UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

April 27, 2010

Mr. Barry Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2, Mail Stop A-DB-3080
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION INTEGRATED INSPECTION REPORT 05000346/2010-002

Dear Mr. Allen:

On March 31, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Davis-Besse Nuclear Power Station. The enclosed inspection report documents the inspection findings which were discussed on April 6, 2010, with you and other members of your staff.

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

Based on the results of this inspection, two NRC-identified findings and one self-revealed finding of very low safety significance were identified. Two of the findings involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, 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 a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Davis-Besse Nuclear Power Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Davis-Besse Nuclear Power Station. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-346
License No. NPF-3

Enclosure: Inspection Report 05000346/2010-002
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2010-002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	January 1, 2010, through March 31, 2010
Inspectors:	J. Rutkowski, Senior Resident Inspector A. Wilson, Resident Inspector J. Cassidy, Senior Health Physicist T. Go, Health Physicist D. Jones, Reactor Inspector D. Melendez-Colon, Acting Project Engineer P. Voss, Reactor Engineer
Approved by:	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

Inspection Report 05000346/2010-002; 1/1/10-3/31/10; Davis-Besse Nuclear Power Station; Flooding, Annual Heat Sink Performance, and Refueling Outage Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified. Two of the findings were considered Non-Cited Violations of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance for the licensee's failure to maintain normally energized medium voltage cables BPGD302C, C1, D, and D1 in an environment consistent with the cable design. The cables, which are output cables for the station blackout diesel generator and were not designed for long-term water submergence, were in a manhole that was shown to be flooded regularly. Water submergence of energized medium voltage cables, not designed for water submergence, can accelerate deterioration of such cables and potentially affect the ability of the cables to withstand electrical transients. The licensee's procedures and program for medium voltage cables did recognize the issue but did not identify the submergence issue with these cables. In response to the finding the licensee increased the frequency of monitoring for water in the manhole. No violation of NRC requirements was identified.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the station blackout diesel generator was to provide electrical power to emergency core cooling systems (ECCSs) in the event of a loss of all alternating current power. The inspectors determined that the finding was of very low safety significance because it did not result in any inoperability of required equipment and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of human performance, work control component, because the licensee failed to appropriately plan work activities incorporating risk insights and job site conditions, including environmental conditions, which may impact plant system and components. Specifically, although the intent was to address water submergence of energized medium voltage risk-significant cables to reduce the risk of early cable failure, the licensee failed to identify and address site and component conditions that regularly submerged the energized 4160 volt cable associated with the electrical output of the station blackout diesel generator. (H.3(a)) (Section 1R06)

- Green. A self-revealed finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was identified for the failure to adequately implement post-maintenance testing (PMT) when restoring emergency core cooling system (ECCS) room cooler 4, in ECCS train 1 pump room, to service after performing preventive maintenance. The licensee did not discover the failure of the room cooler's service water inlet valve during PMT and inappropriately declared the room cooler operable after completion of testing. This condition existed until the following day, when sufficient flow was not obtained during performance testing of the cooler. As an immediate corrective action, an engineering technical evaluation determined that under current conditions, room cooler 5, the other cooler in the room, would provide sufficient heat transfer to maintain the room temperature within the bounds of design basis, thus assuring operability of ECCS train 1 equipment. Also, the work orders for the ECCS room coolers have been revised to have Operations document that the system is at normal operating pressure before performing a PMT leak check.

  The finding is more than minor because it is associated with the equipment reliability attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the inadequate PMT did not ensure operability of ECCS room cooler 4, which also affected the operability of ECCS train 1 equipment. The inspectors determined that the finding was of very low safety significance because the inspectors answered "no" to all five screening questions under the mitigating systems cornerstone column. This finding has a cross-cutting aspect in the area of human performance, work practices component, because the licensee did not use appropriate human error prevention techniques. Specifically, the licensee did not properly document that the system reached normal operating pressure or temperature when performing the PMT. (H.4(a)) (Section 1R07)

Cornerstone: Other

- Green. The inspectors identified an NCV of 10 CFR 72.212, "Conditions of general license issued under 72.210," having very low safety significance for non-compliance with transient combustible material control procedures required for the Davis-Besse spent fuel dry horizontal storage modules (HSMs). A mobile crane and a utility truck were parked and unattended within an area designated by signs as a 75 foot exclusion area around the HSMs. The issues identified were not in compliance with the licensee's procedures, specifically DB-FP-7 for control of combustible transient material. Control of transient combustible material was required to ensure conformance with temperature limitations for the HSMs as outlined in the NRC-issued HSM Certificate of Compliance. Procedure DB-FP-7 specifically requires that vehicles within 75 feet of the HSMs shall have a vehicle attendant at all times. The licensee re-emphasized the procedural requirements with involved personnel.

  This finding was greater than minor because it was associated with the protection against potential fire damage to the HSMs, and, if left uncorrected, would become a more significant safety concern since repeated presence of unattended combustible material in the vicinity of the HSMs increased the vulnerability of the HSMs to damage from a fire. Additionally, contractor personnel not adhering to station procedures, if left uncorrected, could become a more significant issue. The inspectors determined that the finding was not suitable for SDP evaluation because the noncompliance did not involve permanently installed plant equipment. The finding was reviewed by regional management, in accordance with IMC 0609, Appendix M and determined to be of very low safety significance. The unattended time was short and the equipment was placed in a location easily visible to plant locations that are always manned. The finding is related to the cross-cutting area of Human Performance because licensee personnel did not ensure sufficient oversight of contractor work activities to ensure compliance with site procedures associated with protection of the dry spent fuel storage modules. (H.4.(c)) (1R20)

B. <u>Licensee-Identified Violations</u>

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period at full rated thermal power. During the week of January 10, 2010, the licensee started end-of-cycle activities to maintain electrical output to the beginning of refueling outage 16 on February 28, 2010. The sequential actions included withdrawing axial power shaping rods, gradually reducing average reactor coolant temperature, and gradually reducing reactor power. The actions resulted in reactor power being reduced to approximately 90 percent with a reactor coolant system (RCS) average temperature of approximately 572.5 degrees immediately prior to the shutdown for the refueling outage. The unit remained in a refueling outage condition at the end of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition – Heavy Snowfall Conditions

a. Inspection Scope

On February 9, 2010, a winter weather advisory was issued for expected snow fall and high winds through February 10, 2010. The inspectors observed the licensee's preparations and planning for the significant winter weather potential. The inspectors reviewed licensee procedures and discussed potential compensatory measures with control room personnel. The inspectors focused on plant management's actions for implementing the station's procedures for ensuring adequate personnel for safe plant operation and emergency response would be available. The inspectors conducted a site walkdown including walkdowns of various plant structures and systems to check for maintenance or other apparent deficiencies that could affect system operations during the predicted significant weather. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment - Quarterly Partial System Walkdowns (71111.04)

a. Inspection Scope

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

- decay heat and low pressure injection train 2 during scheduled inoperability of decay heat and low pressure injection train 1 for preventive maintenance on support systems on January 13, 2010;
- high pressure injection train 2 during scheduled inoperability of high pressure injection train 1 for preventive maintenance on January 19, 2010; and
- high pressure injection train 1 during scheduled inoperability of emergency core cooling system (ECCS) train 2 for scheduled maintenance activities on train 2 support cooling systems on February 2, 2010.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Technical Specification (TS) requirements, condition reports (CRs), and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. Documents reviewed are listed in the Attachment.

These activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- auxiliary building elevation 603 foot corridor and passageway (Rooms 400 and 404, Fire Area V);
- mechanical penetration room 1 (Room 208, Fire Area AB);
- radwaste exhaust equipment and main station exhaust fan room (Rooms 500, 501, and 515, Fire Area EE);

- auxiliary building 545 foot elevation hallway and adjoining rooms (Rooms 104, 106, 106A, 108, 109, 109A, 110, 111, 116, 120, and 121; Fire Area A); and
- containment annulus east and west (Rooms 127E and 127W; Fire Area A and AB).

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events (IPEEE) with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Annual Fire Protection Drill Observation</u> (71111.05A)

a. <u>Inspection Scope</u>

On February 24, 2010, the inspectors observed a fire brigade activation for indication of smoke in the wet wash facility within the protected area. Based on this observation, the inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) smoke removal operations; and (7) utilization of pre-planned strategies. Documents reviewed are listed in the Attachment to this report.

These activities constituted one annual fire protection inspection sample as defined in IP 71111.05-05.

b. <u>Findings</u>

No findings of significance were identified.

1R06 Flooding (71111.06)

.1 (Closed) Unresolved Item 05000346/2009005-01, “Ability of Medium Voltage Cable from Blackout Diesel to Function Long Term in Water Submerged State”

During observation of the condition of cables in manhole 3045 during the fourth quarter of 2009, the inspectors noted that the manhole was flooded and that the conditions inside the manhole gave indications that flooding was not an uncommon occurrence. Discussions with licensee personnel provided additional indication that the manhole was susceptible to flooding sufficient to submerge electrical cables in the manhole. The cable of interest to the inspectors was the normally energized 4160 volt cable that would carry the output of the station’s blackout diesel generator to the station’s electrical buses. The licensee entered the issue in their CAP as CR 09-67489. The inspectors asked for additional information on the design of the cable and licensee commitments and received that information but did not have the opportunity to review all of the material during the fourth quarter of 2009. The inspectors completed review of the requested material and reviewed the licensee’s investigation for the CR during this inspection interval. URI 05000346/2009005-01, “Ability of Medium Voltage Cable from Blackout Diesel to Function Long Term in Water Submerged State,” is closed.

.2 Station Blackout Diesel Generator Output Cables Not Maintained In an Environment Consistent with Design

Introduction: A finding of very low safety significance (Green) was identified by the inspectors for the licensee’s failure to maintain normally energized medium voltage cables BPGD302C, C1, D, and D1 in an environment consistent with the cable design. Specifically, the cables, which were not designed for long-term water submergence, were in a manhole that was shown to be flooded regularly. Industry experience has shown that normally energized medium voltage cables, not designed for water submergence, can experience accelerated deterioration in a water submerged state. The licensee’s procedures and program for medium voltage cables did recognize the issue but did not identify the submergence issue with these cables. No violation of NRC requirements was identified.

Description: On November 5, 2009, the inspectors observed that electrical manhole 3045 was opened for scoping of future design changes and that the manhole was flooded with the electrical cables in the manhole submerged. The inspectors subsequently determined that this manhole did not contain any safety-related medium voltage cables, but did contain medium voltage cables that were normally energized for delivering the 4160 volt output of the station blackout diesel generator to station bus D2. In discussions with plant personnel, the inspectors also learned that this manhole communicated via underground conduits with at least two other manholes and that finding the manholes flooded was a common occurrence. Manhole 3045 and the other manholes that connected to this manhole were not provided with sump pump capability. In addition to the November 5, 2009, finding, manhole 3045 was found filled with water on June 4, 2009, and January 27, 2010. The original inspection interval for this manhole was specified as every 3 years. The June 2009 inspection was the original inspection for the manhole although the potential need for inspection was identified in 2007 in CR 06-11583. After seeing the results from the November 2009 inspection, a new initial nominal inspection interval of once every 84 days was established in December 2009.

The station blackout diesel generator and the associated output cables were designated as “Augmented Quality” components. “Augmented Quality” components were to have applied all nuclear quality assurance program requirements except as specifically exempted. Under the NRC’s Maintenance Rule, the station blackout diesel generator and associated support systems were classified as risk significant and whose failure could prevent safety-related structures, systems, and components (SSC) from fulfilling their safety-related function. In NORM-ER-3112, “Cable Monitoring,” dated July 31, 2008, the licensee noted that plant sites are implementing engineering programs to ensure the proper function of the electrical manholes, including those with cabling addressed by the Maintenance Rule. The document continued that any leakage of groundwater or rainwater has been addressed to prevent the accumulation of water in the manhole unless the cables are suitable for submergence.

The inspectors noted that the cables for the output of the station blackout diesel generator were in the station’s medium voltage wetted cable replacement program; the existing cables were installed in 1991. Manufacturer certification records indicate that the cables were manufactured before 1982 by Okonite with Okoguard (ethylene-propylene rubber) insulation with an Okolon (vulcanized chlorosulfonated polyethylene) jacket. Licensee documents indicated that the cables were intended for potentially wet environments but were not designed as water submerged cables. Cables of this construction have shown susceptibility to accelerated deterioration when energized in a water submerged environment. The inspectors’ visual observation of the cables in the manhole did not provide any indication that would question cable present operability. No cable test results were available for the present condition of the cable.

The licensee entered the issue in their CAP as CR 09-67489. The licensee has initiated action to review the need for a permanent sump pump for manhole 3045. The licensee also planned on replacing the existing blackout diesel generator output cable during the next operating cycle.

<u>Analysis</u>: The inspectors determined that not identifying and addressing station blackout diesel generator output cable water submergence was contrary to licensee’s stated intent of addressing water submergence of medium voltage cables as specified in industry guides and company documents and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the mission of the station blackout diesel generator was to provide electrical power to emergency core cooling systems in the event of a loss of all alternating current power. Water submergence of energized medium voltage cables, not designed for water submergence, can accelerate deterioration of such cables and potentially affect the ability of the cable to withstand electrical transients that would occur with the loss of all alternating current electrical power and subsequent restoration of power with the station blackout diesel generator.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems cornerstone, since the station blackout diesel generator was designed to provide power for mitigating systems. The inspectors determined that the finding was of very low safety significance (Green) because it did not result in any inoperability of required equipment and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

This finding has a cross-cutting aspect in the area of human performance, work control component, because the licensee failed to appropriately plan work activities incorporating risk insights and job site conditions, including environmental conditions, which may impact plant system and components. Specifically, although the intent was to address water submergence of energized medium voltage risk-significant cables to reduce the risk of early cable failure, the licensee failed to identify and address site and component conditions that regularly submerged the energized 4160 volt cable associated with the electrical output of the station blackout diesel generator. (H.3(a))

Enforcement: Because this finding does not involve a violation of regulatory requirements and has a very low safety significance, it is identified as FIN 05000346/2010002-01, "Failure to Maintain Station Blackout Diesel Generator Output Cables In an Environment Consistent with Design."

1R07 Annual Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the licensee's testing of ECCSs room cooler 4 to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria and the correlation of scheduled testing and the frequency of testing. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this document.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

Introduction: A self-revealed finding of very low safety significance (Green) and associated non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XI, "Test Control," was identified for the failure to adequately implement post-maintenance testing (PMT) when restoring ECCS room cooler 4 to service after performing preventive maintenance.

Description: On January 13, 2010, ECCS room cooler 4 was removed from service to perform an inspection of the cooler in accordance with work order (WO) 200375179. As part of the work process ECCS train 1 was declared inoperable but available due to reduced train 1 pump room cooling capability. The order required the end covers of the cooler be removed to provide access to the service water tubes/channels internal to the cooler. The inspection revealed that the tubes were clear and that no cleaning was required. Only small amounts of loose debris were removed from the channel heads. After the channel heads were reinstalled, the order specified a leak check as a PMT. Upon completion of the leak test, the Shift Manager declared the ECCS room cooler 4 operable on January 14, 2010, and exited the applicable technical specifications action statements for ECCS train 1 equipment.

Performance monitoring testing of ECCS room cooler 4 took place the following day, January 15, 2010. During this test, the flow rate through the cooler, as calculated by the procedure, was determined to be negative 1.19 gallons per minute. Indications revealed that differential pressure across the cooler was negative 0.74 pounds per square inch gauge. Further investigation revealed flow blockage between SW 110 (ECCS room cooler 4 inlet valve) and the cooler. SW 110 appeared to be in the correct position (open), and therefore, it was believed that its wedge separated from the stem of the valve causing the blockage in the line to room cooler 4.

Based on the results of the performance monitoring test, ECCS room cooler 4 was declared inoperable, which affected the operability of train 1 ECCS equipment. An immediate investigation was initiated to evaluate ECCS room cooler 5 capability of providing sufficient heat transfer to ECCS room 1. Cooler 4 and cooler 5 provide for cooling of ECCS train 1 pump room. The engineering technical evaluation determined that room cooler 5 would provide sufficient heat transfer to maintain the room temperature within the bounds of the design basis under the conditions that service water forebay temperature remain below 45 degrees F and ECCS room cooler 5 service water flowpath remain unchanged. The licensee issued a standing order containing the bounding limitations, and ECCS train 1 equipment was declared operable at 21:34 on January 16, 2010.

The licensee had an opportunity to discover the failure of SW 110 during the PMT leak check performed on January 13. The PMT for WO 200375179 states, “Coordinate with Operations to fill, vent and gradually bring the system to normal operating pressure; this will allow for performance of an initial service pressure test.” Maintenance was assigned to verify with Operations that the system was at or near normal operating pressure prior to performing the leak check. This step in the WO was signed off as completed, yet there is no reference to what Operations based their information on to conclude that the system was at or near normal operating pressure or temperature. The performance monitoring test uses local inlet pressure indicators to verify normal operating pressures, but these were not used during the PMT leak check. The use of the pressure indicators for verification would have detected the failure of SW110 during the PMT. As a consequence, the room cooler was returned to service without demonstrating that the cooler would function correctly. The PMT did not ensure that service water was flowing through the room cooler. As a corrective action, WOs associated with cleaning/inspection of the ECCS room coolers were modified to have Operations record the local inlet pressure indicators to ensure the system is at normal operating pressure prior to performing the leak check PMT. Condition Report 10-70082 was initiated to track the repair of SW110. A temporary modification removed the internals to the valve with replacement of the entire valve to be completed later.

<u>Analysis</u>: The inspectors determined that the failure to adequately implement PMT when restoring ECCS room cooler 4 to service was a performance deficiency. This deficiency was reasonably within the licensee’s ability to foresee and correct and could have been prevented. The finding is more than minor because it is associated with the equipment

reliability attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the inadequate PMT did not ensure operability of ECCS room cooler 4, which also affected the operability of ECCS train 1 equipment. Therefore, the finding was evaluated using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems cornerstone since the ECCS room cooler is used to support the operability of systems that respond to initiating events. The inspectors answered “no” to all five screening questions under the mitigating systems cornerstone column and determined that the finding was of very low safety significance (Green).

This finding has a cross-cutting aspect in the area of human performance, work practices component, because the licensee did not use appropriate human error prevention techniques. Specifically, the licensee did not properly document that the system reached normal operating pressure or temperature when performing the PMT. (H.4(a))

Enforcement: Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XI, “Test Control,” states, in part, that a test program is established to assure that all testing required to demonstrate that SSC will perform satisfactorily in service is identified and performed in accordance with written test procedures. Contrary to these requirements, the licensee failed to ensure test activities involving ECCS room cooler 4 were appropriately performed to demonstrate the cooler would perform satisfactorily. Specifically, the inadequate PMT did not ensure operability of ECCS room cooler 4. This condition existed until the following day, when sufficient flow was not obtained during performance monitoring testing of the cooler. As an immediate corrective action, an engineering technical evaluation determined that under then current conditions, room cooler 5 would provide sufficient heat transfer to maintain the room temperature within the bounds of design basis, thus assuring operability of ECCS train 1 equipment. Also, the WOs for the ECCS room coolers have been revised to have Operations document the system is at normal operating pressure before performing the PMT leak check. Because this violation was of very low safety significance and since it was entered in the licensee’s CAP (CR 10-70078), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2010002-02, Inadequate Post-Maintenance Testing of ECCS Room Cooler)

1R08 Inservice Inspection Activities (71111.08P)

From March 1, 2010, through March 25, 2010, the inspectors conducted a review of the implementation of the licensee’s Inservice Inspection (ISI) Program for monitoring degradation of the RCS, steam generator tubes, emergency feedwater systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, 1R08.3, 1R08.4 and 1R08.5 below constituted one ISI sample as defined in Inspection Procedure (IP) 71111.08-05.

On March 17, 2010, the NRC issued a news release stating that a special inspection team had been dispatched to the site and would follow activities associated with the examination and subsequent repair of the reactor vessel head. Details associated with Section 1R08.02 inspection will be documented in a separate inspection report.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed the following non-destructive examinations mandated by the ASME Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements, and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Ultrasonic examination (UT) of the RCS 36 inch pipe to steam generator inlet nozzle weld MK36 to MK 70, report number 16-UT-012;
- Liquid penetrant examination (PT) of integrated control system (ICS) valve ICS11B Body Weld J, Report Number 16-PT-043; and
- Liquid penetrant examination (PT) of ICS valve ICS11B Body Weld K, Report Number 16-PT-046.

During non-destructive surface and volumetric examinations performed since the previous refueling outage, the licensee had not identified any recordable indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary welds completed for risk significant systems during the last refueling outage to determine if the licensee applied the pre-service non-destructive examinations and acceptance criteria required by ASME Code Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedures were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Repair/replacement welding of reactor coolant system (RCS), ASME Class 1, weld overlay of pressurizer decay heat line to hot leg nozzle (DH 33A-CCA-4-1-FW1), Work Order Number 200249953.

b. Findings

No findings of significance were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the vessel upper head, a special inspection was initiated to investigate control rod drive nozzle indications and boric acid leakage identified during the outage (RFO16). This portion of the baseline inspection procedure will be completed as part of the special inspection.

.3 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors performed an independent walkdown of the RCS and related lines in the containment including the under vessel penetrations, which had received a recent licensee boric acid walkdown and verified whether the licensee's BACC visual examinations emphasized locations where boric acid leaks can cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of RCS components with boric acid deposits to determine if degraded components were documented in the corrective action system. The inspectors also evaluated corrective actions for any degraded RCS components to determine if they met the ASME Section XI Code.

- CR 10-73145; 16RFO BACC-A Packing Leak Was Found On RC1AB; and
- CR 10-72582; BACC-A Packing Leak Was Found On MU409.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- CR 10-72427; 16RFO BACC-A Threaded Connection Leak Was Found On PTRC2A3;
- CR 10-73144; 16RFO BACC-A Packing Leak Was Found On RC1AA; and
- CR 10-72883; BACC-A Packing Leak Was Found On MU72.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition of eddy current (ET) data, interviewed ET data analysts, and reviewed documentation related to the SG ISI program to determine if:

- in-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) TR-107620, Steam Generator In-Situ Pressure Test Guidelines and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;
- the numbers and sizes of SG tube flaws/degradation identified was bound by the licensee's previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to meet the TSs, and the EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6;
- the SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;

- the licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- the licensee implemented an inappropriate "plug on detection" tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons per day or the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6; and
- the licensee performed secondary side SG inspections for location and removal of foreign materials.

The licensee did not perform in-situ pressure testing of SG tubes. Therefore, no NRC review was completed for this inspection attribute.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems entered into the licensee's corrective action program and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI/SG related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On January 22, 2010, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness - Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

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

- plant computer system; and
- reactor protection system.

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

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;

- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for SSCs/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- work activities during the week of January 18, 2010, which included a high pressure injection train 1 outage, a switchyard J bus outage, and previously unplanned activities to review as-found conditions on the high pressure injection pump motor electrical connections and a plan for isolating the number 2 letdown cooler as part of the effort to identify source of reactor coolant system (RCS) leakage into the component cooling water system;
- work activities during the week of January 25, 2010, which included a switchyard K bus outage, a late identified need to remove startup transformer 2 from service to ensure sufficient physical separation between scheduled switchyard work and energized conductors, and the start of planned RCS average temperature reduction at end of the operating cycle;
- work activities during the week of February 15, 2010, which included continuing pre-refueling outage reduction of RCS average temperature to 572.5 degrees, continuing issues with service water pump 2 discharge strainer, and an unanticipated need to recalculate and change gain adjustments associated with integrated control system boards that monitor steam generator heat transfer limits (BTU limits);
- work activities during the week of February 28, 2010, which included plant shutdown and cooldown to mode 5, initial inspections of the reactor containment including review of conditions under the reactor vessel;
- yellow shutdown risk activities on March 6 and 7, 2010, which included draining the RCS from fill and vented to a water level that was just below the reactor

vessel flange, and included addressing unanticipated high levels of hydrogen in the pressurizer gas space;

- yellow shutdown risk evolution of lowering reactor water level from just below the flange to 26 inches above the centerline of the reactor vessel hot leg nozzles on March 8, 2010, in preparation for steam generator nozzle dam installation; and
- lift and movement of the reactor vessel head from the reactor vessel to the containment storage stand on March 9, 2010.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift engineer, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed Technical Specifications (TS) requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These maintenance risk assessments and emergent work control activities constituted seven samples as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- CR 10-70072 which documented inadequate flow to ECCS room cooler 4;
- CR 10-70197 which documented and evaluated the acceptability of a less than desired bend radius of the electrical power supply cables in the motor junction connection box of high pressure injection pump 1 motor;
- CR 10-72359 which documented and evaluated the minimum water level required for operability of the steam generators in Mode 5; and
- CR 10-72688 which documented and evaluated the condition that steam-feedwater rupture control system channel 4 energized after a test specified loss of power in a block condition with the logic not resetting.

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

This operability inspection constituted four samples as defined in IP 71111.15-05.

b. Findings

Introduction: The inspectors reviewed the licensee's determination that the failure of the of steam-feedwater rupture control system (SFRCS) channel 4, which also was determined to affect channel 3, was not reportable under 10 CFR 50.72 because the issue was found during a mode in which the equipment was not required to be operable. The inspectors did not complete their review of the licensee's notification determination. Additionally, the inspectors did not complete their review of the consequences of the reported failure. The issues are considered a URI pending further review by the licensee's staff and the inspectors.

Description: On March 2, 2010, after performance of the integrated safety features actuation system test, the licensee identified that SFRCS channel 4 had energized in a blocked condition. With such a condition existing, the licensee stated the channel could fail to operate correctly after a loss of offsite power. Subsequently, the licensee also determined that SFRCS channel 3 could also experience this condition and potentially result in auxiliary feedwater being supplied to a ruptured steam generator. In CR 10-73067 the licensee determined that the condition resulted in an unanalyzed condition that significantly degraded plant safety but that no 10 CFR 50.72 report was required because at the time of discovery the plant was in Mode 5 and SFRCS was not required to be operable.

The licensee additionally documented the equipment conditions in CR 10-72446 and CR 10-72688. A root cause report was prepared as part of CR 10-73076. The root cause was issued for final site review on March 31, 2010. At the end of the inspection period the inspectors had not completed their review of the licensee's reportability determination and the root cause report. Pending further review of the licensee's evaluation and supporting documentation by the inspectors, the issue is considered an unresolved item (URI) 05000346/20100002-03, Inoperability of Steam-Feedwater Rupture Control System).

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modifications:

- ECP 09-0550-000, "End-of-Cycle Reactor Coolant System Average Temperature Reduction and Associated Alarm Setpoint Changes"; and
- ECP 10-0029-000, "Remove Valve Internals from SW110."

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected systems. The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted two temporary modification samples as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The following engineering design package was reviewed and selected aspects were discussed with engineering personnel:

- ECP 08-0571, "Reroute AFW Common Suction Piping to Resolve Non-Conformance."

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated or scheduled to be updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The modification provided for additional auxiliary feedwater pump safety-related suction piping to provide for additional water volume to mitigate the potential for pump cavitation in the event of a break in suction piping not missile protected. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted one permanent plant modification samples as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance (PM) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- refill and testing of high pressure injection train 1 after scheduled preventive maintenance on the train's pump, motor, and motor-operated valves and maintenance on the internals of a manual isolation valve in the system;
- leakage testing and performance monitoring testing of emergency core cooling system (ECCS) room coolers 1 and 2 after preventive maintenance to clean and inspect the train 2 ECCS room coolers;
- integrated steam-feedwater rupture control system (SFRCS) actuation channel 2 testing and PM testing following preventive maintenance work involving replacement of critical SFRCS relays. Testing revealed that relays associated with the turbine trip inputs to SFRCS had been replaced with new relays having AC coils rather than the required DC coils; and
- overspeed trip and high speed stop testing of the auxiliary feed pump turbine 1 governor after governor planned maintenance.

These activities were selected based upon the SSC's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with PM tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted four post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

.2 Post-Maintenance Testing Associated with Temporary Instruction 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

a. Inspection Scope

When reviewing PM testing of high pressure injection train 1 after working on the internals of a system valve, the inspectors verified that the procedures and operations directions were acceptable for refilling the portion of the drained system and ultra-sonic testing (UT) was appropriate for identifying any potential gas pockets within the refilled lines.

The inspectors reviewed procedures used for determination of void volumes to ensure that the void criteria was satisfied (Temporary Instruction (TI) 2515/177, Section 04.03.a). Also, the inspectors reviewed the procedure used for filling and venting the following conditions which introduced voids into the subject system to verify that the procedures acceptably addressed testing for such voids and provided acceptable processes for their reduction or elimination (TI 2515/177, Section 04.03.b). Specifically, the inspectors verified that:

- gas intrusion prevention, refill, venting, evaluation, and void correction activities were acceptably controlled by approved operating procedures or operations orders (TI 2515/177, Section 04.03.c.1);
- procedures ensured the system did not contain voids that may jeopardize operability (TI 2515/177, Section 04.03.c.2);
- the licensee entered any identified issues or detected gas accumulations into the CAP as needed to ensure acceptable response to issues (TI 2515/177, Section 04.03.c.5); and
- procedure or clearance restoration directions included independent verification that critical steps were completed (TI 2515/177, Section 04.03.c.6).

The inspectors verified the following with respect to void detection:

- venting procedures and practices utilized criteria such as adequate venting durations and observing a steady stream of water (TI 2515/177, Section 04.03.d.7);
- an effective sequencing of void removal steps was followed to ensure that gas does not move into previously filled system volumes (TI 2515/177, Section 04.03.d.8);
- qualitative void assessment methods included expectations that the void will be significantly less than allowed by acceptance criteria (TI 2515/177, Section 04.03.d.9); and
- surveillances were conducted at any location where a void may form, including high points, dead legs, and locations under closed valves in vertical pipes (TI 2515/177, Section 04.03.d.11).

The inspectors verified the following with respect to void control:

- void removal methods for the drained and refilled piping were acceptably addressed by approved procedures or operations orders (TI 2515/177, Section 04.03.f.1).

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

1R20 Outage Activities - Refueling Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the refueling outage (RFO), started on February 28, 2010, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TSs when taking equipment out of service;
- implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes, systems, and components;
- controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity;
- maintenance of containment as required by TS;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- reconstitution, using stainless steel pins, of fuel elements identified with leaking fuel pins and scheduled to be reinserted in the reactor core;
- identification and storage of fuel element with a fuel pin exhibiting a 360 degree circumferential clad crack;
- work schedules of several plant sections and the schedules' adherence to work hour limits; and
- licensee identification and resolution of problems related to RFO activities.

Documents reviewed during the inspection are listed in the Attachment to this report.

This inspection constituted one partial RFO sample as defined in IP 71111.20-05. The RFO extended into the next inspection interval and completion of the sample requirements will be documented during the next inspection period.

b. Findings

.1 Boric Acid Identified on Reactor Vessel Head

On March 12, 2010, during planned ultra-sonic testing on the control rod drive nozzles penetrating the reactor vessel closure head, the licensee identified that two of the nozzles examined to that date did not meet applicable acceptance criteria. Each of the nozzles had similar indications that appeared to penetrate into the nozzle walls. The licensee made an 8-hour non-emergency report to the NRC on March 13, 2010, (Event Number 45764) in accordance with 10 CFR 50.72(b)(3)(ii)(A) and stated that examinations were ongoing on other nozzles. On March 15, 2010, the licensee provided an update to the NRC that said that examinations were ongoing and that initial visual examinations of the top of the reactor vessel head found a small amount of dried boric acid and that two nozzles have indications of RCS leakage. On March 17, 2010, the NRC issued a news release stating that a special inspection team had been dispatched to the site and would follow activities associated with the examination and subsequent repair of the reactor vessel head. Details associated with that inspection will be documented in a separate inspection report.

.2 Unattended Transient Combustibles Within Exclusion Area for the Dry Fuel Storage Modules

Introduction: The inspectors identified a finding involving an NCV of 10 CFR 72.212, "Conditions of general license issued under 72.210," having very low safety significance (Green) for non-compliance with transient combustible material control procedures required for the Davis-Besse spent fuel dry horizontal storage modules (HSMs).

Description: On March 12, 2010, the inspectors noted a mobile crane and a utility truck parked and unattended within an area designated by signs as a 75 foot exclusion area around the HSMs. The truck did have at least 3 compressed gas bottles (2 oxygen bottles and 1 propane bottle). The issues identified were not in compliance with the licensee's procedures, specifically DB-FP-7 for control of combustible transient material. Control of transient combustible material was required to ensure conformance with temperature limitations for the HSMs as outlined in the NRC-issued HSM Certificate of Compliance. Procedure DB-FP-7 specifically requires that flammable/combustible liquids and gasses shall not be left unattended within 75 feet of the HSMs and that vehicles within 75 feet of the HSMs shall have a vehicle attendant at all times. The attendant shall maintain the capability to contact the Control Room in the event of a fire and that the attendant shall have a portable fire extinguisher. Additionally, the procedure discussed the potential for trailer tires to become projectiles in a fire event and discussed a 100 foot exclusion area. The inspectors considered that the tires on the mobile crane, with the onboard combustible fluids, were potentially capable of becoming energetic projectiles.

Upon notification by the inspectors the licensee immediately investigated the conditions and generated CR 10-73290 to document the issue. The licensee determined that the contractors responsible for the vehicles had been briefed about the requirements for an attendant at all times, but the contractor supervisor had forgotten to assign an attendant during the crew's lunch period. Immediate corrective action documented by the licensee was to reaffirm the requirements with the contractor supervisor. The licensee also determined the utility truck, although inside the exclusion area signs, was a few feet beyond 75 feet from the HSMs; the signs had been moved to allow entrance by the utility truck. The mobile crane was inside of the 75 foot exclusion area.

Analysis: The inspectors determined that the failure to follow fire protection procedures developed for control of transient combustible material in close proximity to the HSMs was a performance deficiency that warranted a significance evaluation. This finding was greater than minor because it was associated with the protection against potential fire damage to the HSMs and, if left uncorrected, would become a more significant safety concern since repeated presence of unattended combustible material in the vicinity of the HSMs increased the vulnerability of the HSMs to damage from a fire. Additionally, contractor personnel not adhering to station procedures, if left uncorrected, could become a more significant issue. The inspectors determined that the finding was not suitable for significance determination process (SDP) evaluation because the noncompliance did not involve permanently installed plant equipment. Therefore, this finding was reviewed by regional management, in accordance with IMC 0609, Appendix M and determined to be of very low safety significance (Green). The unattended time was short, and the equipment was placed in a location easily visible to plant locations that are always manned. These personnel could have notified the control room to dispatch the plant fire brigade early after initiation of a fire. The primary cause of this finding was related to the cross-cutting area of Human Performance because licensee personnel did not ensure sufficient oversight of contractor work activities to ensure compliance with site procedures associated with protection of the dry spent fuel storage modules (H.4.(c)).

Enforcement: 10 CFR 72.212 "Conditions of general license issued under 72.210," section b(9) states, in part, that the licensee shall "Conduct activities related to storage of spent fuel under this general license only in accordance with written procedures." Procedure DB-FP-00007, "Control of Transient Combustibles," provided, among other things, controls for limiting transient combustible material in the area around the HSMs. Contrary to the above, transient combustibles were left unattended near the dry spent fuel storage pad inside the area prohibited by station procedures. Once identified, the licensee initiated actions to properly control the transient combustible material and entered the issue into its CAP as CR 10-73290. However, because this violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000346/2010002-04, Unattended Transient Combustibles Within Dry Fuel Storage Exclusion Area)

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- DB-SP-3160, “AFP 2 Quarterly Test,” on January 6, 2010 (IST);
- DB-SC-3071, “Emergency Diesel Generator 2 Monthly Test,” on January 7, 2010 (routine);
- DB-SP-3136; “Decay Heat Train1 Pump and Valve Test,” on January 12, 2010 (IST);
- DB-SP-3338; “Containment Spray Train 2 Quarterly Pump and Valve Test,” on January 27, 2010 (IST);
- DB-PF-3001; “Main Steam Safety Valve Setpoint Test,” on February 26, 2010 (routine);
- DB-MM-9234; “Equipment Hatch Removal and Reinstallation,” and DB-OP-6904; “Shutdown Operations,” on March 4, 2010 (routine); and
- DB-SC-3121, “SFAS Train 2 Integrated Response Time Test,” on March 1 and 2, 2010 (routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and

- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted four routine surveillance testing samples and three inservice testing samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constitutes a partial sample as defined in IP 71124.01-5.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings of significance were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors evaluated if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors assessed whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last two radiological surveys from three to six selected plant areas. The inspectors evaluated the thoroughness and frequency of the surveys is appropriate for the given radiological hazard.

The inspectors conducted walk-downs of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation:

- alloy 600 weld overlay activities;
- steam generator jumps including mobilization, dam installations, platform work; and
- core flood nozzles shielding activities/platform inside the vessel bio-shield.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- presence of alpha emitters;
- potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (this evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel);
- hazards associated with work activities that could suddenly and severely increase radiological conditions; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors assessed whether the licensee had a program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings of significance were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected three to five containers holding nonexempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, “Labeling Containers,” or met the requirements of 10 CFR 20.1905(g).

The inspectors reviewed the following radiation work permits (RWPs) used to access high radiation areas (HRAs) and evaluated the specified work control instructions or control barriers:

- RWP 2010-5600; Task 6 Equipment Setup and Demobilization, Temp Power; WSI Control Operator; Task 8 Piping and Interference Removal and Reinstallation in Support of Alloy 600 Weld Overlay of RCP Suction and Discharge and Cold Leg Drain;
- RWP 2010-5104; Reactor Head Disassembly and Reassembly Work Activities; Removed and Replaced CRDM Bulkhead Connections and Related Cables; Removal and Installation and Included Installation Supports, Snubbers and CRDM Shield Platform Assembly Works;
- RWP 2010-5301; Install and Remove Steam Generator Nozzle Dams and ALARA Briefs and Mockup Training was Required for All Tasks;
- RWP 2010-5601; Containment Activities that Included Cutting Penetrations in Concrete to Access North Core Flood and South Core Flood Nozzles; Install and Removed Work Platforms; Restore Access Opening Including Block and Closure Plates; and
- RWP 2010-5602; Alloy -600 Weld Overlay North and South Core Flood Nozzles Including Support Activities Such as Scaffold and Shielding Installation and Removal.

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter (EPD) alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's EPD noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP and dose evaluations were conducted as appropriate.

For those work activities selected in 2RS1.2.a, the inspectors assessed whether the licensee had established a means to inform workers of charges that could significantly impact their occupational dose.

b. Findings

No findings of significance were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the RCA, and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures. The inspectors also reviewed whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

b. Findings

No findings of significance were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of EPDs in high noise areas as HRA monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee is properly employing an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 2010-5301; Install and Remove Steam Generator Nozzle Dams and ALARA Briefs and Mockup Training was Required for All Tasks;
- RWP 2010-5601; Containment Activities that Included Cutting Penetrations in Concrete to Access North Core Flood and South Core Flood Nozzles; Install and Removed Work Platforms; Restore Access Opening Including Block and Closure Plates; and
- RWP 2010-5602; Alloy-600 Weld Overlay North and South Core Flood Nozzles Including Support Activities Such as Scaffold and Shielding Installation and Removal.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potentials for significant airborne levels (e.g., grinding, grit blasting, cutting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air (HEPA) ventilation system operation for selected airborne radioactive material areas.

The inspectors inspected the posting and physical controls for selected HRAs and very high radiation areas (VHRAs), to verify conformance with the Occupational PI.

b. Findings

No findings of significance were identified

.6 Risk-Significant High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the Radiation Protection Manager (RPM) the controls and procedures for high-risk HRAs and VHRAs. The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors reviewed special areas that have the potential to become VHRAs during certain plant operations (e.g., pressurized-water reactor (PWR) thimble withdrawal into the reactor cavity sump). The inspectors discussed these areas with first-line health physics (HP) supervisors (or equivalent positions having backshift HP oversight authority) to assess whether the communication beforehand with the HP group would allow for corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization. The inspectors evaluated licensee controls for VHRAs, and areas with the potential to become a VHRA, and assessed whether or not an individual was not able to gain unauthorized access to the VHRA.

b. Findings

No findings of significance were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the significant radiological conditions in their workplace and the RWP controls/limits in place and that their performance reflects the level of radiological hazards present.

The inspectors reviewed a maximum of ten radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the RPM any problems with the corrective actions planned or taken.

b. Findings

No findings of significance were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technician with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed a maximum of ten radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings of significance were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator (PI) for the period from the first quarter of 2009 through the fourth quarter of 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of the first quarter of 2009 through the fourth quarter of 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and were identified.

This inspection constituted one unplanned scrams per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant System (RCS) Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage performance indicator for the period from the first quarter of 2009 through the fourth quarter of 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period of the first quarter of 2009 through the fourth quarter of 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified.

This inspection constituted one RCS leakage sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached List of Documents Reviewed.

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

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily CR packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

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

.1 Identification of Sample Well Water With Tritium Contamination

a. Inspection Scope

The inspectors reviewed the plant's response to identification of a water sample from onsite protected area sample well MW-105A taken on October 13, 2009, with a tritium concentration of 2,285 picocuries per liter (pCi/L). The sample results were available on December 29, 2009. The sample results were documented in CR 09-69415. Another sample was taken from well MW-105A on January 6, 2010, and the results were available on January 21, 2010. That sample had a tritium concentration of 3,799 pCi/l and was documented in CR 10-70347. The sampling of the well was done as part of the licensee's voluntary groundwater monitoring initiative. The inspectors reviewed the licensee's compliance to their stated offsite agency reporting requirements. The inspectors also reviewed the licensee's stated belief that the increasing trend of tritium can be attributed to a pipe leak that occurred in 2008; the inspectors documented their review of the licensee's response to that leak in Inspection Report 05000346/2008005. Documents reviewed in this inspection are listed in the Attachment.

This event follow-up review constituted one sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified. The sample results were above the 2000 pCi/L groundwater monitoring program threshold for making courtesy notifications to state and local government officials and NRC resident inspectors. The sample results were below the 30,000 pCi/L reporting limit in the licensee's Offsite Dose Calculation Manual.

.2 Identification of Water With Tritium From Failed Drain Line

a. Inspection Scope

On March 1, 2010, the licensee was pumping water from their main condenser east pit sump to their settling basins and observed water coming out of the ground, inside the Protected Area, in an area close to the routing of the discharge line. The discharge pipe in use was designed for use during outages. Upon observation of the water, the sump pump was stopped. An analysis of a grab sample of the sump water showed an activity level of 24,000 picocuries per liter (pCi/L) of tritium. Estimates are that the sump was pumped twice into the outage discharge line and resulted in a spill of in excess of 100 gallons. The inspectors reviewed the licensee's compliance to their stated offsite agency reporting requirements. Documents reviewed as part of this inspection are listed in the Attachment.

This event follow-up review constituted one sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified. The sample results were above the 2000 pCi/L groundwater monitoring program threshold for making courtesy notifications to state and local government officials and NRC resident inspectors. The sample results were below the 30,000 pCi/L reporting limit in the licensee's Offsite Dose Calculation Manual.

.3 Identification and Categorization of Nuclear Fuel Assembly Pin Failures

a. Inspection Scope

During nuclear fuel off-loading of the reactor core during RFO 16, the licensee, using fuel sipping, identified several assemblies as potentially containing fuel pin defects. Three of those assemblies were scheduled to be re-inserted in the core. The inspectors reviewed the licensee's requirements that positively identified fuel pin defects and their requirements for addressing the fuel pin defects prior to fuel assembly re-insertion in the reactor core. The inspectors also reviewed CR 10-74001, "Eddy Current Testing of Fuel Assembly NJ14HD Identified Clad Degradation," and recommendations for onsite investigations to characterize the extent of corrosion product deposition on interior fuel pins and the potential for issues associated with crud-induced localized corrosion during the next operating cycle.

This event follow-up review constituted one sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Licensee Activities and Meetings

In addition to regularly attending daily plant status meetings, the inspectors observed select portions of other licensee activities and meetings and met with licensee personnel to discuss various topics. The activities that were sampled included:

- project review meeting on February 1, 2010;
- operator and chemistry work-around closeout work status meeting on February 1, 2010;
- end-of-cycle reactivity management review meeting on February 10, 2010;
- radiation protection morning planning meeting on February 17, 2010;
- Davis-Besse site all-hands meeting on February 5, 2010; and
- plant review committee meetings on March 11, 2010, to discuss changes to the Offsite Dose Calculation Manual and supporting calculations for the radionuclide concentrations allowed in the borated water storage tank.

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

a. Inspection Scope

As documented in Section 1R19, the inspectors confirmed the acceptability of the described licensee's actions associated with high pressure injection train 1 in the auxiliary building. This inspection effort counts towards the completion of TI 2515/177 which will be closed on a later inspection report.

On March 17 and 22, 2010, the inspectors conducted a walkdown of the high pressure injection system inside containment in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The licensee's walkdown results were not made available to the inspectors at the end of the inspection interval but will be reviewed in the next inspection interval.

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

- high point vents were identified and were situated to vent the high points of the lines;

- there were no apparent line high points that did not have venting capability;
- all pipes and fittings were clearly shown; and
- the drawings were up-to-date with respect to recent hardware changes.

The inspectors verified that Piping and Instrumentation Diagrams (P&IDs) accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and that there were no apparent discrepancies between as-built configurations, the isometric drawings, and the P&IDs. (TI 2515/177, Section 04.02.b)

Documents reviewed are listed in the Attachment to this report.

b. Findings

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

.3 Licensee Event Report (LER) 05000346/2009001-00

a. Inspection Scope

The inspectors reviewed the licensee's classification of LER 2009-001, "Containment Air Coolers Fans Inoperable Due to Misapplication of Potter and Brumfield Rotary Relays," as an event that was reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by the TSs.

b. Findings

Introduction: The inspectors determined that the licensee's classification that both trains of containment air coolers were inoperable was potentially classifiable under 10 CFR 50.72(b)(3)(v) and 50.73(a)(2)(v) as an event or condition that could have prevented fulfillment of a safety function. The failure to report a safety system function failure caused by relay misapplication is considered a URI pending further review by the licensee's staff and the inspectors.

Description: On October 13, 2009, the licensee identified that they had misapplied Potter and Brumfield Rotary Relays in the control circuitry of their containment air coolers and that this misapplication potentially affected the ability of the coolers to automatically switch from the normal high speed fan operation to post-ECCS actuation low speed fan operation. The licensee manually switched the fans to slow speed operation and declared the coolers operable. The inspectors reviewed the event and determined that the misapplication of the relays was a finding which was documented in Inspection Report 05000346/2009005 as NCV 05000346/2009005-03.

On December 14, 2009, the licensee submitted LER 2009-001 in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by the TSs. The inspectors' review of the LER determined that with both trains of containment air coolers the event was potentially classifiable under 10 CFR 50.72(b)(3)(v) and 50.73(a)(2)(v) as an event or condition that could have prevented fulfillment of a safety function. This inspectors' potential classification was discussed with licensee personnel and with NRC sections involved in reviewing classifications of LERs. At the end of the inspection period, the licensee and the inspectors had not completed their review of the data associated with classification of this event. Pending further review of the licensee's evaluation and supporting documentation by the inspectors to determine if this constitutes a failure to report a safety system functional failure, the issue is considered a URI 05000346/20100002-05, Potential Missed Reporting Requirement for Inoperable Containment Air Coolers).

.4 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 6, 2010, the inspectors presented the inspection results to Mr. B. Allen and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- the results of Radiological Hazard Assessment and Exposure Controls inspection with the Site Maintenance Director, Mr. J. Dominy, on March 19, 2010; and
- the results of the inservice inspection with Site Vice President, B. Allen, on March 25, 2010.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Allen, Site Vice President
B. Boles, Director, Site Operations
J. Dominy, Director, Site Maintenance
V. Kaminskas, Director, Site Engineering
D. Noble, Radiation Protection Manager
C. Price, Director, Site Performance Improvement
G. Wolf, Regulatory Compliance Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000346/2010002-03	URI	Inoperability of Steam-Feedwater Rupture Control System
05000346/2010002-05	URI	Potential Missed Reporting Requirement for Inoperable Containment Air Coolers

Opened and Closed

05000346/2010002-01	FIN	Failure to Maintain Station Blackout Diesel Generator Output Cables in an Environment Consistent with Design
05000346/2010002-02	NCV	Inadequate Post-Maintenance Testing of ECCS Room Cooler
05000346/2010002-04	NCV	Unattended Transient Combustibles Within Dry Fuel Storage Exclusion Area

Closed

05000346/2009005-01	URI	Ability of Medium Voltage Cable from Blackout Diesel to Function Long Term in Water Submerged State

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

Procedures:
- NOP-OP-1007; Risk Management; Revision 6
- DB-OP-06913; Seasonal Plant Preparation Checklist; Revision 20
- RA-EP-02870; Station Isolation; Revision 4

1R04 Equipment Alignment

Procedures:
- DB-OP-6011; High Pressure Injection System; Revision 23
- DB-OP-6012; Decay Heat and Low Pressure Injection Operating Procedure; Revision 43

Drawings:
- OS-3; High Pressure Injection System; Revision 32
- OS-4, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 45

1R05 Fire Protection

Condition Reports:
- 10-72039; Fire Brigade Response to Smoke in the Wet Wash Facility
- 10-72045; Configuration Control in Wet Wash Facility Concern

Procedures:
- DB-FP-7; Control of Transient Combustibles; Revision 8
- DB-OP-2529; Fire Procedure; Revision 5
- PFP-AB-208; No. 1 Mechanical Penetration Room, Room 208, Fire Area AB; Revision 5
- PFP-AB-500; Radwaste and Fuel Handling Areas, Room 500, Fire Area EE; Revision 3
- PFP-AB-501; Radwaste Exhaust Equipment and Main Station Exhaust Fan Room, Room 501, Fire Area EE; Revision 3
- PFP-AB-515; Purge Exhaust Equipment Room, Room 515, Fire Area EE; Revision 3

Drawings:
- A-221F; Fire Protection, General Floor Plan El. 545’ and 555’; Revision 9
- A-222F; Fire Protection, General Floor Plan El. 565’; Revision 15
- A-224F; Fire Protection, General Floor Plan El. 603’; Revision 22
- A-225F; Fire Protection, General Floor Plan El. 623’; Revision 17

1R06 Flooding

Condition Reports:
- 02-08545; SHRR – Closed CR Review for 13.8 KV Cables in Water

- 05-01499; NRC PI&R LOG1-4634 Corrective Actions Adequacy for Underground Wetted Cables
- 06-00069; Corrective Actions for CR 1999-1648 Ineffective
- 06-11583; CDBI: Preventive Maintenance of Electrical Manholes
- 09-67489; NRC Concern – Submerged Cables in Electrical Manhole MH3045
- 10-70666; Electrical Manhole MH3045 – Cables Submerged

Procedures:
- EN-DP-1130; Structure, System, and Component Quality Classification; Revision 10
- NORM-ER-3112; Cable Monitoring; Revision 1

Work Orders:
- 200158879; Replace Blackout Diesel Output Cables (In Planning)

Other:
- DB-Rev-07-0084; New PM Request for Periodic Testing of Blackout Diesel Output Cable; January 2007
- DB-Rev-08-0178; New PM Request for Manholes, 3040, 3043, 3044, 3045, 3046; March 2008
- DB-Rev-09-2056; New PM Request for De-Watering Manhole 3045; December 2009
- Design Report for Mod 89-0109, Supplement 3; Station Blackout Diesel; June 17, 1991
- Quality Classification List, Page 9; Revision 18
- Maintenance Rule Program Manual; SBODG Scoping Sheet; Revision 28
- Okonite Letter; Certification for Davis-Besse P.O. 061-Q62475A-E1; September 1, 1981

1R07 Heat Sink Performance

Condition Reports:
- 10-70072; Inadequate Flow Found to DB-E42-4 During DB-PF-04736
- 10-70078; Evaluate Post Maintenance Testing of 4 ECCS Room Cooler
- 10-70082; SW110, ECCS Room Cooler 4 Inlet, Stem to Wedge Is Likely Separated

Procedures:
- DB-PF-04736; ECCS Room Cooler Monitoring Test; Revision 4
- NORM-ER-3201; Room Coolers; Revision 0

Work Orders:
- 200375179; E42-4 – Inspect, Clean Cooler

Other:
- EPRI NP-7552; Heat Exchanger Performance Monitoring Guidelines; December 1991

1R08 Inservice Inspection Activities (71111.08P)

Condition Reports:
- 10-74043; 16RFO OTSG Eddy Current Visual Examination for Leakage of Reactor Head Penetrations dated Examination Identified Defects in the OTSG Tubing
- 10-73394; Cracked OTSG Welded Plug

Other:
- 54-ISI-240; Visible Solvent Removable Liquid Penetrant Examination Procedure; Revision 44

- 54-ISI-367-11; Visual Examination for Leakage of Reactor Head Penetrations; dated January 28, 2010
- 54-ISI-835; Ultrasonic Inspection of Ferritic Piping Welds; Revision 12
- 51-5001484-006; Qualified Eddy Current Examination Techniques for Davis-Besse; dated February 2, 2010
- 51-9114644-000; Davis Besse Degradation Assessment for 16th Refueling Outage; dated January 22, 2010

1R11 Licensed Operator Requalification Program

Procedures:
- DBBP-TRAN-0017; Conduct of Simulator Training; Revision 2
- DBBP-TRAN-0502; Development and Conduct of Simulator Training Simulator Evaluations; Revision 3
- DB-OP-02527; Loss of Decay Heat Removal; Revision 12
- RA-EP-01500; Emergency Classification; Revision 11

Other:
- OTLC-201001 DB-S101; Unannounced Simulator Outage Scenario with NLOC, Loss of DHR; Revision 0

1R12 Maintenance Effectiveness

Condition Reports:
- 08-33163; Plant Computer MUX Power Supply Failure
- 08-46705; Plant Computer Analog Multiplexer Failure
- 08-49133; RPS Channel 2 Reactor Trip Module – Intermittent Test Switch Operation
- 08-49708; Plant Computer Problems
- 09-55460; Unexpected Trip of RPS Channel 2
- 09-56472; RPS Ch. 4 Intermediate Range Data Unsat
- 09-57560; Failure of NSSDATA Program
- 09-62297; Human Performance Error During ARTS Jumper Installation
- 09-66895; Power Pumps Bistable Tripped With One Trip Input From Field
- 09-69501; Failure of Plant Computer Multiplexer Communication

Procedures:
- DB-PF-00003; Maintenance Rule; Revision 28
- DB-SC-4112; DSS Channel 1 Functional Test; Revision 4
- NOP-ER-3004; FENOC Maintenance Rule Program; Revision 1

Work Orders:
- 200040012; Adjust Meter +15 VDC PWR Supply Ind.
- 200296179; RC2A2B – Repack Valve
- 200310181; FTRC1A1 – Replace 5 Valve Manifold
- 200352398; EWR 01-0385-01; Replace PPCS MUX C4601
- 200386938; PSL-4535A Leaking
- 200398832; EWR 01-0385-003; Replace C752 Temporary MUX

Other:
- System Health Report-Plant Computer-System 31-01; Fourth Quarter 2009
- System Health Report-Reactor Protection System 58-01; Fourth Quarter 2009

- EPIX Failure Reports for January 1, 2007, through December 2009; Davis-Besse Plant Computer System; February 2, 2010
- USAR Section 7.2; Reactor Protection System

1R13 Maintenance Risk Assessments and Emergent Work Control

Condition Reports:
- 10-70197; Cable for MP58-1 Minimum Band Radius Violated
- 10-71436; Steam Generator 2 BTU Limit is Periodically Coming Into Alarm During Tave Reduction
- 10-71489; Tave Reductions Are Challenging Reactor Safety Margins
- 10-71496; 10CFR50.36©(2)(ii) Concerns for Tave Reductions
- 10-72839; HIS DH1B Pushbutton Sticks in the Close Position

Procedures:
- DBBP-ESAF-2010; Hydrogen Safety While Sampling the Reactor Coolant System; Revision 0
- NG-DB-00117; Shutdown Defense in Depth Assessment; Revision 6
- NOBP-OP-0007; Conduct of Infrequently Performed Tests or Evolutions; Revision 2
- NOP-OP-1005; Shutdown Defense in Depth; Revision 12
- NOP-OP-1007; Risk Management; Revision 6
- 03-1221650; Areva Reactor Vessel Plenum Removal and Reinstallation Procedure; Revision 8
- 03-9060727; Areva Davis-Besse Reactor Vessel Head Removal; Revision 1

Work Orders:
- 20040973; Adjust modules for ICS BTU limit alarm

Other:
- 16RFO Shutdown Defense in Depth Report; dated January 15, 2010
- DB-SA-10-011; Shutdown Defense in Depth Independent Assessment
- ECP 10-0083-000; Adjust Gain Setting for ICS Btu Limit Modules During Tave Reduction Maneuver; Revision 0
- ECP 10-0083-001; Adjust Gain Setting for ICS Btu Limit Modules During Tave Reduction Maneuver: Documents to be Issued; Revision 0
- Key Shutdown Defense in Depth Function Status; for the weeks of February 28, 2010, and March 7, 2010
- Weekly Maintenance Risk Summary for Week of January 18, 2010; Revision 0 and 1
- Weekly Maintenance Risk Summary for Week of January 25, 2010; Revision 0, 1, and 2
- Weekly Maintenance Risk Summary for Week of February 15, 2010; Revision 0 and 1
- Instrument Data Sheet: FW/5-1-7; Multiplier – Gains Reactor Coolant Flow Limit-Loop 1; Revision 3
- Instrument Data Sheet: FW/5-1-15; Multiplier – Gains Reactor Coolant Flow Limit-Loop 2; Revision 3
- DSP-10-1; Infrequently Performed Tests or Evolutions – Isolation of Letdown Coolers for Leakage Identification; Memo from Director – Site Operations; January 19, 2010
- Operations Evolution Order; Isolate Letdown Coolers for CCW Leak Identification; dated January 19, 2010
- Operations Evolution Order; H2 Concentration Check and Purge During RCS Drain; dated March 6, 2010
- Operations Checklist for Protected Equipment Postings; for March 8, 2010
- Reactivity Plan Review Package; February 26-28, 2010, End-of-Cycle 16 Shutdown

- OTLC-JIT-DB-10010; Training Material: Tave Reduction for Cycle 16 EOL; Revision 0
- Unit Operating Logs; dated March 5 through March 7, 2010

1R15 Operability Evaluations

Condition Reports:
- 10-70072; Inadequate Flow Found to DB-E42-4 During DB-PF-04736
- 10-70082; SW110, ECCS Room Cooler 4 Inlet, Stem to Wedge is Likely Separated
- 10-70197; Cable for MP58-1 Minimum Band Radius Violated
- 10-72359; USAR and Technical Specification Steam Flow Path May Not Exist

Procedures:
- DB-ME-9506; Very Low Frequency Insulation Testing of Electrical Cables; Revision 0
- DB-SP-3005; Service Water Train 1 Cold Forebay Design Flow Verification; Revision 1
- DB-SP-3006; Service Water Train 2 Cold Forebay Design Flow Verification; Revision 1

Work Orders:
- 200267002; PM 6108 MP 58-1 Motor Testing
- 200297815; SP3218-004.005 PO58-01
- 200372776; PM 9332-Cable Inspection and Test 1PAC111A
- 200375179; E42-4 – Inspect, Clean Cooler

Calculations:
- C-NSA-011.01-003; Allowable Service Water Flow Diversion During Cold Weather; Revision 2
- C-NSA-011.01-016; Service Water System Design Basis Flowrate Analysis and Testing Requirements; Revision 1
- C-NSA-011.01-018; Analysis of Service Water System Online Flow Balance Test Data, 9-15-07; Revision 0
- C-NSA-011.01-019; Analysis of Service Water System Online Flow Balance Test Data, 12-23-09; Revision 1
- C-NSA-032.02-006; ECCS Room Heatup During Post LOCA; Revision 3

Other:
- Standing Order 10-001; Bounding Limitations for Service Water due to ECCS Room Cooler 4 Inadequate Flow
- Tan Delta Test Results for Cable 1PAC111A; January 19, 2010
- Technical Specification 3.4.7; RCS Loops- Mode 5, Loops Filled
- Unit Operating Logs; dated January 14, 2010, through January 17, 2010
- USAR Section 6.3; Emergency Core Cooling Systems
- USAR Section 9.2.1; Service Water System

1R18 Plant Modifications

Condition Reports:
- 10-70082; SW110, ECCS Room Cooler 4 Inlet, Stem to Wedge Is Likely Separated
- 10-70291; AFW Pump Suction Piping Not Fabricated to ASME Section XI
- 10-70342; Duplicate Drawing Numbers for C-0831 Issued for ECP 08-0571-002
- 10-71360; SW110 Removed Valve Wedge Missing Stem Guide Ears – FME Issue
- 10-71436; Steam Generator 2 BTU Limit is Periodically Coming Into Alarm During Tave Reduction
- 10-71489; Tave Reductions Are Challenging Reactor Safety Margins

- 10-71496; 10CFR50.36©(2)(ii) Concerns for Tave Reductions
- 10-71562; Feedwater Silica Concentration Exceeds Action Level 2

Procedures:
- DB-PF-6703, Curve CC4.3; Pressurizer Operations; Revision 15
- DB-PF-6703, Curve CC7.9; Steam Generator BTU Limits; Revision 15
- DB-OP-6902; Power Operations; Revision 28
- DB-OP-6261; Service Water System; Revision 40

Work Orders:
- 200398227; Calibrate ICS Modules for Cycle 16 Tave Reduction
- 200400068; Repair or Temp Mod for SW110

Drawings:
- M-553-179-3; ICS Reactor Control Analog Logic; Revision 13
- OS-0020, Sheet 1; Service Water System; Revision 79

Other:
- ECP 08-0571-000; Reroute AFW Common Suction Piping to Resolve Non-Conformance; Revision 3
- ECP 08-0571-002; AFW – Installation of Piping and Supports; Revision 4
- ECP 09-0550-000; End-of-Cycle Reactor Coolant System Average Temperature Reduction and Associated Alarm Setpoint Changes; Revision 0
- ECP 10-0029-000; Remove Valve Internals from SW110; Revision 0
- ECP 10-0029-001; SW110 Valve Internals Removed
- USAR Appendix 4B; Cycle 16 Reload Report; January 2008
- USAR Section 9.2.1; Service Water System
- Periodic Reactivity Plan-Reactor Operating Guidance for February 16, 2010, to End-of-Cycle 16; February 15, 2010;

<u>1R19 Post Maintenance Testing</u>

Condition Reports:
- 03-08917; SFRCS can re-energize in a blocked condition
- 09-59292; ECCS Room Cooler #2 Shows Marginal Signs of Biofouling
- 09-61572; ECCS Room Coolers #1 and #2 Not Meeting Acceptance Criteria
- 10-70197; Cable for MP58-1 Minimum Band Radius Violated
- 10-70245; Unqualified EQ Termination on HP2C
- 10-70246; Unqualified EQ Termination on HP2D
- 10-70320; Gas Voids Detected Upstream of HP60 and HP61
- 10-71008; Flow Obstructed from SW185
- 10-71044; #1 ECCS Room Cooler SW Bolting Not Marked
- 10-71101; ECCS Room Cooler 2 Not Meeting Acceptance Criteria
- 10-72446; SFRCS Channel 4 energized in a blocked condition as indicated by HIS100C
- 10-72515; Could not perform DB-SC-03262 due to issue with turbine inputs to SFRCS Ch 2 & 4
- 10-72585; Wrong relays received from stock code 27002206 for SFRCS relay replacement
- 10-72588; SFRCS Ch 4 block light for FW601 did not work as expected during testing
- 10-72647; Non Q part ordered and installed under SFRCS order

Procedures:
- DB-OP-06406; Steam and Feedwater Rupture Control System Operation Procedure; Revision 11
- DB-PF-04736; ECCS Room Cooler Monitoring Test; Revision 4
- DB-SC 03262; Integrated Test of SFRCS Actuation Channel 2; Revision 8
- DB-SP-3212; Venting of ECCS Piping; Revision 14
- DB-SP-3218; HPI Train 1 Pump and Valve Test; Revision 22
- DB-SP-4152; AFPT 1 HSS and Overspeed Trip; Revision 14
- DB-SP-4212; Venting of ECCS Piping – SOER 97-1; Revision 8

Work Orders:
- 200237418; PM 6653: Replace Relays SFRCS Train 2
- 200237559; PM 6551 MC31-2 Vendor Clean/Inspect/Refurbish
- 200260606; HP29 – Repair Valve Leak-by
- 200375167; E42-1 – Inspect, Clean Cooler
- 200375178; E42-2 – Inspect, Clean Cooler

Drawings:
- 7749-FSK-M-CCB-19-4; Test Line form Make –up Pump to 3" CCB-19; Revision 3
- ISIM2-233D, Sheet 2; HP Injection System, Auxiliary Building; Revision 2
- ISIM2-233D, Sheet 3; HP Injection System, Auxiliary Building; Revision 7
- OS-3; High Pressure Injection System; Revision 32

Other:
- Clearance NDB-Sub052-01-004; Isolate and Drain Portion of High Pressure Injection Train 1
- Operations Evolution Order; HP29 Piping Refill; January 6, 2010

<u>1R20 Outage Activities</u>

Condition Reports:
- 10-73290; Unattended Vehicle in the 75 Foot Exclusion Zone Near the Dry Fuel Storage Pad
- 10-73416; 16RFO – Fuel Assembly NJ10L7 – Fuel Defect Identified Via In-Mast Sipping
- 10-73917; Overly Restrictive Exclusion Zone for Dry Fuel Storage Horizontal Storage Module

Procedures:
- DB-FP-7; Control of Transient Combustibles
- DB-OP-6000; Filling and Venting the Reactor Coolant System; Revision 21
- DB-OP-6001; Boron Concentration Control; Revision 16
- DB-OP-6005; RC Pump Operation; Revision 24
- DB-OP-6012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 43
- DB-OP-6903; Plant Cooldown; Revision 34
- NG-DB-117; Shutdown Defense In Depth Assessment; Revision 6
- NOP-OP-1005; Shutdown Defense in Depth; Revision 12
- NOP-OP-3502; FENOC Shutdown Chemistry Program; Revision 1

Drawings:
- ISIM2-234A; Inservice Inspection Isometric L.P. Injection-Core Flooding Sys. Cont. Bldg.; Revision 2

Other:
- FENOC Memo dated 1/15/2010; Subject: 16th Refueling Outage Milestone 64-4 Closure Shutdown Defense in Depth Report Issued
- Reactor Plant Event Notification Worksheet; 3/13/10 (Event Date 3/12/10)
- DB-SA-10-011; Shutdown Defense in Depth Independent Assessment
- Certificate of Compliance 1004; Dry Spent Fuel Storage Casks Standardized NUHOMS-24P and NUHOMS-52B; Revision 0

<u>1R22 Surveillance Testing</u>

Condition Reports:
- 10-69649; AFP 2 Governor Adjustment Required During DB-SP-03160
- 10-70630; Missed Test Section During Performance of #2 CTMT Spray Pump Quarterly Test
- 10-70740; Test Deficiency DB-SP-03338 CS Train 2 Valve Test
- 10-72035; Calibration Data Discrepancy on MSSV SPVD Testing Equipment
- 10-72049; Main Steam Safety Valve Testing Work Stoppage
- 10-72349; SP17B7 Exhibited Signs of Seat Leakage
- 10-72350; SP17A6 Exhibited Signs of Seat Leakage After Testing
- 10-72365; DB-SC-3121; Integrated SFAS Train 2 Unexpected Response
- 10-72366; Digital Recorder Anomalies During SFAS Integrated Tr2 Test
- 10-72379; DB-SC-3121 Integrated SFAS Train 2 Data Gathering
- 10-73124; Required CR for IST Valve Times from SFAS Integrated Train 2

Procedures:
- DB-PF-3001; Main Steam Safety Valve Setpoint Test; Revision 6 and Revision 7
- DB-PF-6704; Pump Performance Curves; Revision 25
- DB-MM-09234; Equipment Hatch Removal and Reinstallation; Revision 08
- DB-OP-06904; Shutdown Operations; Revision 31
- DB-SC-3071; Emergency Diesel Generator 2 Monthly Test; Revision 21
- DB-SC-3121; SFAS Train 2 Integrated Response Time Test; Revision 0
- DB-SP-3136; Decay Heat Train 1 Pump and Valve Test; Revision 26
- DB-SP-3160; AFP 2 Quarterly Test; Revision 22
- DB-SP-3338; Containment Spray Train 2 Quarterly Pump and Valve Test; Revision 19
- ISTB2; Pump and Valve Basis Document; Volume II – Pump Basis; Revision 10

Drawings:
- OS-017A, Sheet 1; Auxiliary Feedwater System; Revision 22

Calculations:
- C-NSA-050.03-028; Auxiliary Feedwater Minimum Performance; Revision 1

Other:
-IPTE Worksheet; Main Steam Safety Valve Setpoint Testing; dated February 24, 2010

<u>2RS1 Radiological Hazard Assessment and Exposure Controls</u> (71124.01)

Condition Reports:
- 10-72847; MG Dose Rate Alarm Received While Verifying a Valve Location LLRT
- 10-72915; LHRA Key Turnover in the Field was Not In Accordance With Procedure NOP-OP 4101
- 10-73156; Elevated Dose Rates on Core Flood Shielded Work Platforms with Incores Pulled

- 10-73933; RWP 2010-5405; Letdown Cooler Replacement Exceeds Dose Estimate
- 10-74127; Unauthorized Movement of HRA Boundaries

Procedures:
- DBBP-RP-0011; Remote Monitoring for Radiological Job Coverage; Revision 2
- DBBP-RP-0015; Pre-Outage and Outage Tasks; Revision 4
- DBBP-RP-0016; Guidance for Work In Progress (WIP) ALARA Reviews; Revision 0
- DBBP-RP-1001; Locked High and Very High Radiation Area Key Authorization; Revision 9
- NOBP-OP-4008; Response to Radiological Events; Revision 0
- NOBP-OP-4009; Radworker Expectations; Revision 0
- NOBP-OP-4110; Pre-Outage ALARA Plan; Revision 0
- DB-HP-1109; Significant Radiological Evolution Barriers; Revision 28
- DB-HP-1115: Radiation Protection Procedure for Once through Steam Generator (OTSG) Entries; Revision 13
- DB-HP-1152; Radiation Protection Procedure for Performance; Revision 12
- DB-HP-1301; Radiation Protection Procedure Use of Respiratory Protection; Revision 09
- DB-HP-1453; Radiation Protection Procedure Continuous Particulate Air Monitor AMS-3; Calibration and Use; Revision 06
- DB-HP-1802; Radiation Protection Procedure Control of Shielding; Revision 08

Other:
- RWP 2010-5104; Reactor Head Disassembly and Reassembly Work Activities; Removed and Replaced CRDM Bulkhead Connections and Related Cables; Removal and Installation and included Installation Supports, Snubbers and CRDM Shield Platform Assembly Works
- RWP 2010-5108; Refueling Decontamination Activities including Prep Work and Initial Decontamination pre-flood-up of Refuel Canal
- RWP 2010-5301; Install and Remove Steam Generator Nozzle Dams and ALARA Briefs and Mockup Training was Required for All Tasks
- RWP 2010-5600; Task 6 Equipment Setup and Demobilization, Temp Power; WSI Control Operator; Task 8 Piping and Interference Removal and Reinstallation in Support of Alloy 600 Weld Overlay of RCP Suction and Discharge and Cold Leg Drain;
- RWP 2010-5601; Containment Activities that Included Cutting Penetrations in Concrete to Access North Core Flood and South Core Flood Nozzles; Install and Removed Work Platforms; Restore Access Opening Including Block and Closure Plates
- RWP 2010-5602; Alloy -600 Weld Overlay North and South Core Flood Nozzles Including Support Activities Such as Scaffold and Shielding Installation and Removal
- RWP 2010-5405; All Tasks for Letdown Cooler Removal and Replacement

<u>4OA3 Followup of Events and Notices of Enforcement Discretion</u>

Condition Reports:
- 09-69415; Groundwater Monitoring Sample Shows Tritium Concentration of Over 2,000 pCi/l
- 10-70347; Groundwater Monitoring Sample Tritium Concentration Above 2,000 pCi/l
- 10-72255; Underground Line Break/Contaminated Leak – Tritium
- 10-72241; Discovered Water Leaking From Ground South of Intake Structure
- 10-72288; West Condenser Pit Flood Pump Not Working
- 10-72419; Discharge Piping for the Condenser Pit Sumps Is Blocked
- 10-73329; 16RFO – Fuel Assembly NJOA43 – Fuel Defect Identified Via In-Mast Sipping
- 10-73405; 16RFO – Fuel Assembly NJ0A2W Spacer Grid Damage
- 10-73406; 16RFO – Fuel-Baffle Interaction Wear On Assembly NJ14H5
- 10-73590; 16RFO – Fuel Assembly NJ14GK – Fuel Defect Identified Via In-Mast Sipping

- 10-74001; 16RFO Eddy Current Testing of Fuel Assembly NJ14HD Identified Clad Degradation

Procedures:
- NOP-LP-5003; Communication Events of Potential Public Interest; Revision 1
- NOP-OP-2012; Groundwater Monitoring; Revision 4
- NOP-OP-4705; Response to Contaminated Spills/Leaks; Revision 2
- DB-PF-6704, Curve CC14.24; Pump Design Curve for Condenser Pit Sump Pumps; Revision 25

Other:
- Offsite Dose Calculation Manual; Revision 23
- Areva Letter FAB10-235; Areva Recommendations for Investigating Potential CILC Conditions Prior to Start-Up; March 23, 2010
- Areva Letter FAB10-241; Additional Areva Recommendations for Investigating Potential CILC Conditions Prior to Start-Up; March 23, 2010

<u>4OA5 Other Activities</u>

Drawings:
- ISIM2-233E, Sheet 1; H.P. Injection System, Containment; Revision 1
- ISIM2-233E, Sheet 2; H.P. Injection System, Containment; Revision 2
- OS-3; High Pressure Injection System; Revision 32
- M-33A; High Pressure Injection System; Revision 41

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EPD	Electronic Personal Dosimeter
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
HEPA	High-Efficiency Particulate Air
HP	Health Physics
HRA	High Radiation Areas
HSM	Horizontal Storage Modules
ICS	Integrated Control System
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
ISI	Inservice Inspection
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PI	Performance Indicator
P&ID	Piping and Instrumentation Diagrams
PI&R	Problem Identification and Resolution
PM	Post Maintenance
PMT	Post-Maintenance Testing
PT	Penetrant Examination
PWR	Pressurized-Water Reactor
RCS	Reactor Coolant System
RFO	Refueling Outage
RPM	Radiation Protection Manager
RWP	Radiation Work Permit
SDP	Significance Determination Process
SFRCS	Steam-Feedwater Rupture Control System
SSC	Structures, Systems, and Components
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USAR	Updated Safety Analysis Report
UT	Ultrasonic Testing/Examination
VHRA	Very High Radiation Areas
WO	Work Order

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Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

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