced inventory for reactor coolant system vacuum fill

- November 6, 2015, plant start-up to criticality

In addition, the inspectors assessed the operators' adherence to plant procedures, including conduct of operations procedure and other operations department policies.

These activities constitute completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed one instance of degraded performance or condition of safety-related structures, systems, and components (SSCs):

- November 10, 2015, Unit 2, 480 volt vital switchgear room ventilation

The inspectors reviewed the extent of condition of possible common cause SSC failures and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the SSCs. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constituted completion of one maintenance effectiveness sample, as defined in Inspection Procedure 71111.12.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed two risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- October 28, 2015, Unit 1 and 2, plant protected equipment in the 4 kV, emergency diesel generator rooms, 480 volt, and component cooling water rooms

- October 30-31, 2015, reactor coolant fill and venting activity including vacuum refill of the reactor coolant from a reduce inventory / mid-loop condition

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the result of the assessments.

Additionally, on November 12, 2015, the inspectors also observed portions of one emergent work activity that had the potential to cause an initiating event or to affect the functional capability of mitigating systems:

- November 12, 2015, Unit 2, 480 volt vital switchgear room ventilation

The inspectors verified that the licensee appropriately developed and followed a work plan for these activities. The inspectors verified that the licensee took precautions to minimize the impact of the work activities on unaffected structures, systems, and components (SSCs).

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

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed three operability determinations that the licensee performed for degraded or nonconforming structures, systems, or components (SSCs):

- November 11-12, 2015, Unit 1, operability determination of increased flow into the reactor coolant drain tank

- November 13, 2015, Unit 2, operability determination of 480 volt vital switchgear room ventilation

- December 11-15, 2015, Unit 1 and 2, operability determination of operator work around conditions to compensate for degraded or non-conforming conditions

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded SSC to be operable, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability of the degraded SSC.

The inspectors reviewed operator actions taken or planned to compensate for degraded or nonconforming conditions. The inspectors verified that the licensee effectively managed these operator workarounds to prevent adverse effects on the function of mitigating systems and to minimize their impact on the operators' ability to implement abnormal and emergency operating procedures.

These activities constitute completion of three operability and functionality review samples, which included one operator work-around sample, as defined in Inspection Procedure 71111.15.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed two post-maintenance testing activities that affected risk-significant structures, systems, or components (SSCs):

- October 19, 2015, Unit 1, containment isolation valve testing of pressurizer relief tank isolation valves RCS 8034 A/B following air operator calibration description

- November 3-4, 2015, Unit 1, seal water return isolation valve, CVCS-8100, post maintenance testing

The inspectors reviewed licensing- and design-basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

These activities constitute completion of two post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

During the station's refueling outage that concluded on November 4, 2015, the inspectors evaluated the licensee's outage activities. The inspectors verified that the licensee considered risk in developing and implementing the outage plan, appropriately managed personnel fatigue, and developed mitigation strategies for losses of key safety functions. This verification included the following:

- Review of the licensee's outage plan prior to the outage
- Review and verification of the licensee's fatigue management activities
- Monitoring of shut-down and cool-down activities
- Verification that the licensee maintained defense-in-depth during outage activities
- Observation and review of reduced-inventory and mid-loop activities
- Observation and review of fuel handling activities
- Monitoring of heat-up and startup activities

These activities constitute completion of one refueling outage sample, as defined in Inspection Procedure 71111.20.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed five risk-significant surveillance tests and reviewed test results to verify that these tests adequately demonstrated that the structures, systems, and components (SSCs) were capable of performing their safety functions:

In-service tests:

- October 5, 2015, Unit 1, containment spray pump 1-1, comprehensive in-service pump testing

Containment isolation valve surveillance tests:

- October 19, 2015, Unit 1, penetration 76B containment isolation valve leak testing
- November 4, 2015, Unit 1, containment isolation valve leak testing of seal water return valve CVCS-1-8100

Other surveillance tests:

- November 5, 2015, Unit 1, control rod drop measurement and rod position functional testing
- November 17, 2015, Unit 1, emergency diesel generator 1-3 testing

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected SSCs following testing.

These activities constitute completion of five surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors assessed the licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities. The inspectors assessed the licensee's implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures. The inspectors walked down various portions of the plant and performed independent radiation dose rate measurements. The inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors reviewed licensee performance in the following areas:

- The hazard assessment program, including a review of the licensee's evaluations of changes in plant operations and radiological surveys to detect dose rates, airborne radioactivity, and surface contamination levels

- Instructions and notices to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions

- Programs and processes for control of sealed sources and release of potentially contaminated material from the radiologically controlled area, including survey performance, instrument sensitivity, release criteria, procedural guidance, and sealed source accountability

- Radiological hazards control and work coverage, including the adequacy of surveys, radiation protection job coverage and contamination controls, the use of electronic dosimeters in high noise areas, dosimetry placement, airborne radioactivity monitoring, controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools, and posting and physical controls for high radiation areas and very high radiation areas

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

- Audits, self-assessments, and corrective action documents related to radiological hazard assessment and exposure controls since the last inspection

These activities constitute completion of one sample of radiological hazard assessment and exposure controls as defined in Inspection Procedure 71124.01.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

The inspectors evaluated whether the licensee controlled in-plant airborne radioactivity concentrations consistent with ALARA principles and that the use of respiratory protection devices did not pose an undue risk to the wearer. During the inspection, the inspectors interviewed licensee personnel, walked down various portions of the plant, and reviewed licensee performance in the following areas:

- The licensee's use, when applicable, of ventilation systems as part of its engineering controls
- The licensee's respiratory protection program for use, storage, maintenance, and quality assurance of NIOSH certified equipment, qualification and training of personnel, and user performance
- The licensee's capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions, status of SCBA staged and ready for use in the plant and associated surveillance records, and personnel qualification and training
- Audits, self-assessments, and corrective action documents related to in-plant airborne radioactivity control and mitigation since the last inspection

These activities constitute completion of one sample of in-plant airborne radioactivity control and mitigation as defined in Inspection Procedure 71124.03.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity (BI01)

a. Inspection Scope

The inspectors reviewed the licensee's reactor coolant system chemistry sample analyses for the period of October 2014 through October 2015 to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the reactor coolant system specific activity performance indicator for Units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Reactor Coolant System Identified Leakage (BI02)

a. Inspection Scope

The inspectors reviewed the licensee's records of reactor coolant system (RCS) identified leakage for the period of October 2014 through October 2015 to verify the accuracy and completeness of the reported data. The inspectors observed the performance of RCS leakage surveillance procedure on February 4, 2015. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the reactor coolant system leakage performance indicator for Units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors verified that there were no unplanned exposures or losses of radiological control over locked high radiation areas and very high radiation areas during the period of July 1, 2014 to September 30, 2015. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 mrem. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the occupational exposure control effectiveness performance indicator as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.4 Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed corrective action program records for liquid or gaseous effluent releases that occurred between July 1, 2014 and September 30, 2015, and were reported to the NRC to verify the performance indicator data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the radiological effluent technical specifications (RETS)/offsite dose calculation manual (ODCM) radiological effluent occurrences performance indicator as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee's corrective action program and periodically attended the licensee's condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee's problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors reviewed the licensee's corrective action program, performance indicators, system health reports, and other documentation to identify trends that might indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address identified adverse trends.

To verify that the licensee was taking corrective actions to address identified adverse trends that might indicate the existence of a more significant safety issue, the inspectors reviewed corrective action program documentation associated with the following licensee-identified trends:

- A negative trend identified involving reactor cavity seal leakage (Notifications 50817738 and 50818639)

- A potential adverse trend in foreign material exclusion trends (Notifications 50695123 and 50802021)

The specific documents reviewed during this trend review are listed in the attachment. These activities constitute completion of one semiannual trend review sample, as defined in Inspection Procedure 71152.

b. Observations and Assessments

The inspectors' review of the trends identified above produced the following observations and assessments:

- For negative trend in reactor cavity seal leakage, the licensee chartered a root cause team to evaluate the repeated leakage of the reactor cavity seal during the Unit 1 and 2 refueling outages. The repeated seal leakage resulted in the replacement of excore nuclear instrumentation, increased outage dose to workers, and boric acid leakage from the refueling cavity onto the reactor vessel. Previous corrective actions to modify the existing cavity seal was not fully successful and had limited results. The inspector discussed with the root cause evaluation team the technical causes and failure modes of reactor cavity seal leakage from historical data and current information gathered during recent Unit 1 refueling outage 1R19.

  The inspectors assessed the licensee's response to the cavity seal leakage, the impact to the reactor vessel from the exposure to boric acid, and the ongoing root cause evaluation to determine the cause of this negative trend. The inspectors determined the licensee had assigned the appropriate level of evaluation to this trend to determine the root cause and implement the necessary corrective actions to prevent recurrence.

- For the negative trend related to foreign material exclusion events, the licensee performed an evaluation and a quick hit self-assessment under Notifications 50695123, 50802021, and 50799008 due to online and outage events involving a loss of foreign material exclusion (FME) controls. The evaluation included a review of seventy-eight FME events from September 1, 2013 through June 9, 2015.

  The license performed a quick hit self-assessment to identify areas of potential weaknesses in the FME program and to identify potential focus areas for improvements.

  The self-assessment concluded workers understand the FME program requirements and goals but do not always apply the human performance tools necessary in determining risk with the work activity.

The inspectors concluded that the licensee had conducted an appropriate evaluation to determine areas of weaknesses for the FME events and implemented appropriate bulletins to increase workers awareness.

On November 6, 2015, during a Mode 3 walkdown of the Unit 1 containment lower elevations and near the containment sump, the inspectors observed several loose items and debris left in containment. The inspectors identified tape, adhesive labels, and remnants of a plastic bootie, not properly identified or controlled prior to Mode 4 entry. The inspectors verified the licensee removed the debris from the containment.

The inspectors determined the licensee failed to apply appropriate measures to ensure containment cleanliness and loose debris be controlled and removed during the Unit 1 refueling outage 1R19, in accordance with station procedures STP M-45C, “Outage Management Containment Inspection,” and AD4.ID9, “Containment Housekeeping and Material Controls.” Specifically, the foreign material total square footage was very small compared to the available recirculation sump screen surface area margin. The licensee appropriately documented this condition in the corrective action program as Notifications 50817774, 50817820, and 50817822. The inspectors determined that this performance deficiency was a minor violation of 10 CFR Part 50, Appendix B, Criterion V, “Procedures.”

c. Findings

No findings were identified.

.3 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected one issue for an in-depth follow-up:

- Notification 50812917, which documented two fuel assemblies removed during the 1R19 refueling outage that had unusual corrosion patterns on several grids which were face-adjacent to the core baffle. Specifically, the corrosion buildup on those fuel bundle straps were less than seen on other fuel bundles removed from the core.

  The inspectors reviewed the licensee’s evaluation of the current notification, select evaluations of fuel inspections conducted during previous refueling outages, and the licensee’s evaluation of contemporary operating experience relative to the phenomena known as “baffle jetting” (as described in NRC Information Notice 82-27, “Fuel Rod Degradation Resulting from Baffle Water-Jet Impingement”).

The inspectors assessed the licensee’s problem identification threshold, cause analyses, extent of condition reviews, and compensatory actions for the selected issue. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate.

These activities constitute completion of one annual follow-up sample as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

(Closed) LER 05000275/1-2013-009-01: Unanalyzed Condition Affecting Emergency Diesel Generators

a. Inspection Scope

The inspectors verified the accuracy and completeness of the Licensee Event Report (LER) and the appropriateness of the licensee's corrective actions. The inspectors reviewed the immediate corrective actions that developed compensatory measures to allow the Unit 1 emergency diesel generators (EDGs) to remain operable with sustained winds up to 80 miles per hour (mph) with ambient air temperature up to 97 degrees Fahrenheit.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III "Design Control." The licensee's failure to implement design control measures to verify the adequacy of the Unit 1 emergency diesel generators (EDGs) cooling system design to ensure operation of the EDGs under worst-case environmental conditions was a performance deficiency. Specifically, since initial licensed operation in 1984, the licensee failed to ensure the Unit 1 EDGs were designed and built to operate under worst-case high wind and temperature conditions. As a result, sustained high winds from specific directions could have impacted EDG radiator performance resulting in the unavailability of the Unit 1 EDGs.

Description. On December 14, 2013, the licensee determined that sustained high winds could result in not having adequate engine heat removal to support continuous operation of the Unit 1 EDGs. The licensee initiated an evaluation of its EDGs because the NRC requested all licensees determine the impact to air-cooled EDGs by winds blowing in the opposite direction of the radiator cooling air. The licensee's engineers identified a vulnerability to extreme weather. A combination of sustained high winds over 60 miles per hour and ambient air temperatures exceeding 97 degrees Fahrenheit could reduce cooling capacity such that the emergency diesel generators would exceed design limits.

Unit 1 has three emergency diesel generators that provide vital emergency AC power to three electrical buses to mitigate the consequences of a design basis accident (DBA) whenever normal or offsite power sources are unavailable. The EDGs are designed to function so that a single failure of any EDG will not jeopardize the capability of the remaining EDGs to start and provide power to operate the shutdown systems required to mitigate any DBA condition. The EDGs use engine-driven fans that provide cooling air to the EDG radiators. The radiator fan draws air through the radiator, maintaining jacket water temperature and, in turn, maintaining lubricating oil temperature. Jacket water to the aftercooler also affects combustion air temperature. The radiator fan also draws ambient air through the engine compartment to cool the equipment housed within it.

Inability to maintain adequate radiator airflow will result in a rise in EDG jacket water temperature, higher component temperatures in the engine compartments, derating of the engine due to increased combustion air temperature, higher lubricating oil temperatures, and high cylinder jacket temperatures. This could result in a failure of the EDGs to perform their safety function. Due to the physical orientation and layout of the site, the Unit 2 EDGs were determined not to be susceptible to effects of the extreme weather conditions as described in the LER.

Upon identification of this condition, Operations shift orders were issued that require, when conditions warrant, the implementation of existing procedural guidance to open plant doors to allow additional air flow, shown to provide adequate emergency diesel generator cooling to support continuous operation of the Unit 1 emergency diesel generators. An apparent cause evaluation concluded that the licensee did not have a design process requirement in place to evaluate GDC-2 design criteria relative to SSC functional requirements other than for structural integrity issues. The licensee revised plant procedures to include this requirement and PG&E will issue a design change to permanently modify the plant to resolve the wind issue.

Analysis. The failure to implement design control measures to ensure the emergency diesel generators could perform their design basis function was a performance deficiency. The performance deficiency was more than minor and is therefore a finding because it was associated with the design control attribute of the mitigating system cornerstone, and affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events. Specifically, the performance deficiency resulted in a condition where sustained high winds from specific directions could have impacted EDG radiator performance resulting in the unavailability of the Unit 1 EDGs. Using Table 2 of Inspection Manual Chapter (IMC) 0609.04, “Significance Determination Process Initial Characterization of Findings,” dated June 19, 2012, the inspectors concluded that the finding affected the mitigating system cornerstone. The inspectors evaluated the finding using Exhibit 2 of IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” dated June 19, 2012. The inspectors determined that a detailed risk evaluation by an NRC senior reactor analyst was required since the finding was associated with a loss of EDG function. The regional senior reactor analyst performed a Phase 3 significance determination process (SDP) analysis for the finding as follows:

The licensee performed a realistic evaluation of emergency diesel generator temperatures, given varying wind and ambient temperature combinations. Specifically, the licensee computed the emergency diesel generator room temperature, jacket water radiator inlet temperature, and the jacket water radiator outlet temperature. The design limits for the analysis were room temperatures less than 120 degrees, inlet temperatures less than 195 degrees, and outlet temperatures less than 160 degrees. The first two values represented points, if exceeded, at which the emergency diesel generator was assumed to fail. The last value, if exceed, was assumed to cause a derating of the diesel generator as opposed to direct failure. The analyst bounded the analysis by placing all postulated weather conditions resulting in diesel failure into the following three bins:

1. Winds of 50 mph or greater with ambient temperatures of 90 degrees or greater

2. Winds of 70 mph or greater with ambient temperatures of 80 degrees or greater

3. Winds of 80 mph or greater with ambient temperatures of 70 degrees or greater

The analyst determined that the subject performance deficiency impacted plant risk from initial reactor startup through December 2013. Therefore, in accordance with the Risk Assessment of Operational Events Handbook, Volume 1, “Internal Events,” Revision 2, Section 2.6, “Exposure Time Greater than 1 Year,” the maximum exposure time (EXP) was set to the 1-year assessment period.

In order for the emergency diesel generators to be of importance, a loss of offsite power must occur. The analyst noted that the frequency of a loss of offsite power (λ_{LOOP}) was 3.59×10^{-2}/year from the Standardized Plant Analysis Risk (SPAR) model for Diablo Canyon Power Plant, Units 1 and 2.

Upon the occurrence of a loss of offsite power, the analyst assumed that high winds would need to occur within 24 hours to have any major impact on risk. Using the Severe Weather Database (1950-2011) from the National Weather Service, the analyst reviewed wind data over a 29-year period and found 47 occurrences of winds greater than 40 mph, four occurrences of winds over 60 mph, and no occurrences of winds over 80 mph. Using a noninformative prior distribution, the analyst calculated the following frequencies of exceedance (λ_{Wind}):

Minimum Wind Speed (mph)	Occurrences	Frequency of Exceedance (per day)
40	47	4.44E-03
60	4	3.78E-04
80	0.5	4.72E-05

The probability of exceeding the minimum wind of interest (P_{Wind}) is the λ_{Wind} multiplied by the 24 hour (1 day) period of the analysis.

The analyst determined that the best available information regarding temperatures onsite was the meteorological tower data. The analyst reviewed the following information provided by the licensee to determine the probability of exceedance (P_{Temp}):

Ambient Temperature (at EDG Inlet)	Met Tower Temperature	Probability of Exceedance
60	50	9.15E-01
70	60	1.47E-01
80	70	1.50E-02
90	80	2.20E-03
100	90	4.59E-04

Using the SPAR model, the analyst quantified the conditional core damage probability (CCDP) for a postulated loss of offsite power with the failure of all emergency diesel generators on Unit 1. The result was 3.41×10^{-1}. Therefore, assuming that, for the sake of this analysis, wind speed and temperature are independent variables, the incremental conditional core damage probability (ICCDP) for any given bin can be calculated as follows:

$$ICCDP = \lambda_{LOOP} * P_{Wind} * P_{Temp} * EXP$$

Bin	Wind Speed (mph)	Wind Probability	Temperature (degrees F)	Temperature Probability	ICCDP
1	50	4.44E-03	90	2.20E-03	1.20E-07
2	70	3.78E-04	80	1.50E-02	6.94E-08
3	80	4.72E-05	70	1.47E-01	8.50E-08
				Total ICCDP:	2.74E-07

Given that the internal events change in core damage frequency (equivalent to the ICCDP) is less than 1 x 10^{-6}, the analyst determined that the subject finding was of very low safety significance (Green).

Contributions from External Events (Fire, Flooding, Seismic and High Winds): In accordance with the guidance in NRC Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for findings At-Power,” the analyst performed an evaluation of the external event risk contribution because the internal events detailed risk evaluation results were greater than 1.0 x 10^{-7}. The analyst determined that the plant response to the external events fire, flooding and seismic would not be greatly impacted by this performance deficiency because none of the initiators would impact the probability of high winds and high temperatures occurring at the same time. Therefore, only high winds related initiators were considered to be potentially impacted by this performance deficiency.

For high winds initiators to be affected by the performance deficiency, winds must be strong enough to result in a loss of offsite power. As stated previously, using the Severe Weather Database (1950-2011) from the National Weather Service there were no winds greater than 80 mph reported within a 100 kilometer radius of the plant over the 29 years that strong winds were documented. Additionally, using SeverePlot output from January 1, 1950 to December 31, 2006, there were no F2 or stronger tornadoes within a 100 km radius of the plant over the 56-year period of the data. Traditionally, analysts have not correlated F0 and F1 tornadoes with winds that could cause widespread loss of offsite power. By definition, F1 tornado damage is limited to peeling off residential roof surfaces, breaking windows and pushing moving automobiles off roads, and not the more severe damage that would be associated with a complete loss of offsite power. Therefore, the analyst determined qualitatively that the impact of this performance deficiency on plant response to high winds external initiators was negligible.

Potential Risk Contribution from Large, Early Release Frequency: In accordance with the guidance in NRC Inspection Manual Chapter 0609, Appendix H, “Containment Integrity Significance Determination Process,” this finding would not involve a significant increase in risk of a large, early release of radiation because Diablo Canyon Power Plant, Unit 1, has a large, dry containment and the dominant sequences contributing to the change in the core damage frequency did not involve either a steam generator tube rupture or an inter-system loss of coolant accident.

A cross-cutting aspect was not assigned to the finding since the finding did not represent current licensee performance. The condition existed since original construction of the plant.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that measures shall be established to assure the design basis for those SSCs to which this appendix applies are correctly translated into specifications and drawings and that the design control measures shall provide for verifying or checking the adequacy of design. FSARU Section 3.1.2, “Criterion 2-Design Basis for Protection against Natural Phenomena,” states, in part, that SSCs important to safety shall be designed to withstand effects of natural phenomena such as hurricanes without loss of capability to perform their safety function. FSARU Section 2.3.2.2.2, “Maximum ambient Temperature,” states that the maximum design temperature applied to the EDG radiator performance is 97 degrees Fahrenheit. FSARU Section 2.3.1.3, “Maximum Wind Speed,” states, in part, that the maximum wind assumed to act on the EDG radiators is 80 mph. Contrary to the above, from initial licensed operations to December 14, 2013, for quality-related components associated with Unit 1 emergency diesel generators to which 10 CFR Part 50, Appendix B applies, the licensee failed to assure the design basis was correctly translated into specifications and drawings. Specifically, the specifications for the Unit 1 EDGs air cooling systems failed to ensure that the design basis for the Unit 1 EDGs, as specified in FSARU Sections 3.1.2 were not adversely impacted by sustained high temperature and winds specified in FSARU Section 2.3.2.2.2 and 2.3.1.3. The licensee entered the issue into their corrective action program as Notification 50599190; revised plant procedures to include improve design reviews, as well as, issued a design change to permanently modify the plant to resolve the issue. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s corrective action program as Notification 50599190. (NCV 05000275/2015004-03, “Failure to Design the Emergency Diesel Generators to operate under Worst Case Environmental Conditions)

These activities constitute completion of one event follow-up sample, as defined in Inspection Procedure 71153.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 15, 2015, the inspectors presented the inservice inspection results to Barry Allen, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On October 15, 2015, the inspectors presented the radiation safety inspection results to Mr. E. Halpin, Senior Vice-President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On January 11, 2016, the resident inspectors presented the inspection results to Mr. E. Halpin, Senior Vice-President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

<u>SUPPLEMENTAL INFORMATION</u>

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>

B. Allen, Vice President Nuclear Services
J. Arhar, Advising Engineer, Senior
T. Baldwin, Director, Nuclear Site Services
D. Evans, Director, Security & Emergency Services
R. Gagne, Supervisor, Radiation Protection
P. Gerfen, Director of Operation Services
M. Ginn, Manager, Nuclear Emergency Planning
D. Gonzalez, Supervisor, Nuclear Engineering
E. Halpin, Sr. Vice President, Chief Nuclear Officer
H. Hamzehee, Manager, Regulatory Services
A. Heffner, NRC Interface, Regulatory Services
B. Highland, Foreman, Radiation Protection
J. Hill, Nuclear Lead ISI, NDE Specialist
J. Hinds, Director, Quality Verification
K. Hinrichsen, Foreman, Radiation Protection
L. Hopson, Assistant Director, Nuclear Maintenance
T. Irving, Manager, Radiation Protection
J. MacIntyre, Director of Equipment Reliability
M. McCoy, NRC Interface, Regulatory Services
L. Million, General Foreman, Radiation Protection
J. Morris, Senior Advising Engineer
C. Neary, Nuclear Advising Engineer, Senior
J. Nimick, Station Director
A. Peck, Director, Nuclear Engineering
R. Rogers, General Foreman and ALARA Supervisor, Radiation Protection
L. Sewell, Nuclear Radiation Protection Engineer
R. Simmons, Manager, Nuclear Maintenance
P. Soenen, Manager, Nuclear Regulatory Services
A. Warwick, Supervisor, Emergency Planning
J. Welsch, Site Vice President
D. Wilson, Lead ISI Inspector/NDE Specialist
M. Wright, Nuclear Engineering, Manager

<u>NRC Personnel</u>

D. Loveless, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000275/2015004-01	NCV	Failure to Properly Evaluate for Aggregate Impact of Fire Impairments (Section 1R05)
05000275/2015004-02	NCV	Failure to Identify a Cause and Implement Actions to Prevent Recurrence of a Significant Condition Adverse to Quality (Section 1R08.5)
05000275/2015004-03	NCV	Failure to Design the Emergency Diesel Generators to operate under Worst Case Environmental Conditions (Section 4OA3)

Closed

05000275/1-2013-009-01	LER	Unanalyzed Condition Affecting Emergency Diesel Generators (Section 4OA3)

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
OP O-28	Intake Management	18
CP M-16	Severe Weather	7
MA1.ID23	Review of Intake Preparedness for High Debris Loading Event	3
ENV.EM2	Ocean Jellyfish Influx Monitoring	1

Notifications

50824808	50698209

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
OP A-2:X	RVRLIS Alignment for Refueling Outages	9
MP I-2.28	Activation and Deactivation of the Rx Vessel Refueling Level Indication System (RVRLIS)	26
OP A-2:II	Reactor Vessel – Draining the RCS to the Vessel Flange – With Fuel in Vessel	26
OP J-6B:II-A	Diesel Generator 2-2 – Alignment Checklist	0
OP1.DC20	Sealed Components	20

OP K-10	Systems Requiring Sealed Component Checklists	38
OP J-6B:II	Diesel Generator 2-2 – Make Available	30
OP J-6B:III-A	Diesel Generator 1-3 – Alignment Checklist	0

Notifications

50708572	50689137	50821650	50640977	50593705
50820595	50812333	50815619	50708060	50803175
50703699				

Work Orders

64071190	60083058

Drawings

Number	Title	Revision
107721	Jacket Water Cooling System and Lube Oil System, Unit 2	58
108021	Combustion Air and Exhaust System 2-2	44
663082	Equipment Arrangement	2
108021	Engine Fuel Oil System 2-2	6
500852	Piping and Mechanical Plans at 85 and 107 foot elevation	11

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
AD4.ID3	SISIP Housekeeping Activities	13
OM8.ID4	Control of Flammable and Combustible Materials	23
OM8.ID2	Fire System Impairment	19
STP M-70D	Inspection of Fire Barriers, Rated Enclosures, Credited Cable Tray Fire Stops, and Equipment Hatches	18
PEP 18-02	Firewater Hose Station Flow Test	1
STP M-67C	Monthly Hose Reel Station Inspection	24A
STP M-80B	Indoor Fire Hose Operability Test	20
STP M-80D	Fire Hose Hydrostatic Testing	1

Notifications

50681419	50817032	50826569	50803608	50803505

Work Orders

64092332

Other Documents

Number	Title	Date
5391	Transient Combustible Permit – Fire Zone 14-A-2, Turbine Building, 104 ft.	September 22, 2015
5505	Transient Combustible Permit – Fire Zone 14-A-2, Turbine Building, 104 ft.	September 9, 2015
SDP 15-03	Risk Evaluation for Both Vital Switchgear Ventilation Trains Unavailable	0

Drawings

Number	Title	Revision
111906 sheet 7	Unit 1, Turbine Building Elevation 104 foot	4
111906 sheet 26	Unit 1, Containment Building Elevation 91 foot	2
111906 sheet 27	Unit 1, Containment Building Elevation 117 foot	3
111906 sheet 28	Unit 1, Containment Building Elevation 140 foot	2
TB-16	Unit 2, Fire Strategy: Turbine Building Buttress Elev. 85 & 104 foot	4
TB-14	Unit 2, Fire Drawing: Turbine Building Elev. 85 foot	7
106718	Unit 1 and 2 Operating Valve Identification Diagram (OVID), Fire Protection, Turbine Building Units 1 & 2, Sheet 7	180
TB-10	Turbine Building Elev. 140’ Unit 1	5
TB-11	Turbine Building Elev. 140’ Unit 1	2
TB-21	Turbine Building Elev. 140’ Unit 2	3
TB-22	Turbine Building Elev. 140’ Unit 2	2

Section 1R06: Flood Protection Measures

Procedure

Number	Title	Revision
MA1.ID14	Plant Crane Operation Restrictions	5

Notification

50594166

Work Orders

60077639 64094051

Drawings

Number	Title	Revision
438116	Unit 1 Plant Drawing Pull Boxes	23A

Section 1R07: Heat Sink Performance

Procedures

Number	Title	Revision
OP B-7	Spent Fuel Pool (SFP) System	16
MA1.ID22	Heat Exchanger Program	2

Notifications

50583220 50275659 50802150

Section 1R08: Inservice Inspection Activities

Procedures

Number	Title	Revision
WPS 51	Welding of P8 Materials with GTAW and/or SMAW ASME III	8
WPS 11	Welding of PS Materials with GTAW and/or SMAW ASME I, ASME III [b], ASME VIII, ANSI 831.1, and AWS 5.2.	8
WPS 5	Welding of P1 Materials with GTAW and/or SMAW ASME I, ASME III, and ASME VIII, ANSI 831.1, and AWS 5.2	8
GWS-ASME	ASME General Welding Standard	14
ISI X CRDM	Reactor Vessel Top and Bottom Head Visual Inspections	6
NDE PDI-UT-2	Ultrasonic Examination of Austenitic Piping	10
NDE VT-2-1	Visual Examination During Section XI System Pressure Test	2
NDE VT-1-1	Visual Examination of Component Surfaces	1
NDE PT 1	Visible Dye Liquid Penetrant Examination Procedure	5
ER1.ID2	Boric Acid Corrosion Control Program	7

AD4.ID2	Plant Leakage Evaluation	11
NDE VT 2 1	Visual Examination During Section XI System Pressure Test	2
STP R 8C	Containment Walk down for Evidence of Boric Acid Leakage	10
OM7.ID3	Root Cause Evaluations	40
WDI-SSP-1035	Manual Ultrasonic Examination of the Reactor Vessel Threads in Flange	2
03-9222009	Fuel Assembly Spring Breaker Tool Operating Instruction	1
ER1.DC1	Component Classification	2
ECG 7.6	Structural Integrity	2

Miscellaneous

Title	Revision/Date
1R19 Steam Generator Degradation Assessment	0
23-477R Pipe Support Modification Request For RHR-8708	0
LER 2013-005 Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld	August 22, 2013
DCL-15-062 Expected Submittal Date for Licensee Event Report 2015-001-01	May 7, 2015
DCL-15-033 Licensee Event Report 2015-001-00. Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld	March 2, 2015

Notifications

50809126	50809162	50680750	50571052	50611088
50611453	50612250	50614618	50619608	50680117
50570623	50625470	50690586	50634677	50700977
50809126	50749524	50614618		

Work Orders

60076166	60075395	60069613

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Procedures

Number	Title	Revision
AD8.DC50	Outage Safety Management	4
EOP E-0.1	Reactor Trip Response	41
CP M-4	Earthquake	33
EOP E-0	Reactor Trip or Safety Injection	44
OP1.DC10	Conduct of Operations	44A

Notifications

50708380	50703388

Work Order

64123783

Lesson

Number	Title
R154S1	Feedwater Failures

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

Number	Title	Revision
AD8.ID1	Outage Planning and Management	24A
AD8.DC50	Outage Safety Management	4
AD8.DC55	Outage Safety Scheduling	38
OP A-2:IX	Reactor Vessel-Vacuum Refill of the RCS	22

Other Document

Number	Title	Date
	1R19 Outage Safety Plan	August 4, 2015

Section 1R15: Operability Determinations and Functionality Assessments

Procedures

Number	Title	Revision
AD7.ID2	Standard Plant Priority Assignment	21
AD8.DC58	Outage Scope Control	10
OP1.DC40	Operations Equipment Deficiency Tracking	8
OP1.ID2	Time Critical Operator Action	8A

Notifications

50673779	50679028	50819536	50825752	50818175
50818963	50818962			

Section 1R19: Post-Maintenance Testing

Procedure

Number	Title	Revision
MP E-53.10V1	Motor Operated Valve Diagnostic Testing	14

Notification

50813672

Work Orders

64114901	64110775	64114900	64063996

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision
STP M-15	Integrated Test of Safeguard and Diesel Generator	65
OP L-6	Cold Shutdown/Refueling	72
STP V-676B	Penetration 76B Containment Isolation Valve Leak Testing	12
STP V-600	General Containment Isolation Valve Leak Tests	26
AD13.DC5	Containment Leakage Rate Testing Program	8A
STP V-645	Penetration 45 Containment Isolation Valve Leak Testing	30
STP R-1B	Rod Drop Measurement	37

STP R-1C	Digital Rod Position Indicator Functional Test	21

Work Orders

64148692	64063996	60037990

Drawing

Number	Title	Revision
102008-sheet 3C	Chemical & Volume Control System	115

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures

Number	Title	Revision
HTP-ZZ-02004	Control of Radioactive Sources	39
RCP D-200	Writing RWPs and ALARA Processes	53
RCP D-202	RWP Work Instructions	12
RCP D-220	Control of Access to High, Locked High, and Very High Radiation Areas	47
RCP D-240	Radiological Posting	23
RCP D-500	Routine and Job Coverage Surveys	40
RCP D-620	Radioactive Source Control Program	11
RP-DTI-SLT	Source Leak Testing	5
RP1	Radiation Protection	7
RP1.DC4	Radiological Hot Spot Identification and Control Program	6
RP1.ID16	Radiation Worker Expectations	6

Notifications

50565830	50660746	50661934	50662233	50662462
50662471	50663595	50663600	50663772	50663821
50664516	50664518	50665579	50666918	50667364
50667375	50668180	50668197	50668487	50668491
50669218	50669924	50669939	50671321	50671323
50671331	50672376	50672539	50673590	50673595
50676028	50677880	50679189	50682180	50683598

50686252	50686915	50690001	50690385	50691877
50696699	50696882	50697433	50698073	50698635
50699868	50702125	50705170	50705207	50706598
50706888	50708491	50710186	50804465	50807763

Audit and Self-Assessment

Number	Title	Date
50805466	Self Assessment – Quick Hit: Radiological Hazard Assessment and Exposure Control	September 17, 2015

Radiological Work Permits

Number	Title	Revision
15-1004-00	1R19 Radiation Protection in Containment	00
15-1014A-00	1R19 Upper Reactor Cavity Decontamination	00
15-1015-00	1R19 Minor Work in Posted HRA/LHRA/VHRA in Ctmt	00
15-1023-00	1R19 Fuel Movement and Under Water Work in Ctmt	00
15-1041-00	1R19 Primary Steam Generator Manway Work	00
15-1042-00	1R19 Primary Steam Generator Nozzle Dam Work	00
15-1061-00	1R19 Ctmt Valves and Breaches	00

Radiological Surveys

Number	Title	Date
37189	Follow Up Survey for PED Dose Rate Alarm	October 10, 2014
40671	115 East Yard Monthly	April 8, 2015
41897	CTF RCA Set Up With Hi-TRAC Mounted to Mating Device	June 19, 2015
42901	Filter Alley – South 100’ EL	August 20, 2015
43956	RCP 1-2 Seal #1 Removal (Air Sample)	October 8, 2015
44138	1R19 SG 1-1 Manway and Insert Removal (Air Sample)	October 13, 2015
44168	U1 Ctmt 91’	October 12, 2015
44192	Containment 140’ EL	October 12, 2015

Radiological Surveys

Number	Title	Date
44204	Filter Alley – North 100' EL	October 12, 2015
44337	U-1 100' Pen Scaffold Survey	October 14, 2015

Miscellaneous Documents

Number	Title	Date
	NSTS Annual Inventory Reconciliation Report	January 8, 2015
	Source Leak Test Report	May 5, 2015
	Source Inventory	May 18, 2015
	1R19 Plan of the Day – Day 12	October 15, 2015

Section 2RS3: In-plant Airborne Radioactivity Control and Mitigation

Procedures

Number	Title	Revision
OM6.ID10	Respiratory Protection Program	8
PEP M-96	TSC Ventilation Systems – DOP and Halide Penetration Tests	8
RCP D-707A	MSA FireHawk® (NIOSH) Self-Contained Breathing Apparatus Inspection	1A
RCP D-761	Instructions for Use of In-Line Auto-Air Breather Cart	1
RCP D-772	UNICUSIII Cylinder Recharging Station Operation	2
RCP D-772A	UNICUS TCOM-25 Trailer Cylinder Recharging Station Operation	0
STP M-53	Control Room Ventilation System – DOP and Halide Penetration Tests	21

Notifications

50537574	50538420	50540344	50549008	50567029
50573628	50579199	50579416	50585134	50596642
50602603	50606816	50626419	50636906	50637686
50637689	50664463	50674410	50685845	50700018

50711215	50806583

Audit and Self-Assessment

Number	Title	Date
50505531	Quick Hit Self-Assessment for 71124.03	September 21, 2015

Respirator Testing, Inspection, and Inventory Records

Number	Title	Date
	Monthly E-Plan Minimum Quantity SCBA Inventory	September 29, 2015
	Weekly Respirator Locker Inventory	September 28, 2015
EP-OAB274227	Posi3 USB Test Results (Complete SCBA Test)	April 24, 2015
EP-OAB274270	Posi3 USB Test Results (Complete SCBA Test)	April 10, 2015
EP-OAB274280	Posi3 USB Test Results (Complete SCBA Test)	March 5, 2015

HEPA and Charcoal Filter Testing Records

W/O Number	Title	Date
64038031	Control Room Ventilation System – DOP and Halide Penetration Tests (Unit 1)	April 24, 2013
64046932	TSC Ventilation Systems – DOP and Halide Penetration Testing	September 23, 2013
64074322	Control Room Ventilation System – DOP and Halide Penetration Tests (Unit 1)	April 14, 2015
64081429	TSC Ventilation Systems – DOP and Halide Penetration Testing	September 23, 2015

Compressed Air System Testing Records

Report Number	Title	Date
15-23394	TCOM Trailer Yard #2	August 25,2015
15-23395	TCOM 1	August 25,2015
15-23602	ACPP1 BAC 0-1	August 24,2015
15-23603	Fire Engine Bay	August 24,2015
15-23604	ACPP2 BAC 0-2	August 24,2015

Miscellaneous Documents

Number	Title	Date
	SCBA Qualified Personnel	September 23, 2015
5-14-13-01	SCBA Hydro Testing Report	May 14, 2013
RWP 15-1022	Respirator Use Evaluation	September 11, 2015
RWP 15-1042	Respirator Use Evaluation	July 20, 2015
RWP 15-1044	Respirator Use Evaluation	September 10, 2015

Section 4OA1: Performance Indicator Verification

Procedures

Number	Title	Revision
CY2.ID1	Radioactive Effluent Controls Program	13
AWP O-001	NRC Performance Indicators: RCS Specific Activity	12

Notification

50710240

Other

Title	Date
RCS Leakage Cycle 19 Data Unit 1 and 2	October 28, 2014 through December 2, 2015
RCS Dose Equivalent Iodine	Quarter Data Diablo Canyon 2015 and 2014

Section 4OA2: Problem Identification and Resolution

Procedures

Number	Title	Revision
OM4.ID3	Operating Experience Program	26A
PEP R-8F	Fuel Assembly Examinations in the Spent Fuel Pool	11
OP1.ID3	Reactivity Management Program	13
OM7.ID3	Root Cause Evaluation	42

Notifications

50812917	50818639	50817738	50815826	50614618
50357183				

Other Documents

Number	Title	Date
	Fuel Inspection – Face Outline Sketch (PEP R-8F, Attachment 9.2) – Fuel Assembly AB32	October 10, 2015
	Fuel Inspection – Face Outline Sketch (PEP R-8F, Attachment 9.2) – Fuel Assembly AB15	October 11, 2015
	Review Report: Failed Fuel in (North Anna Unit 2) Cycle 23 Core with Release of Pellets from Fuel Rods	November 27, 2014
DCL-00-068	10 CFR 50.59 Report for Facility Changes, Tests, and Experiments for the Report Period March 29, 1998 through December 31, 1999	April 26, 2000
DCL-06-050	10 CFR 50.59 Report for Facility Changes, Tests, and Experiments for the Report Period January 1, 2004, through December 31, 2005	April 21, 2006
DCL-99-139	Licensee Event Report 2-1999-002-00, Damaged Reactor Fuel Assemblies Due to Baffle Jetting	November 24, 1999

Drawing

Number	Title	Revision
663226	Cavity Seal Assembly	7

PAPERWORK REDUCTION ACT STATEMENT

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, Control Number-3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid Office of Management and Budget control number.

In accordance with 10 CFR 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).

Information Request
July 1, 2015
Notification of Inspection and Request for Information
Diablo Canyon Nuclear Power Plant
NRC Inspection Report 05000323/2014005

On October 5, 2015, reactor inspectors from the Nuclear Regulatory Commission's (NRC) Region IV office will perform the baseline inservice inspection at Diablo Canyon, Unit 1, using NRC Inspection Procedure 71111.08, "Inservice Inspection Activities." Experience has shown that this inspection is a resource intensive inspection both for the NRC inspectors and your staff. In order to minimize the impact to your onsite resources and to ensure a productive inspection, we have enclosed a request for documents needed for this inspection. These documents have been divided into two groups. The first group (Section A of the enclosure) identified information to be provided prior to the inspection to ensure that the inspectors are adequately prepared.

The second group (Section B of the enclosure) identifies the information the inspectors will need upon arrival at the site. It is important that all of these documents are up to date and complete in order to minimize the number of additional documents requested during the preparation and/or the onsite portions of the inspection.

We have discussed the schedule for these inspection activities with your staff and understand that our regulatory contact for this inspection will be Mr. Andrew Heffner of your licensing organization. The tentative inspection schedule is as follows:

Preparation week: September 28, 2015

Onsite weeks: October 5 through October 16, 2015

Our inspection dates are subject to change based on your updated schedule of outage activities. If there are any questions about this inspection or the material requested, please contact the lead inspector Jim Drake at (817) 200-1558 (James.Drake@nrc.gov).

A.1 <u>ISI/Welding Programs and Schedule Information</u>

a) A detailed schedule (including preliminary dates) of:

   i. Nondestructive examinations planned for ASME Code Class Components performed as part of your ASME Section XI, risk informed (if applicable), and augmented inservice inspection programs during the upcoming outage.

   ii. Examinations planned for Alloy 82/182/600 components that are not included in the Section XI scope (If applicable)

   iii. Examinations planned as part of your boric acid corrosion control program (Mode 3 walkdowns, bolted connection walkdowns, etc.)

   iv. Welding activities that are scheduled to be completed during the upcoming outage (ASME Class 1, 2, or 3 structures, systems, or components)

b) A copy of ASME Section XI Code Relief Requests and associated NRC safety evaluations applicable to the examinations identified above.

   i. A list of ASME Code Cases currently being used to include the system and/or component the Code Case is being applied to.

c) A list of nondestructive examination reports which have identified recordable or rejectable indications on any ASME Code Class components since the beginning of the last refueling outage. This should include the previous Section XI pressure test(s) conducted during start up and any evaluations associated with the results of the pressure tests.

d) A list including a brief description (e.g., system, code class, weld category, nondestructive examination performed) associated with the repair/replacement activities of any ASME Code Class component since the beginning of the last outage and/or planned this refueling outage.

e) If reactor vessel weld examinations required by the ASME Code are scheduled to occur during the upcoming outage, provide a detailed description of the welds to be examined and the extent of the planned examination. Please also provide reference numbers for applicable procedures that will be used to conduct these examinations.

f) Copy of any 10 CFR Part 21 reports applicable to structures, systems, or components within the scope of Section XI of the ASME Code that have been identified since the beginning of the last refueling outage.

g) A list of any temporary noncode repairs in service (e.g., pinhole leaks).

h) Please provide copies of the most recent self-assessments for the inservice inspection, welding, and Alloy 600 programs

A.2 Reactor Pressure Vessel Head

a) Provide a detailed scope of the planned bare metal visual examinations (e.g., volume coverage, limitations, etc.) of the vessel upper head penetrations and/or any nonvisual nondestructive examination of the reactor vessel head including the examination procedures to be used.

   i. Provide the records recording the extent of inspection for each penetration nozzle including documents which resolved interference or masking issues that confirm that the extent of examination meets 10 CFR 50.55a(g)(6)(ii)(D).

   ii. Provide records that demonstrate that a volumetric or surface leakage path examination assessment was performed.

   Copy of current calculations for EDY, and RIY as defined in Code Case N-729-1 that establish the volumetric and visual inspection frequency for the reactor vessel head and J-groove welds.

A.3 Boric Acid Corrosion Control Program

a) Copy of the procedures that govern the scope, equipment and implementation of the inspections required to identify boric acid leakage and the procedures for boric acid leakage/corrosion evaluation.

b) Please provide a list of leaks (including code class of the components) that have been identified since the last refueling outage and associated corrective action documentation. If during the last cycle, the unit was shut down, please provide documentation of containment walk down inspections performed as part of the boric acid corrosion control program.

A.4 Steam Generator Tube Inspections

a) A detailed schedule of:

   i. Steam generator tube inspection, data analyses, and repair activities for the upcoming outage (if occurring).

   ii. Steam generator secondary side inspection activities for the upcoming outage (if occurring).

b) Copy of SG history documentation given to vendors performing eddy current (ET) testing of the SGs during the upcoming outage.

c) Copy of procedure containing screening criteria used for selecting tubes for in-situ pressure testing and the procedure to be used for in-situ pressure testing.

d) Copy of previous outage SG tube operational assessment. Also include a copy of the following documents as they become available:

   i. Degradation assessment

   ii. Condition monitoring assessment

e) Copy of the document defining the planned SG ET scope (e.g., 100 percent of unrepaired tubes with bobbin probe and 20 percent sample of hot leg expansion transition regions with rotating probe) and identify the scope explanation criteria, which will be applied. Also identify and describe any deviations in this scope or expansion criteria from the EPRI Guidelines.

f) Copy of the document describing the ET acquisition equipment to be applied including ET probe types. Also identify the extent of planned tube examination coverage with each probe type (e.g. rotating probe -0.080 inches, 0.115 inches pancake coils and mid-range +point coil applied at the top-of-tube-sheet plus 3 inches to minus 12 inches).

g) Identify and quantify any SG tube leakage experienced during the previous operating cycle. Also provide documentation identifying which SG was leaking and corrective actions completed and planned for this condition.

h) Copy of steam generator eddy current data analyst guidelines and site validated eddy current technique specification sheets. Additionally, please provide a copy of EPRI Appendix H, "Examination Technique Specification Sheets," qualification records.

i) Provide past history of the condition and issues pertaining to the secondary side of the steam generators (including items such as loose parts, fouling, top of tube sheet condition, crud removal amounts, etc.).

Indicate where the primary, secondary, and resolution analyses are scheduled to take place.

A.5 Additional Information Related to all Inservice Inspection Activities

a) A list with a brief description of inservice inspection, and boric acid corrosion control program related issues (e.g., Condition Reports) entered into your corrective action program since the beginning of the last refueling outage. For example, a list based upon data base searches using key words related to piping such as: inservice inspection, ASME Code, Section XI, NDE, cracks, wear, thinning, leakage, rust, corrosion, boric acid, or errors in piping examinations.

b) Provide training (e.g. Scaffolding, Fall Protection, FME, Confined Space) if they are required for the activities described in A.1 through A.4.

Please provide names and phone numbers for the following program:

- leads: Inservice inspection (examination, planning)
- Containment exams
- Reactor pressure vessel head exams
- Snubbers and supports
- Repair and replacement program
- Licensing
- Site welding engineer
- Boric acid corrosion control program
- Steam generator inspection activities (site lead and vendor contact)

B. <u>Information to be Provided Onsite to the Inspector(s) at the Entrance Meeting (February 10, 2014)</u>:

B.1 <u>Inservice Inspection/Welding Programs and Schedule Information</u>

a) Updated schedules for inservice inspection/nondestructive examination activities, including planned welding activities, and schedule showing contingency repair plans, if available.

b) For ASME Code Class welds selected by the inspector from the lists provided from section A of this enclosure, please provide copies of the following documentation for each subject weld:

i. Weld data sheet (traveler).

ii. Weld configuration and system location.

iii. Applicable Code Edition and Addenda for weldment.

iv. Applicable Code Edition and Addenda for welding procedures.

v. Applicable welding procedures used to fabricate the welds.

vi. vi. Copies of procedure qualification records (PQRs) supporting the weld procedures from B.1.b.v.

vii. Copies of welder's performance qualification records (WPQ).

viii. Copies of the nonconformance reports for the selected welds (If applicable).

ix. Radiographs of the selected welds and access to equipment to allow viewing radiographs (if radiographic testing was performed).

x. Copies of the preservice examination records for the selected welds.

xi. Readily accessible copies of nondestructive examination personnel qualifications records for reviewing.

c) For the inservice inspection related corrective action issues selected by the inspectors from section A of this enclosure, provide a copy of the corrective actions and supporting documentation.

d) For the nondestructive examination reports with relevant conditions on ASME Code Class components selected by the inspectors from Section A above, provide a copy of the examination records, examiner qualification records, and associated corrective action documents.

e) A copy of (or ready access to) most current revision of the inservice inspection program manual and plan for the current interval.

f) For the nondestructive examinations selected by the inspectors from section A of this enclosure, provide a copy of the nondestructive examination procedures used to perform the examinations (including calibration and flaw characterization/sizing procedures). For ultrasonic examination procedures qualified in accordance with ASME Code, Section XI, Appendix VIII, provide documentation supporting the procedure qualification (e.g. the EPRI performance demonstration qualification summary sheets). Also, include qualification documentation of the specific equipment to be used (e.g., ultrasonic unit, cables, and transducers including serial numbers) and nondestructive examination personnel qualification records.

B.2 Reactor Pressure Vessel Head (RPVH)

a) Provide drawings showing the following (if performing any RPVH inspection activities):

   i. RPVH and control rod drive mechanism nozzle configurations.

   ii. RPVH insulation configuration.

   Note: The drawings listed above should include fabrication drawings for the nozzle attachment welds as applicable.

b) Copy of the documents which demonstrate that the procedures to be used for volumetric examination of the reactor vessel head penetration J-groove welds were qualified by a blind demonstration test in accordance with 10 CFR 50.55a(g)(6)(ii)(D).

c) Copy of volumetric, surface and visual examination records for the prior inspection of the reactor vessel head and head penetration J-groove welds.

B.3 Boric Acid Corrosion Control Program

a) Please provide boric acid walk down inspection results, an updated list of boric acid leaks identified so far this outage, associated corrective action documentation, and overall status of planned boric acid inspections.

b) Please provide any engineering evaluations completed for boric acid leaks identified since the end of the last refueling outage. Please include a status of corrective actions to repair and/or clean these boric acid leaks. Please identify specifically which known leaks, if any, have remained in service or will remain in service as active leaks.

B.4 Steam Generator Tube Inspections

a) Copies of the Examination Technique Specification Sheets and associated justification for any revisions.

b) Please provide a copy of the eddy current testing procedures used to perform the steam generator tube inspections (specifically calibration and flaw characterization/sizing procedures, etc.).

c) Copy of the guidance to be followed if a loose part or foreign material is identified in the steam generators.

d) Identify the types of SG tube repair processes which will be implemented for defective SG tubes (including any NRC reviews/evaluations/approvals of this repair process). Provide the flaw depth sizing criteria to be applied for ET indications identified in the SG tubes.

e) Copy of documents describing actions to be taken if a new SG tube degradation mechanism is identified.

f) Provide procedures with guidance/instructions for identifying (e.g. physically locating the tubes that require plugging) and plugging SG tubes.

g) List of corrective action documents generated by the vendor and/or site with respect to steam generator inspection activities.

B.5 Codes and Standards

a) Ready access to (i.e., copies provided to the inspector(s) for use during the inspection at the onsite inspection location, or room number and location where available):

    i. Applicable Editions of the ASME Code (Sections V, IX, and XI) for the inservice inspection program and the repair/replacement program.

b) Copy of the performance demonstration initiative (PDI) generic procedures with the latest applicable revisions that support site qualified ultrasonic examinations of piping welds and components (e.g., PDI-UT-1, PDI-UT-2, PDI-UT-3, PDI-UT-10, etc.).

c) Boric Acid Corrosion Guidebook Revision 1 – EPRI Technical Report 100097

The following items are requested for the
Occupational Radiation Safety Inspection
at Diablo Canyon
(October 12 thru 15, 2015)
Integrated Report 2015004

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **October 2, 2015**.

Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for Inspection Procedure 71124.01 should be in a file/folder titled "1- A," applicable organization charts in file/folder "1- B," etc.

If information is placed on *ims.certrec.com*, please ensure the inspection exit date entered is at least 30 days later than the onsite inspection dates, so the inspectors will have access to the information while writing the report.

In addition to the corrective action document lists provided for each inspection procedure listed below, please provide updated lists of corrective action documents at the entrance meeting. The dates for these lists should range from the end dates of the original lists to the day of the entrance meeting.

If more than one inspection procedure is to be conducted and the information requests appear to be redundant, there is no need to provide duplicate copies. Enter a note explaining in which file the information can be found.

If you have any questions or comments, please contact the lead inspector, Natasha Greene at (817)200-1154 or Natasha.Greene@nrc.gov.

Currently, the other inspector will be John O'Donnell. He may be contacted at (817)200-1441 or John.Odonnell@nrc.gov.

PAPERWORK REDUCTION ACT STATEMENT

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, control number 3150-0011.

1. **Radiological Hazard Assessment and Exposure Controls (71124.01)**
Date of Last Inspection: October 6, 2014

A. List of contacts (with official title) and telephone numbers for the Radiation Protection Organization Staff and Technicians

B. Applicable organization charts

C. Audits, self-assessments, and LERs written since date of last inspection, related to this inspection area

D. Procedure indexes for the radiation protection procedures

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.
   1. Radiation Protection Program Description
   2. Radiation Protection Conduct of Operations
   3. Personnel Dosimetry Program
   4. Posting of Radiological Areas
   5. High Radiation Area Controls
   6. RCA Access Controls and Radworker Instructions
   7. Conduct of Radiological Surveys
   8. Radioactive Source Inventory and Control
   9. Declared Pregnant Worker Program

F. List of corrective action documents (including corporate and subtiered systems) since date of last inspection
   a. Initiated by the radiation protection organization
   b. Assigned to the radiation protection organization
   c. Identify any CRs that are potentially related to a performance indicator event

   NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide documents which are "searchable" so that the inspector can perform word searches.

   If not covered above, a summary of corrective action documents since date of last inspection involving unmonitored releases, unplanned releases, or releases in which any dose limit or administrative dose limit was exceeded (for Public Radiation Safety Performance Indicator verification in accordance with IP 71151)

G. List of radiologically significant work activities scheduled to be conducted during the inspection period (If the inspection is scheduled during an outage, please also include a list of work activities greater than 1 rem, scheduled during the outage with the dose estimate for the work activity.)

H. List of active radiation work permits

I. Radioactive source inventory list

3. In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

Date of Last Inspection: February 11, 2013

A. List of contacts and telephone numbers for the following areas:
   1. Respiratory Protection Program
   2. Self-contained breathing apparatus

B. Applicable organization charts

C. Copies of audits, self-assessments, vendor or NUPIC audits for contractor support (SCBA), and LERs, written since date of last inspection related to:
   1. Installed air filtration systems
   2. Self-contained breathing apparatuses

D. Procedure index for:
   1. use and operation of continuous air monitors
   2. use and operation of temporary air filtration units
   3. Respiratory protection

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.
   1. Respiratory protection program
   2. Use of self-contained breathing apparatuses
   3. Air quality testing for SCBAs

F. A summary list of corrective action documents (including corporate and subtiered systems) written since date of last inspection, related to the Airborne Monitoring program including:
   1. continuous air monitors
   2. Self-contained breathing apparatuses
   3. respiratory protection program

   NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide documents which are "searchable."

G. List of SCBA qualified personnel - reactor operators and emergency response personnel

H. Inspection records for self-contained breathing apparatuses (SCBAs) staged in the plant for use since date of last inspection.

I. SCBA training and qualification records for control room operators, shift supervisors, STAs, and OSC personnel for the last year.

   A selection of personnel may be asked to demonstrate proficiency in donning, doffing, and performance of functionality check for respiratory devices.