UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

October 26, 2011

Mr. Barry Allen
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION INTEGRATED INSPECTION REPORT 05000346/2011004

Dear Mr. Allen:

On September 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Davis-Besse Nuclear Power Station. The enclosed report documents the results of this inspection, which were discussed on October 11, 2011, 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, three findings of very low safety significance (Green) were identified by the NRC. Each of these findings was determined to involve a violation of NRC requirements. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in Section 4OA7 of this report. However, because of the 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 (NCV), in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with 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 cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Davis-Besse Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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 05000346/2011004
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:	05000346/2011004
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	July 1, 2011, through September 30, 2011
Inspectors:	D. Kimble, Senior Resident Inspector A. Wilson, Resident Inspector J. Beavers, Emergency Preparedness Inspector M. Bielby, Senior Operations Engineer T. Bilik, Reactor Inspector T. Go, Health Physicist M. Holmberg, Reactor Inspector L. Jones, Engineering Inspector R. Jones, Resident Inspector – Fermi Station J. Maynen, Senior Physical Security Inspector J. Neurauter, Senior Engineering Inspector G. O'Dwyer, Engineering Inspector J. Steffes, Reactor Engineer
Approved by:	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000346/2011004; 7/1/2011-9/30/2011; Davis-Besse Nuclear Power Station; Operability Evaluations, Identification and Resolution of Problems, and Other 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 by the inspectors. Each of the findings was also considered a non-cited violation (NCV) 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: Initiating Events

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," were identified by the inspectors for the licensee's failure to establish adequate measures (e.g., perform a review of radiographic (RT) film weld records) to ensure material procured from a contractor (replacement control rod drive mechanism (CRDM) housings) met the American Society of Mechanical Engineers (ASME) Code. Consequently, two replacement CRDM housings were procured with RT film weld records that did not conform to the ASME Code-required film density ranges. As a corrective action, the licensee returned the affected CRDM housings to a vendor facility for completion of new RT film records prior to installation on the replacement vessel head. The violation was entered into the licensee's corrective action program (CAP) as condition report (CR) 2011-00750.

  The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Absent NRC identification, the failure to complete an adequate RT examination of welds on two CRDM housings could have allowed unacceptable weld flaws to be placed in service. Specifically, weld flaws such as cracks, can reduce the CRDM housing integrity, and place the reactor coolant system (RCS) at an increased risk for through-wall leakage and/or failure. Because this finding was identified prior to placing the CRDM housings into service, the inspectors answered "No" to the Significance Determination Process Phase 1 screening question: "Assuming worst case degradation, would the finding result in exceeding the Technical Specification (TS) limit for any RCS leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function assuming the worst case degradation?" Therefore, the finding screened as having very low safety significance. This finding had a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee staff failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Absent NRC intervention, the failure to establish adequate measures to ensure material procured from a contractor (replacement CRDM housings) met the

ASME Code would have allowed welds on two housings with non-conforming RT records to be placed into service. (H.4(c)) (Section 4OA5.1).

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” were identified by the inspectors for the licensee’s failure to control the configuration of the emergency core cooling system (ECCS) room cooler service water (SW) outlet valves in accordance with procedures. Specifically, the licensee failed to update procedures used to set the appropriate throttle position for the valves, and by using information tags to control valve position, failed to follow plant status control procedures.

  The inspectors determined that the finding was more than minor because it was associated with the Mitigating Systems Cornerstone attributes of Design Control and Configuration Control and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, an incorrect throttle position of the ECCS room cooler outlet valves could have an effect on the reliability or availability of ECCS train 2 equipment. A past operability review determined that the as-found flowrate to ECCS room coolers 1 and 2 was reduced with outlet valves SW87 and SW103 mispositioned, however, the flow was sufficient to not affect the operability of ECCS room coolers 1 and 2. Using the Phase 1 SDP worksheet for the Mitigating Systems Cornerstone, the finding screened as very low safety significance (Green) because the inspectors answered “No” to the screening questions in Table 4a. Specifically, the finding was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically, the licensee did not process a document change request to update procedures used to verify SW valve alignments. (H.2(c)) (Section 1R15)

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Procedures, Instructions, and Drawings,” were identified by the inspectors for the licensee’s failure to correct deficiencies, deviations, and/or nonconformances associated with safety-related systems, structures, and components (SSCs) in a timely manner, as required by the licensee’s Quality Assurance Program Manual (QAPM) and CAP implementing procedure. Specifically, the inspectors identified a trend on the part of the licensee to leave certain low significance/low priority corrective actions for various safety-related SSCs completely unscheduled and unaddressed, in some cases for extensive periods of time that ranged up to 8 years. The licensee initiated their own review to determine the full extent of condition of this issue, and entered the issue into their CAP as CR 2011-00385.

  The finding, which was associated with the Mitigating Systems Cornerstone, was determined to be of more than minor significance because the issue represented a programmatic deficiency associated with the licensee’s CAP that if left uncorrected would have the potential to lead to a more significant safety concern. Using the Phase 1 SDP worksheet for the Mitigating Systems Cornerstone, the inspectors determined that the finding was of very low safety significance because each of the SSC deficiencies, deviations, and/or nonconformances identified by the inspectors represented an issue that did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity. Specifically, for certain deficiencies, deviations, and/or nonconformances associated with safety-related SSCs the licensee took no corrective actions whatsoever, instead allowing the corrective actions associated with those issues to be placed in the plant's backlog of unscheduled work. (P.1(d)) (Section 4OA2.3).

B. Licensee-Identified Violation

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

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period operating at full power, and operated at or near full power for the remainder of the inspection period except for the following:

- On August 22, 2011, a failure of the No. 2 main feedwater pump auto demand signal caused power to be reduced to approximately 94 percent. The failed component was replaced and the unit returned to full power operation on August 23, 2011.
- On September 6, 2011, plant operators reduced power to approximately 98 percent in order to transfer the main feedwater flow inputs to the main feedwater flow venturis after experiencing a failure of the normal means of measuring feedwater flow by the leading edge flow meter. The issue was corrected and the unit returned to full power operation on September 7, 2011.
- On September 14, 2011, the plant experienced a trip of the 2-2 low pressure feedwater heater caused by a suspected heater tube rupture. Operators reduced power to approximately 95 percent to isolate the feedwater heater. The unit returned to full power operation on September 15, 2011.
- On September 16, 2011, power was reduced to approximately 95 percent due to an integrated control system (ICS) input mismatch. The transient was induced during post-maintenance testing (PMT) of the high pressure injection (HPI) 3B flow instrument signal monitor. Later that same day after stabilizing the plant, control room operators restored the unit to full power operation using manual controls.
- On September 30, 2011, the plant began reducing power in preparation for a mid-cycle outage to replace the reactor vessel closure head (RVCH). At midnight on September 30/October 1, 2011, the main generator output breakers were opened and the unit was taken offline.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Alignment Verifications

a. Inspection Scope

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

- emergency diesel generator (EDG) No. 1 when the station blackout diesel generator (SBODG) was unavailable during a maintenance outage the week ending July 16, 2011;

- the SBODG when EDG No. 2 was inoperable and unavailable for testing during the week ending July 23, 2011;
- HPI train No.1 when HPI train No. 2 was inoperable and unavailable during a maintenance work window the week ending July 23, 2011;
- HPI train No.2 when HPI train No. 1 was inoperable and unavailable during a maintenance work window the week ending August 6, 2011; and
- low pressure injection (LPI) train No. 2 when LPI train No. 1 was inoperable and unavailable for testing during the week ending August 27, 2011.

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, Updated Safety Analysis Report (USAR), Technical Specification (TS) requirements, outstanding work orders (WOs), 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. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the Corrective Action Program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted five partial system alignment verification samples as defined in Inspection Procedure (IP) 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

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

- No. 1 Electrical Penetration Room (Room 402, Fire Area DG);
- No. 2 Electrical Penetration Room (Room 427, Fire Area DF);
- EDG Room 1-2 (Rooms 319 and 319A, Fire Area J); and
- Component Cooling Water (CCW) Heat Exchanger and Pump Room (Room 328, Fire Area T).

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 four quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P)

.1 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the mid-cycle outage the licensee had procured a replacement reactor vessel head. The licensee completed a repair replacement activity on Control Rod Drive Housing Mechanism (CRDM) Nozzle No. 21 for the installation of a vent line for the replacement head. From August 15–18, 2011, the inspectors observed the welding conducted on the vent line pipe elbow-to-reducer nozzle weld, reviewed weld procedures and welder qualification records to determine if the activities were conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Additionally, following fabrication of the new J-groove weld at CRDM Nozzle No. 21, the inspectors reviewed the results of the dye penetrant (PT) examinations, eddy current examinations and ultrasonic examinations to determine if the examination results met the requirements of the ASME Code Section III, Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D).

The review described above was completed in accordance with IP 71111.08 Section 02.02(d) for welded repairs to the upper head penetration nozzles. This review does not constitute a full inservice inspection sample as defined in IP 71111.08-05. The remaining Sections of IP 71111.08 will be documented in a future inspection report.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On Friday, August 19, 2011, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training 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. In addition, the inspectors reviewed the licensee's training activities in conjunction with the NRC's Operating Experience Smart Sample (OpESS) FY2010–02, "Sample Selections for Reviewing Licensed Operator Examinations and Training Conducted on the Plant-Referenced Simulator." 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 were identified.

.2 Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the annual operating test, administered by the licensee from May 17 through August 11, 2011, required by 10 CFR 55.59(a). The results were compared to the thresholds established in Inspection Manual Chapter (IMC) 0609, Appendix I, "Licensed Operator Requalification SDP," to assess the overall adequacy of the licensee's licensed operator requalification and training program to meet the requirements of 10 CFR 55.59.

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following systems:

- control room emergency ventilation system; and
- SW tunnel sump pumps.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in actual or potential plant or system issues 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 structures, systems, and components (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 were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

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:

- emergent work on the ICS No. 2 main feedwater pump controller during the week ending August 27, 2011;
- emergency work on the control rod drive system after discovery of a degraded power supply and logic module during the week ending September 10, 2011;
- emergent realignment of the station's non-vital direct current (dc) loads in response to design issues during the weeks ending July 30, 2011 through August 13, 2011;
- station battery No. 2 vent fan maintenance during the weeks ending August 13, 2011, through September 9, 2011;
- planned outage preparations and equipment staging during the weeks ending August 27, 2011, through September 3, 2011; and
- emergent work in response to an ICS inputs mismatch and subsequent plant transient during the weeks ending September 17, 2011, through October 1, 2011.

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 technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed 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 six samples as defined in IP 71111.13-05.

b. Findings

No findings were identified by the inspectors. One unresolved item (URI) was identified concerning a plant transient that occurred during a maintenance activity of HPI 3B flow instrument signal monitor.

On September 15, 2011, instrumentation and controls (I&C) technicians replaced the HPI 3A and 3B flow instrument signal monitors with refurbished modules. Upon insertion of the module into the cabinet, the control room received an unexpected alarm for ICS Input Mismatch. The alarm immediately cleared and was attributed to a slight disruption in voltage when the modules were inserted. A decision was made to continue replacement activities. On September 16, 2011, I&C technicians commenced PMT of the signal monitors. During the string check of the HPI flow instrument alarms, annunciator alarm 14-4-E, "ICS Input Mismatch," was received. The alarm initially cleared, then returned. Coincident with ICS Input Mismatch alarm, the plant's ICS began reducing reactor power without any operator input. On-watch plant operators entered procedure DB-OP-02526, "Primary to Secondary Plant Upset," and went through actions of placing ICS stations in manual control. The I&C technicians performing the HPI flow instrument signal monitor refurbishment were directed to stop their activities. Reactor power initially dropped to approximately 95 percent before operators stabilized the plant, and then returned reactor power to approximately 100 percent using manual controls.

The refurbished HPI flow instrument signal monitor modules were removed from the system and taken to the I&C shop for inspection and testing, while the original signal monitor modules were reinstalled. Inspection and testing of the refurbished modules in the I&C shop did not reveal any issues. The modules have been sent to the licensee's testing laboratory for further analysis.

The inspectors continued to review the circumstances surrounding the event to determine if the issue was within the licensee's ability to foresee and correct and should have been prevented. Pending further review of the licensee's cause analysis, the issue is considered an unresolved item. (URI 05000346/2011004-01)

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- the operability of emergency core cooling system (ECCS) LPI train 2 following identification of issues with a safety-related seismic restraint, snubber No. DB-SNA87, as documented in CR 2011-97823;
- the operability of both station EDGs when outside ambient air temperature rises above 100 degrees Fahrenheit (deg F), as documented in CR 2011-97975;
- the operability of service water (SW) pump 1 following identification of a random 30 VAC voltage source in the pump strainer circuitry that had the potential to make the strainer and pump inoperable, as documented in CR 2011-97198; and
- the operability of ECCS room coolers 1 and 2 after the room cooler SW outlet valves were discovered out of the required position, as documented in CR 2011-96718.

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 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.

These operability evaluation reviews constituted four inspection samples as defined in IP 71111.15-05.

b. Findings

Introduction

A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," were identified by the inspectors for the licensee's failure to control the configuration of the ECCS room cooler SW outlet valves in accordance with procedures. Specifically, the licensee failed to update procedures used to set the appropriate throttle position for the valves, and by using information tags to control valve position, failed to follow plant status control procedures.

Description

On June 20, 2011, the inspectors discovered a discrepancy regarding the valve position of SW103, ECCS Room Cooler 2 Outlet Valve. SW103 had an information tag hanging on the valve indicating that the valve was 9¼ turns from full open position. Contrary to this, the system operating procedure valve checklist indicated that the valve needed to be throttled to 4 turns from full open. In addition, SW87, ECCS Room Cooler 1 Outlet Valve, had an information tag on the valve indicating the valve was throttled 8⅓ turns from full open, while the system operating procedure did not list a throttle position.

The inspectors raised the question regarding the correct throttle positions for SW87 and SW103. Subsequent investigation determined that neither valve was throttled correctly. After SW piping was cleaned during the 2010 refueling outage (RFO), an SW system online flow balance test was conducted to demonstrate that the flowrate to ECCS room coolers 1 and 2 was sufficient for fuel cycle 17. Work order 200240643 collected data during the flow balance test that was used to set the appropriate amount of flow through the ECCS room coolers by throttling SW87 and SW103. The results of the calculation determined that SW103 was required to be 8⅛ turns from full open and that SW87 was required to be 8 turns from full open. The data from this test was submitted to Operations in the form of a document change request to update SW procedures with the correct throttle information. However, the document change request was never processed, resulting in a failure to update the SW valve checklist procedures.

Upon discovery, Operations immediately reset each valve to the correct position to ensure operability of the ECCS room coolers. The as-found position of SW87 and SW103 was in accordance with the information tag hanging on each valve, which was not aligned with the position required by the most recent SW flow balance test. The information tags contained throttle position relating to a previous SW flow balance test that had since been superseded in 2010. In addition, it was discovered that information tags are not appropriate to use to specify component positioning. Procedure NOP-OP-1014, "Plant Status Control," states, in part, that "Information Tags shall not be used in lieu of programmatic controls such as Red Danger Tags, Caution Tags, Maintenance Deficiency Tags, etc." Also, "Information Tags should not be used to provide long-standing operating instructions or information." A note in the plant status control procedure specifically states that "Operations Information Tags do NOT specify position." Furthermore, the incorrect as-found position of the valves was found to be in alignment with the monthly SW valve verification procedure, DB-SP-03261, but not with the system operating procedure valve checklist in DB-OP-06261.

The licensee generated CR 2011-96718 to address the concerns raised by the inspectors. Corrective actions were initiated to address several issues: (1) why the document change request which provided the correct positions of SW87 and SW103 was never implemented; (2) why the SW valve verification checklist and the SW system operating procedure were not aligned with the same throttle positions; and (3) why Operations was controlling valve position using an information tag, contrary to configuration control standards. The CR evaluation was in progress, but had not been completed at the end of this inspection period.

Analysis

The inspectors reviewed this finding using the guidance contained in Appendix B, Issue Screening, of Inspection Manual Chapter 0612, Power Reactor Inspection Reports. The inspectors determined that the licensee's failure to ensure the correct position of ECCS room cooler outlet valves was a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented. The inspectors determined that the finding was more than minor because it was associated with the Mitigating Systems Cornerstone attributes of Design Control and Configuration Control and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, an incorrect throttle position of the ECCS room cooler outlet valves could have an effect on the reliability or availability of ECCS train 2 equipment. A past operability review determined that the as-found flowrate to ECCS room coolers 1 and 2 was reduced with SW87 and SW103 mispositioned, however the flow was sufficient enough not to affect the operability of ECCS room coolers 1 and 2.

The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, using the Phase 1 SDP worksheet for the Mitigating Systems Cornerstone. The finding screened as very low safety significance (Green) because the inspectors answered "No" to the screening questions in Table 4a. Specifically, the finding was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

This finding has a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee did not process a document change request to update procedures used to verify SW valve alignments. (H.2(c))

Enforcement

Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, and shall be accomplished in accordance with these instructions, procedures, and drawings. Contrary to the above, the licensee had not prescribed the appropriate throttle position for SW87 and SW103 into procedures and failed to properly control configuration of each valve. Until discovered on June 20, 2011, the valves had been controlled in an incorrect position dating back to the April 2010 flow balance test. The licensee included this issue in their CAP as CR 2011-96718. An immediate corrective action was taken to return the ECCS room cooler outlet valves to the correct throttle position. 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 the Enforcement Policy. (NCV 05000346/2011004-02)

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary equipment use against the licensee's criteria for performing modifications to the plant:

- use of a temporary diesel-driven air compressor to supply station and instrument air loads during summer months when the turbine plant cooling water system was challenged by elevated ambient temperatures.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the plant's design and licensing basis to verify that the application did not affect the operability or availability of the affected systems. The inspectors observed ongoing and completed work activities to ensure that the equipment was installed as directed and consistent with the design control documents; that the equipment operated as expected; that testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the equipment did not impact the operability of any interfacing systems. In addition, the inspectors also discussed the use of the temporary equipment with operations, engineering, and training personnel to ensure that the individuals were aware of how operation with the temporary equipment 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 one temporary modification sample as defined in IP 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- operational testing of the SBODG during the week ending July 16, 2011, following an extensive equipment outage to perform the 6-year preventive maintenance activities;
- operational testing of No. 3 CCW pump during the week ending July 30, 2011, following a planned maintenance work window; and
- functional testing of the ultrasonic flow meter used to perform daily heat balance calculations for TS Surveillance 3.3.1.2 during the week ending September 10, 2011, following replacement of failed transducers.

These activities were selected based upon the structure, system, or component'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 TSs, the USAR, 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 the PMTs 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.

The review of these activities by the inspectors constituted three PMT samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

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-SC-03071; “EDG 2 Monthly Test,” during the week ending July 23, 2011 (routine);
- DB-SC-04271; “SBODG Monthly Test,” during the week ending July 23, 2011 (routine);
- DB-FP-04043; “Bus Tie Transformer Alternating Current (AC) Deluge Test,” during the week ending July 30, 2011 (routine); and
- DB-SP-03337; “Containment Spray Train 1 Quarterly Pump and Valve Test,” during the week ending July 30, 2011 (inservice testing).

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 was 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 (IST) activities, testing was performed in accordance with the applicable version of Section XI, ASME 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;
- 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.

These inspection activities constituted three routine surveillance testing samples and one IST sample as defined in IP 71111.22, Sections -02 and -05.

b. <u>Findings</u>

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

.1 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, Revision 27 of the Emergency Plan remained unchanged and Revision 13 of the emergency action level (EALs) was implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease in effectiveness of the Plan and that the revised Plan as changed continues to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The inspectors conducted a sampling review of the Emergency Plan changes and a review of the EAL changes made between December 2010 and June 2011 to evaluate for potential decreases in effectiveness of the Plan. However, this review does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety.

This emergency action level and emergency plan changes inspection constituted one sample as defined in IP 71114.04-05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2RS7 Radiological Environmental Monitoring Program (71124.07)

The following inspections constituted a single inspection sample as defined in IP 71124.07-5.

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the annual radiological environmental operating reports and the results of any licensee assessments since the last inspection to assess that the radiological environmental monitoring program was implemented in accordance with the TSs and Offsite Dose Calculation Manual (ODCM). This review included report changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, inter-laboratory comparison program, and analysis of data.

The inspectors reviewed the ODCM to identify locations of environmental monitoring stations.

The inspectors reviewed the USAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation.

The inspectors reviewed quality assurance (QA) audit results of the program to assist in choosing inspection "smart samples" and audits and technical evaluations performed on the vendor laboratory program.

The inspectors reviewed the annual effluent release report and the 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste" report, to determine whether the licensee is sampling, as appropriate, for the predominant and dose-causing radionuclides likely to be released in effluents.

b. Findings

No findings were identified.

.2 Site Inspection

a. Inspection Scope

The inspectors walked down selected air sampling stations and thermoluminescent dosimeter (TLD) monitoring stations, to determine whether they are located as described in the ODCM, and to determine the condition of the equipment material. Consistent with smart sampling, the air sampling stations were selected based on the locations with the highest X/Q and D/Q wind sectors, and TLD dosimeters were selected based on the most risk significant locations (e.g., those that have the highest potential for public dose impact).

For the air samplers and TLD dosimeters selected, the inspectors reviewed the calibration and maintenance records to assess that the licensee demonstrated adequate operability of these components. Additionally, the review included the calibration and maintenance records of select composite water samplers.

The inspectors performed an assessment of whether the licensee had initiated sampling of other appropriate media upon loss of a required sampling station.

The inspectors observed the collection and preparation of environmental samples from different environmental media (e.g., ground and surface water, milk, vegetation, sediment, and soil) as available to assess that environmental sample locations were representative of the release pathways as specified in the ODCM and that sampling techniques were in accordance with procedures.

Based on direct observation and review of records, the inspectors assessed whether the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the USAR, NRC Regulatory Guide (RG) 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," and licensee procedures. The inspectors assessed whether the meteorological data readout and recording instruments in the control room, and if applicable, at the tower were operable.

The inspectors evaluated whether missed and/or anomalous environmental samples were identified and reported in the annual environmental monitoring report. The inspectors selected events that involved a missed sample, inoperable sampler, lost TLD dosimeter, or anomalous measurement in order to assess that the licensee had identified the cause and implemented corrective actions. The inspectors reviewed the licensee's assessment of any positive sample results (i.e., licensed radioactive material detected above the lower limits of detection) and reviewed the associated radioactive effluent release data that was the source of the released material.

Inspectors selected structures, systems, or components that involve or could reasonably involve licensed material for which there is a credible mechanism for licensed material to reach ground water, and assessed whether the licensee has implemented a sampling and monitoring program sufficient to detect leakage of these structures, systems, or components to ground water.

The inspectors reviewed records required by 10 CFR 50.75(g), of leaks, spills, and remediation records since the previous inspection. Data was retained in a retrievable manner.

The inspectors reviewed any significant changes made by the licensee to the ODCM, as the result of changes to the land census, long-term meteorological conditions (3-year average), or modifications to the sampler stations since the last inspection. The inspectors also reviewed technical justifications for any changed sampling locations and assessed that the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases in the environment.

The inspectors assessed whether the appropriate detection sensitivities with respect to TSs/ODCM are used for counting samples (i.e., the samples meet the TSs/ODCM required lower limits of detection). Currently, the licensee used a vendor laboratory to analyze the radiological environmental monitoring program samples; and therefore, the inspectors reviewed these results of the vendor's quality control in order to assess the adequacy of the vendor's program.

The inspectors analyzed the adequacy of the vendor's inter-laboratory comparison program in order to assess the accuracy of the licensee's environmental sampling program. The inspectors assessed that the inter-laboratory comparison test included the media/nuclide mix appropriate for the licensee's facility. The inspectors reviewed the licensee's determination of any bias to the data and the overall effect on the radiological environmental monitoring program.

b. Findings

No findings were identified

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors assessed whether problems associated with the radiological environmental monitoring program were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. Additionally, the inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved the radiological environmental monitoring program.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Heat Removal System performance indicator (PI) for the period from July 2010 through June 2011. 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 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, MSPI derivation reports, and NRC Integrated Inspection Reports for the period to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's CAP database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI heat removal system sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Residual Heat Removal System performance indicator for the period from July 2010 through June 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable

NEI guidance. The inspectors also reviewed the licensee's CAP database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI residual heat removal system sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 MSPI - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Cooling Water Systems performance for the period from July 2010 through June 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's CAP database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI cooling water system sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the occupational radiological occurrences PI for the period from the first quarter 2010 through the second quarter of 2011. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of their data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate and accumulated dose alarms and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational exposure control effectiveness sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.5 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the radiological effluent TS/ODCM radiological effluent occurrences PI for the period from the first quarter 2010 through the second quarter of 2011. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline" Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's CAP database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates between the first quarter of 2010 and the second quarter of 2011 to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one radiological effluent TS/ODCM radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.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: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; 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 Attachment to this report.

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 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 were identified.

.3 Selected Issue Follow-up Inspection: Review of the Licensee's Backlog of Unscheduled Maintenance Work Orders Associated With Safety-Related Systems, Structures, and Components

a. Introduction

On July 29, 2011, during the course of observing an operational test of the No. 3 component cooling water (CCW) pump following a planned maintenance work window, the inspectors noted an aging maintenance work request tag (Tag No. AAA1359, dated February 8, 2003) affixed to safety-related manual valve CC4, "Cross Connect From CCW Pump No. 1." The tag indicated that the valve was difficult to operate.

Following up on this issue, the inspectors identified that the WO associated with the tag had been given a default scheduled work date of December 31, 2020, essentially making it an unscheduled WO for all practical purposes. In order to determine whether this issue represented an isolated case for a safety-related SSC or a larger programmatic issue with the licensee's CAP, the inspectors selected the licensee's unscheduled work order database for an in-depth review in accordance with IP 71152 requirements. Documents reviewed during this inspection are listed in the Attachment to this report.

The inspectors' review of this selected follow-up issue constituted one inspection sample as defined in IP 71152-05.

b. Effectiveness of Problem Identification and Resolution

(1) Inspection Scope

The inspectors reviewed CRs, WOs, notifications, and licensee self-assessments to verify that the licensee's identification and resolution of issues associated with the safety-related SSCs were complete, accurate, and timely, and that the consideration of extent of condition review, generic implications, common cause, and previous occurrences were adequate. In particular, the inspectors reviewed the CRs and notifications associated with unscheduled WOs for safety-related SSCs.

(2) Findings

Introduction

A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," were identified by the inspectors for the licensee's failure to adhere to procedural requirements regarding the correction of deficiencies, deviations, and/or nonconformances associated with safety-related SSCs.

Description

As discussed above, the inspectors conducted a review of the licensee's WO database, specifically focusing on those WOs that were unscheduled. Unscheduled WOs were arbitrarily assigned a schedule date of December 31, 2020, by the licensee's system. The inspectors reviewed the database for unscheduled outage related WOs (341 WOs) and non-outage related WOs (1622 WOs) to identify those associated with safety-related SSCs. Within this sample, the inspectors identified several WOs written against deficiencies, deviations, and/or nonconformances associated with safety-related SSCs for which the licensee had yet to schedule any corrective actions. These WOs included, but were not limited to:

- WO 200075182; Written to correct degraded fireproofing in the Auxiliary Building, which is a safety-related SSC; 12/17/2003;
- WO 200007358; Written to correct degraded wiring associated with the safety features actuation system (SFAS); 10/02/2002;
- WO 200264969; Written to correct degraded conditions associated with valve SW-273, which supports containment air cooler (CAC) No. 1; 5/23/2007;
- WO 200296926; Written to correct degraded conditions associated with electrical circuit breaker BEF153, which supports CAC No. 3; 1/15/2008; and
- WO 200309699; Written to correct degraded conditions associated with valve SW-81, which supports CAC No. 2; 1/08/2008.

Upon identification of the multiple examples noted above, the inspectors discussed the issue with the licensee. In some instances, the licensee was monitoring the deficiencies, deviations, and/or nonconformances to determine if they were stable or degrading. In some others, the licensee considered the issues to be of such low significance that no corrective actions needed to be scheduled.

The inspectors concluded that the issue constituted a programmatic deficiency associated with the licensee's CAP. Specifically, this programmatic deficiency identified by the inspectors was a trend on the part of the licensee to allow certain corrective actions for safety-related SSC deficiencies, deviations, and/or nonconformances to go unscheduled as long as the issues did not impact SSC operability or availability. In no case did the inspectors identify a WO for a safety-related SSC that was associated with the inoperability or unavailability of that SSC.

When brought to their attention by the inspectors, the licensee initiated their own review of the WO database to determine the full extent of condition of this issue. The licensee placed the original issue identified by the inspectors associated with safety-related manual valve CC4 into their CAP as CR 2011-00385.

Analysis

The inspectors determined that failure of the licensee to correct certain deficiencies, deviations, and/or nonconformances associated with safety-related SSCs in a timely manner was contrary to the requirements in the licensee's Quality Assurance Program Manual (QAPM) and CAP procedure, and as such constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The inspectors reviewed this issue using the guidance contained in Appendix B, Issue Screening, of Inspection Manual Chapter 0612, Power Reactor Inspection Reports, and determined that it was of more than minor safety significance and constituted a finding. Specifically, because the issue represented more than just a single isolated example, but instead a programmatic deficiency associated with the licensee's CAP, the inspectors determined that if left uncorrected it would have the potential to lead to a more significant safety concern.

The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 - Initial Screening and Characterization of Findings. Using the Phase 1 SDP worksheet for the Mitigating Systems Cornerstone, since each of the WOs identified by the inspectors represented a deficiency that did not result in the loss of operability or functionality, the finding screened as of very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, CAP component, because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity. Specifically, for certain deficiencies, deviations, and/or nonconformances associated with safety-related SSCs, the licensee took no corrective actions whatsoever and allowed the WOs associated with those issues to be placed in the plant's backlog of unscheduled work. The examples identified by the inspectors each were several years old and involved issues that had multiple opportunities, including several RFOs, for the licensee to have taken corrective action. (P.1(d))

Enforcement

Criterion V, “Instructions, Procedures, and Drawings,” of 10 CFR Part 50, Appendix B, states, in part, that: “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” Further, Section A.6.b of the licensee’s QAPM states, in part, that: “A corrective action program is established and implemented that includes prompt identification, documentation, significance evaluation, and correction of conditions adverse to quality.” Section A.6.d of the licensee’s QAPM states that: “Non-conforming items are properly controlled to prevent their inadvertent test, installation, or use. They are reviewed and either accepted, rejected, repaired, or reworked.” This requirement is also echoed in Section 4.10 of the licensee’s CAP implementing procedure, NOP-LP-2001, “Hardware Nonconformance Dispositions in CR Evaluations,” which describes the same four options for dispositioning a hardware nonconformance.

Contrary to these requirements, the licensee failed to take or schedule any corrective actions for certain deficiencies, deviations, and/or nonconformances associated with safety-related SSCs. Specifically, the inspectors identified multiple examples of WOs, dating back to as far as October of 2002, written to correct deficiencies, deviations, and/or nonconformances associated with safety-related SSCs that were languishing in the plant’s backlog of unscheduled WOs. These WOs represented deficiencies, deviations, and/or nonconformances that had neither been formally accepted with an appropriate engineering use-as-is evaluation, nor repaired or reworked, nor formally rejected so the SSC could be scrapped and replaced.

Because this finding was of very low safety significance and it was entered into the licensee’s CAP as CR 2011-00385, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000346/2011004-03)

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

.1 Event Notification 47096: Safety-Related Direct Current System Issues.

a. Inspection Scope

On July 26, 2011, the licensee received information from the NRC regarding a design issue identified during a component design basis inspection that had been characterized as an unresolved item (URI 05000346/2007007-05). Two issues were identified by the NRC regarding the safety-related direct current system:

- The plant's licensing basis states that non-safety-related electrical equipment, whose failure under postulated environmental conditions could prevent satisfactory accomplishment of the specified safety-related electrical equipment required safety functions, is qualified as required. However, the reactor coolant pump (RCP) backup lift oil pump motors and the containment emergency lighting panel L49E1 are located inside containment and are not environmentally qualified. This could challenge the adequacy of electrical separation between the potentially grounded non-safety related equipment and the safety related batteries; and

- Automatic transfer switches are installed to automatically transfer non-safety related loads such as non-nuclear instrumentation, station annunciators, the plant computer, and ICS between two non safety-related inverters, which receive power from the safety-related DC power system. If a ground fault existed on one of these switches, the fault could be transferred from one power source to the redundant source, potentially impacting the ability of both safety-related DC power sources to perform their required functions. This type of transfer is not permitted by the plant's licensing basis.

In response to this information, the licensee opened the circuit breakers for the four RCP backup lift oil pump motors and for the emergency lighting system in containment. One train of instrumentation power was placed on its alternate power source from the alternating current (AC) system, eliminating the potential to impact both trains of the DC power system. Additionally, upon review of the situation, the licensee reported the condition per 10 CFR 50.72(b)(3)(ii)(B) as a condition that results in the plant being in an unanalyzed condition that significantly degrades plant safety, and per 10 CFR 50.72(b)(3)(v)(A-D) as an event or condition that could have prevented fulfillment of a safety function.

The inspectors reviewed the licensee's response to the condition, including the realignment of the plant's affected dc equipment and the decision to make an 8-hour non-emergency notification. Documents reviewed in this inspection are listed in the Attachment.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153-05.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Reports 05000346/2010-004-00 and 05000346/2010-004-01: Spent Fuel Pool Rack Patterns Did Not Comply With Technical Specification 3.7.16

(Closed) Unresolved Item 05000346/2010004-02: Compliance With Spent Fuel Pool Storage Requirements

On August 26, 2010, an issue with TS 3.7.16, "Spent Fuel Storage," was identified by licensee personnel. Specifically, TS 3.7.16 required fuel assemblies in the spent fuel pool (SFP) to be placed in racks per the criteria of TS Figure 3.7.16-1, which referenced the TS Bases. The TS Bases provided amplifying information allowing for various fuel loading patterns to be used in different spent fuel racks within the SFP, provided that each rack module utilized only a single loading pattern. The TS 3.7.16 Limiting Condition for Operation (LCO) Action required initiation of actions to move non-complying assemblies to an allowable location "Immediately." At the time of discovery by the licensee, there were spent fuel rack modules within the SFP that, contrary to the TS restrictions discussed in the Bases, utilized more than one loading pattern. This condition had existed since the licensee performed a TS conversion to Improved Standard TS in 2006.

The licensee entered the issue into the CAP as CR 2010-81824. However, based on an erroneous interpretation of the TS and TS Bases by site engineering, site regulatory compliance, and corporate licensing personnel, the on-watch licensed operators misclassified the issue as "an administrative inconsistency" that required no immediate action. On the morning of August 27, 2010, following review of the previous day's CRs, the inspectors identified the error regarding TS use and application and challenged the licensee regarding their interpretation. The licensee subsequently declared the LCO not met and initiated action immediately to bring the plant into compliance. By the end of the day on August 27, 2010, the licensee had processed a change to the TS 3.7.16 Bases that corrected the issue. No spent fuel in the SFP was ever relocated.

In 2006, during the Improved Standard TS conversion, a sentence was omitted from the SFP TS Bases that stated: "Two different loading patterns may be used in a single rack module, subject to certain additional restrictions." Licensee personnel involved in the Improved Standard TS conversion believed that the sentence was redundant and non-consequential. During their review of this issue, the inspectors confirmed that the licensee's SFP technical loading analyses did, in fact, permit more than one spent fuel loading pattern to be safely used in a single spent fuel rack module. The inspectors also confirmed that the licensee's SFP implementing procedure contained the necessary restrictions to ensure that the SFP rack modules were loaded in accordance with the existing safety analyses. As a result, the inspectors determined that while the issue constituted a violation of TS 3.7.16, it was of minor safety significance and, in accordance with Section 2.3 of the NRC Enforcement Policy, not subject to formal enforcement action.

The licensee entered the issue into their CAP as CRs 10-81824 and 10-83814. Documents reviewed as part of this inspection are listed in the Attachment. These Licensee Event Reports (LERs) and the associated URI are closed.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153-05.

.3 Response to Integrated Control System Inputs Mismatch and Subsequent Plant Transient

a. Inspection Scope

On September 16, 2011, I&C personnel were conducting planned maintenance activities involving signal monitor modules associated with HPI flow instrumentation. During an I&C string check of the HPI flow instrument alarms, annunciator alarm 14-4-E, "Integrated Control System (ICS) Input Mismatch," was received. The alarm initially cleared, then returned. Coincident with ICS Input Mismatch alarm, the plant's ICS began reducing reactor power without any operator input. On-watch plant operators entered Procedure DB-OP-02526, "Primary to Secondary Plant Upset," and went through actions of placing ICS stations in manual control. The I&C technicians performing the HPI flow instrument signal monitor refurbishment were directed to stop their activities. Reactor power initially dropped to approximately 95 percent before on-watch operators stabilized the plant. The control room operators then returned reactor power to approximately 100 percent using manual controls.

Responding to the control room at the time of the transient, inspectors observed plant parameters and status; evaluated the performance of plant systems and licensee actions; and confirmed that the licensee had properly evaluated the reporting criteria for the event as required by 10 CFR 50.72. The inspectors also verified that no human performance errors complicated the event response.

The circumstances surrounding the event continue to be reviewed by the inspectors to determine if the issue was within the licensee's ability to foresee and correct and should have been prevented. The issue has been documented as an URI in Section 1R13 of this report. Documents reviewed in this inspection are listed in the Attachment.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153-05.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 Reactor Vessel Head Replacement (Inspection Procedure 71007) – Vendor Fabrication and Preservice Record Review

a. Inspection Scope

The licensee procured a replacement RVCH fabricated by vendor AREVA. The materials used in fabrication of the RVCH included a single piece forging of SA-508 steel material supplied by Japan Steel Works and penetration nozzle tubes of SB-167 Inconel material (UNS N06690) supplied by Valinox. The RVCH machining and fabrication activities occurred at the AREVA facility in Chalon/St Marcel France. Additionally, the licensee procured replacement CRDM assemblies constructed of stainless steel materials (SA-182 -type 304, A276 type 403 and SA312 type 304) originally fabricated in the 1970's by Diamond Power Specialty Company which left the fabrication business in 1983. Subsequently, the ownership of the CRDM assemblies and associated fabrication records went to the Babcock and Wilcox Company (now AREVA) before procurement by the licensee for installation on the RVCH.

From July 18, 2011, through September 9, 2011, inspectors performed a review of fabrication and preservice records related to replacement of the RVCH and CRDM housings in accordance with Section 02.03 and Step 02.05.e of Inspection Procedure 71007 "Reactor Vessel Head Replacement Inspection." This review was performed to determine if the fabrication was completed in accordance with Section III of the ASME Code and to determine if preservice nondestructive examinations (NDE) were completed in accordance with Section XI of the ASME Code. Specifically, the inspectors reviewed samples of the following types of records:

- fabrication process sheets, fabrication drawings, and NDE records to determine if the manufacturing process included provisions for NDE as required by the construction Code;
- NDE records and procedures used for preservice and fabrication examinations to determine if Code qualified examinations were completed and examination results met Code acceptance criteria;
- weld data sheets, weld procedures, and weld procedure qualification records, to determine if Code qualified weld procedures were used in fabrication of the J-groove welds, and CRDM flange-to-adaptor sleeve welds;

- Certified Material Test Reports for materials used in fabrication of the reactor vessel head and CRDM housings including weld materials to determine if appropriate chemical, mechanical tests and heat treatment records existed to meet material and Code specifications;
- non-conformance reports issued by the licensee's fabricator and subcontractors to determine if fabrication related deviations were appropriately tracked, evaluated and resolved; and
- audit records of the head fabricator and subcontractors associated with welding activities and NDE to determine if these activities had been properly controlled in accordance with the contract specifications or Code requirements.

The records reviewed by the inspectors are identified in the Attachment to this report.

b. Findings

(1) Inadequate Weld Records for Control Rod Drive Housing Mechanisms

Introduction

A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion VII "Control of Purchased Material, Equipment, and Services," were identified by the inspectors for the licensee's failure to establish adequate measures (e.g., perform a review of radiographic (RT) film weld records) to ensure material procured from a contractor (replacement control rod drive housing mechanism (CRDM) housings)) met the ASME Code. Consequently, two replacement CRDM housings were procured with RT film weld records that did not conform to the ASME Code required film density ranges.

Description

The inspectors identified that the licensee had not performed a review of RT film weld records for the replacement CRDM housings which prompted a licensee review of RT film records. As a result of this review, RT film weld records for two housings were identified that did not conform to the ASME Code required film density range. The inspectors were concerned that without appropriate film density, the image produced may not have adequate RT contrast to detect weld flaws (if present). Specifically, the density or blackness of an RT image affects the contrast of the image produced, and contrast increases with increasing density, so a minimum film density is needed to achieve adequate contrast. Similarly, there is a maximum limit where contrast is lost due to an excessively dark film image. For this reason, the 1974 Edition and later Editions of the ASME Code establish a minimum and a maximum limit for RT film density in the area of interest (e.g. a weld).

The replacement CRDM motor tube assemblies (including housings) were originally fabricated in the 1970's by Diamond Power Specialty Company for the Babcock and Wilcox Company. These assemblies were subsequently procured by the licensee through their lead fabrication vendor (AREVA) for installation at Davis-Besse (reference purchase order 45344006 issued on July 1, 2010). The replacement CRDM assemblies had been the subject to two licensee QA surveillance audit activities prior to arrival at the site. The first QA source surveillance was completed in May of 2011, at the AREVA vendor facility to ensure that required documentation would be included in the final vendor QA Data Package. A second QA source surveillance of CRDM vendor AREVA was completed in June 2011 to ensure that the QA Data Package met the purchase order before shipping. However, neither of these surveillance audits identified that the RT weld film records and reader sheets were not included in the original QA Data Package nor did these audits prompt a licensee or vendor review of these records.

To support the NRC review of fabrication records beginning on July 18, 2011, the inspectors requested that the licensee make available the RT film weld records for the replacement CRDM housings. This request prompted the licensee to obtain the RT film weld records and reader sheets from the fabrication vendor's off-site storage facility. During review of the RT weld No. 3 film record for CRDM housing serial No.1290, the inspectors measured areas (using an RT film densitometer) which fell below a Code minimum value of 2.0. The inspectors estimated for approximately 20 percent of the weld length, the RT film record fell below the minimum film density of 2.0 required by the ASME Code Section V, Article 2, Paragraph T-233 for single film viewing. The licensee subsequently determined that the RT procedure allowed the use of two stacked films (composite) viewing with a minimum individual film density of 1.3 and therefore, this weld record was in compliance with the Code. Because a review of the RT weld film records had not been required during the procurement process, the licensee completed an extent of condition review for 100 percent of the RT film weld records. As a result of this review, the licensee identified two CRDM housings with RT film weld records which did not meet the film density requirements of the 1974 Edition of Section V, Article 2, Paragraph T-233. Specifically, for weld No. 2 on CRDM housing serial No. 1941, the RT record exceeded the maximum film density limitation of 3.8. Additionally, for weld No. 1 on CRDM housing serial No. 1964, a small section of weld on the RT film record did not meet the minimum Code density of 2.0 (single viewing required). For these weld records with out-of-specification film density ranges, the RT contrast may have been inadequate to detect rejectable weld flaws (if present) such as cracks, voids or lack of fusion.

Because these replacement CRDM housings were not yet inservice, this finding did not affect current plant operation. The licensee issued CR 2011-00750 to document the non-conforming RT film records and returned the affected CRDM housings to a vendor facility. The licensee planned to have the vendor complete new RT film records for the affected welds prior to installation of these replacement CRDM housings on the replacement vessel head.

Analysis

The inspectors determined that failure to establish adequate measures (e.g. perform a review of RT film weld records) to ensure material procured from a contractor (replacement CRDM housings) met the ASME Code was contrary to 10 CFR Part 50 Appendix B, Criterion VII and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Absent NRC identification, the failure to complete an adequate RT examination of welds on two CRDM housings could have allowed unacceptable weld flaws to be placed in service. Specifically, weld flaws such as cracks, can reduce the CRDM housing integrity, and place the reactor coolant system (RCS) at an increased risk for through-wall leakage and/or failure. The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I - Initial Screening and Characterization of Findings,” Table 4a for the Initiating Events Cornerstone. Because this finding was identified prior to placing the CRDM housing inservice, the inspectors answered “No” to the Significance Determination Process Phase I screening question “Assuming worst case degradation, would the finding result in exceeding the TS limit for any RCS leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function assuming the worst case degradation?” Therefore, the finding screened as having very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee staff failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Absent NRC intervention, the failure to establish adequate measures (e.g. perform a review of RT film weld records) to ensure material procured from a contractor (replacement CRDM housings) met the ASME Code would have allowed welds on two housings with non-conforming RT records to be placed in service. The inspectors reached this conclusion based on discussions with licensee staff and review of associated records. (H.4(c))

Enforcement

Title 10 CFR Part 50, Appendix B, Criterion VII “Control of Purchased Material, Equipment, and Services,” required in part that “Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents.” And “This documentary evidence shall be retained at the nuclear power plant, or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment.”

Contrary to the above, as of August 17, 2011, the licensee had not established adequate measures (e.g., perform a review of RT film weld records) to ensure material procured from a contractor for replacement CRDM housings conformed to the procurement document (purchase order 45344006 issued on July 1, 2010), which required compliance to the ASME Code. Specifically, for CRDM housings Serial Nos. 1941 (weld No. 2) and 1964 (weld No. 1), the RT film weld records did not meet the applicable Code required by procurement documents. Because this violation was of very low safety significance and it was entered into the licensee’s CAP as CR 2011-00750, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000346/2011004-04)

(2) Code Surface Examination Requirements Not Applied To Reactor Vessel Closure Head Flange Stud Holes

The RVCH is a single piece forging fabricated to the SA 508 material standard with an ASME NPT stamp to document that this pressure boundary part was fabricated to the requirements of the 1989 Edition of the ASME Code Section III. The requirements for examination of forgings are contained in the ASME Code Section III, Article NB 2540 “Examination and Repair of Forgings and Bars.” Specifically, NB-2541(a) requires in part that, “In addition, all external surfaces and accessible internal surfaces shall be examined by a magnetic particle (MT) method (NB 2545) or a PT method (NB-2546).” Also, NB-4121.3 “Repetition of Surface Examinations After Machining” required “If, during the fabrication or installation of an item, materials for pressure containing parts are machined, then the Certificate Holder shall re-examine the surface of the material in accordance with NB-2500 when: (a) the surface was required to be examined by the MT or liquid penetrant method in accordance with NB-2500; and (b) the amount of material removed from the surface exceeds the lesser of 1/8 inch or 10 percent of the minimum required thickness of the part." For the 60, 7-inch diameter stud holes drilled through the vessel head flange, no surface examinations (e.g., MT or PT) were conducted on the interior bore surfaces of the stud holes.

The inspectors observed a licensee demonstration of the potential accessibility of the flange stud holes for MT examination. Specifically, a licensee MT qualified examiner positioned an AC yoke used for MT examinations on the interior bore surfaces of an RVCH flange stud hole. Based on this demonstration, the inspectors estimated that it would be possible to perform an MT exam for accessible portions of the interior bore surfaces for a depth of about 2 inches from the top and bottom flange faces for each of the 60 stud holes. Because this accessible interior surface on the RVCH forging had not been examined by MT or PT, the inspectors were concerned that the RVCH did not meet the requirements of NB-2541(a) and NB-4121.3 discussed above.

In response to the inspectors' questions, the licensee established a position that accessible interior surfaces of the RVCH stud holes did not require a surface examination. The licensee position was based on Code Interpretation III-1-77-162, which states in part that drilled holes are not considered to be material form surfaces and the requirement for examination of holes (if any) resides in NX-4000 and NX-5000. The licensee concluded that the reexamination of machined surfaces as discussed in NB-4121.3 did not apply to the accessible interior surfaces of the flange stud holes because they were not material form surfaces.

This issue is considered an unresolved item pending completion of an NRC staff review to determine an Agency position on the licensee's interpretation of these Code requirements. The licensee documented this issue in CR 2011-01739.
(URI 05000346/2011004-05)

.2 Reactor Vessel Head Replacement (Inspection Procedure 71007) – Onsite Head Modifications

a. Inspection Scope

For the replacement RVCH, the licensee elected to remove and replace a CRDM nozzle penetration assembly (No. 21) with a head vent assembly under the rules of the ASME Code Section XI. Use of Section XI for construction of this head vent line is allowed by Section XI Article IWA-1200 because all construction code (ASME Section III) requirements where met as evidenced by the ASME NPT stamp applied to the vessel head. The licensee elected to perform this head vent line modification to eliminate a flanged joint and to allow routing of the head vent pipe such that vent pipe dose-rates would be reduced to outage workers. The head vent line terminated at a vent nozzle reducer assembly with an interference fit (e.g., chilled and allowed to expand) inserted into the vessel location where CRDM penetration Nozzle No. 21 was removed. The inspectors observed the set-up and machining to remove the existing CRDM nozzle penetration and observed the vent line pipe elbow-to-reducer nozzle welding activities as documented in Section 1R08 of this report. The inspectors also reviewed the pre-service NDE records for the newly fabricated Nozzle No. 21 J-groove weld in accordance with Step 02.05.e of Inspection Procedure 71007 "Reactor Vessel Head Replacement Inspection."

The records reviewed by the inspectors are identified in the Attachment to this report.

b. Findings

No findings were identified.

.3 Reactor Vessel Head Replacement (Inspection Procedure 71007) – Physical Security Plan

a. Inspection Scope

In accordance with IP 71007, Section 02.02.d.1, the inspectors reviewed the licensee's planned security measures for the reactor vessel head replacement. The inspectors verified that the licensee's planned measures would ensure compliance with 10 CFR 73.55(g)(1). Inspection Procedure 71130.02, "Access Control," inspection items 02.04.a, 02.05.a, and 02.05.j were completed for the reactor vessel head replacement.

The records reviewed by the inspectors are identified in the Attachment to this report.

b. Findings

No findings were identified.

.4 Status of NRC Confirmatory Action Letter 3-10-001 Open Commitments

On June 23, 2010, the NRC issued Confirmatory Action Letter (CAL) 3-10-001 (ADAMS Accession No. ML101740519) to confirm commitments by FirstEnergy Nuclear Operating Company (FENOC) regarding the identification of CRDM nozzle cracks in and reactor pressure boundary leakage from the Reactor Pressure Vessel (RPV) head at the Davis-Besse Nuclear Power Station. The first of the four documented CAL commitments provided the NRC the results of the Reinspection Years calculation for Operating Cycle 17 performed in accordance with ASME Code Case N-729-1 based on calculated RPV Head temperatures. This commitment was completed by the licensee and closed by an NRC letter to FENOC dated July 7, 2010 (ADAMS Accession No. ML101880308). Status of the remaining commitments includes:

a. NRC CAL 3-10-001, Commitment No. 2 – Control Rod Drive Housing Mechanism Nozzle Ring Samples

Commitment No. 2 required that upon completion of destructive examination of the CRDM Nozzle ring samples removed from nozzles No. 4 and No. 10, the licensee quarantined one untested minimum full-length 90 degree sample, and turned it over to the NRC for independent testing. The sample was quarantined immediately by the licensee until arrangements could be made to transport the sample to an independent laboratory selected by the NRC.

The NRC contracted with the Department of Energy's Argonne National Laboratory (ANL) to serve as the independent laboratory. Argonne National Laboratory received a 90 degree section of the ring sample cut from nozzle No. 4 on October 12, 2010. This sample has been the subject of independent crack growth-rate testing, and a separate report is planned to document the ANL test results.

b. NRC Confirmatory Action Letter 3-10-001, Commitment No. 3 – Reactor Coolant System Integrated Leakage Program

Commitment No. 3 required that the licensee beginning with reactor startup (Mode 2) and until RPV head replacement, upon reaching Action Level 3 of EN-DP-01171, "Engineering Implementation of the RCS Integrated Leakage Program," to shutdown the plant in 30 days if RPV head leakage could not be ruled out. In addition, during the subsequent shutdown as part of the containment inspection for RCS leakage, if RPV Head leakage could not be ruled out then the licensee would conduct a bare metal visual examination of the RPV head per applicable ASME Code Case and 10 CFR 50.55a(g)(6)(ii)(D).

Beginning with the operation of the plant following the issuance of the CAL, inspectors monitored the licensee's RCS Integrated Leakage Program results. Daily calculated values for RCS leakage were reviewed and trended. In addition, the inspectors reviewed the licensee's monthly RCS Integrated Leakage Program reports and conducted periodic interviews with the RCS Integrated Leakage Program engineer. No issues or abnormal trends with RCS leakage were identified. The licensee's measured leakage rates were within expected ranges during the entire period. As a result, the licensee did not enter Action Level 3 of EN-DP-01171, "Engineering Implementation of the RCS Integrated Leakage Program," at any time.

c. NRC Confirmatory Action Letter 3-10-001, Commitment No. 4 – Reactor Shutdown for Reactor Pressure Vessel Head Replacement

Commitment No. 4 required that the licensee shut down the unit no later than October 1, 2011, and replace the RPV head. The shutdown was completed on schedule and replacement of the RPV head is ongoing.

.5 (Closed) Unresolved Item 05000346/2011012-01 Unable to Locate Fatigue Analysis for Class I Valves.

During the license renewal application review process, the reviewers identified the licensee could not locate the fatigue analysis for Class I valves. This issue was forwarded to Region III for evaluation. During this inspection period, the inspectors verified there was reasonable assurance the fatigue analysis had been previously completed; however, the documentation was not maintained appropriately. The inspectors considered this to be a minor violation of 10 CFR 50, Appendix B, Criterion XVII, "Quality Assurance Records," which required that records of activities affecting quality to be identifiable and retrievable. In the licensee's letter L-11-292, dated October 7, 2011, the licensee committed to complete a new fatigue analysis for Class 1 valves greater than 4 inches and submit an amendment to the license renewal application no later than May 31, 2012. The inspectors had no further concerns regarding this issue and therefore, this URI is closed.

.6 Review of the Plant Evaluation Report From the Institute of Nuclear Power Operations

As discussed in IMC 0612, Section 13.01, the inspectors completed a review of the report issued by the Institute of Nuclear Power Operations (INPO) for the most recent periodic plant evaluation performed at the Davis-Besse Nuclear Power Station during April 2011.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 11, 2011, the inspectors presented the inspection results to the Site Vice President, Mr. Barry Allen, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed with the licensee the scope of material reviewed that was considered to be proprietary. Proprietary information reviewed by the inspectors was controlled in accordance with appropriate NRC policies regarding sensitive unclassified information, and has been denoted as "proprietary" in the attachment.

.2 Interim Exit Meetings

Interim exits were conducted for:

- the results of the Emergency Preparedness program inspection with Mr. J. Sturdavant of the Regulatory Compliance group, via telephone on August 7, 2011;
- two PIs and radiological environmental monitoring program under the public Radiation Safety Cornerstone with Ms. P. Boissoneault, the Chemistry Manager, on August 12, 2011;
- the review of the annual licensed operator requalification exam results with the Operations Training staff's Mr. D. Hartnett via telephone on September 1, 2011; and
- the reactor vessel head replacement fabrication review (IP 71007) with the Director of Special Projects, Mr. C. Price, and other members of the licensee's staff on September 9, 2011.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violation

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- TS 5.4.1(a) requires the licensee to establish, implement, and maintain applicable written procedures for the safety-related systems and activities recommended in RG 1.33, Revision 2, Appendix A. Section 9.a, "Procedures for Performing Maintenance," of RG 1.33, Revision 2, Appendix A, further states, in part, that: "Maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." Contrary to this requirement, on September 2, 2011, licensee personnel failed to properly rig and lift a new safety-related battery charger (DBC1PN) into position. Specifically, the personnel conducting the rigging activity switched from a four-point lift configuration to a two-point lift configuration when one of the lifting bolts atop the battery charger cabinet was inadvertently sheared off. This lifting configuration change was performed with an approved lift plan that contained inadequate technical/engineering guidance. When the-component was subsequently lifted, unbalanced forces resulting from the two-point lifting configuration caused several welds on the cabinet to crack, rendering the cabinet seismically unqualified.

The objective of the Mitigating Systems Cornerstone of Reactor Safety is to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). A key attribute of this objective is human performance, and specifically, procedure use and adherence. In accordance with NRC IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” the inspectors determined that the violation was of more than minor significance in that it had a direct impact on this cornerstone objective. The licensee’s failure to use technically adequate written procedures or instructions for the rigging and lifting configuration resulted in damage to safety-related battery charger DBC1PN that rendered it seismically unqualified and added significant time to it being inoperable. The licensee had entered this issue into their CAP as CRs 2011-02288 and 2011-02290. Corrective actions planned by the licensee include either weld repairs to the cabinet to restore its seismic qualification or replacement of the entire battery charger, and a re-examination of lifting and rigging practices.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee</u>

B. Allen, Site Vice President
J. Barron, Manager, Site Projects
P. Boissoneault, Manager, Chemistry
B. Boles, Director, Site Operations
K. Byrd, Director, Site Performance Improvement
T. Chowdhary, NRC Liaison
C. Daft, Component Engineering
J. Dominy, Director, Site Maintenance
A. Garza, ALARA Specialist
D. Gerren, Manager, Steam Generator Replacement Project
D. Hartnett, Licensed Operator Requalification Lead
G. Hayes, Supervisor, Reactor Engineering
J. Hook, Manager, Design Engineering
R. Hovland, Manager, Training
V. Kaminskas, Director, Site Engineering
G. Kendrick, Manager, Site Outage Management
P. McCloskey, Manager, Site Regulatory Compliance
D. Munson, NDE Specialist
D. Noble, Manager, Radiation Protection
M. Parker, Manager, Site Protection
R. Patrick, Manager, Site Work Management
A. Percival, Sr. Nuclear Technologist
D. Petro, Manager, Steam Generator Replacement Project
S. Plymale, Manager, Site Operations
C. Price, Director, Special Projects
D. Saltz, Manager, Site Maintenance
S. Steagall, Fleet Oversight Manager
C. Steenbergen, Superintendent, Operations Training
J. Sturdavant, Regulatory Compliance
T. Summers, Manager, Plant Engineering
L. Thomas, Manager, Nuclear Supply Chain
S. Trickett, Superintendent, Radiation Protection
J. Vetter, Manager, Emergency Response
A. Wise, Manager, Technical Services
G. Wolf, Supervisor, Regulatory Compliance

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000346/2011004-01	URI	Plant Transient During HPI Flow Instrument String Checks (Section 1R13.1)
05000346/2011004-02	NCV	Failure to Control ECCS Room Cooler Valve Position (Section 1R15.1)
05000346/2011004-03	NCV	Failure to Take Timely Corrective Actions (Section 4OA2.3)
05000346/2011004-04	NCV	Inadequate Weld Records for CRDM Housings (Section 4OA5.1)
05000346/2011004-05	URI	Code Surface Examination Requirements Not Applied to Closure Head Stud Holes (Section 4OA5.1)

Closed

05000346/2011004-02	NCV	Failure to Control ECCS Room Cooler Valve Position (Section 1R15.1)
05000346/2011004-03	NCV	Failure to Take Timely Corrective Actions (Section 4OA2.3)
05000346/2010-004-00	LER	Spent Fuel Pool Rack Patterns Did Not Comply With Technical Specification 3.7.16. (Section 4OA3.2)
05000346/2010-004-01	LER	Spent Fuel Pool Rack Patterns Did Not Comply With Technical Specification 3.7.16. (Section 4OA3.2)
05000346/2010004-02	URI	Compliance With Spent Fuel Pool Storage Requirements. (Section 4OA3.2)
05000346/2011004-04	NCV	Inadequate Weld Records for CRDM Housings (Section 4OA5.1)
05000346/2011012-01	URI	Unable to Locate Fatigue Analysis for Class I Valves (Section 4OA5.5)

Discussed

05000346/2007007-05	URI	Concern Regarding Safety-Related Battery Electrical Isolation (Section 4OA3.1)
05000346/-00	CAL	CAL 3-10-001, Commitment No. 2 – CRDM Nozzle Ring Samples (Section 4OA5.4)
05000346/-00	CAL	CAL 3-10-001, Commitment No. 3 – RCS Integrated Leakage Program (Section 4OA5.4)
05000346/-00	CAL	CAL 3-10-001, Commitment No. 4 – Reactor Shutdown for RPV Head Replacement (Section 4OA5.4)

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.

1R04 Equipment Alignment

Procedures:
- DB-OP-06011; High Pressure Injection System; Revision 27
- DB-OP-06012; Decay Heat and Low Pressure Injection System; Revision 52
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 48
- DB-OP-06334, Attachment 1; SBODG Valve Check List; Revision 18
- DB-OP-06334, Attachment 2; SBODG Switch and Breaker Check List; Revision 18

Drawings:
- OS-003; Operational Schematic: High Pressure Injection System; Revision 34
- OS-004, Sheet 1; Operational Schematic: Decay Heat Removal / Low Pressure Injection System; Revision 48
- OS-041A, Sheet 1; Operational Schematic: Emergency Diesel Generator Systems; Revision 30
- OS-041B; Operational Schematic: Emergency Diesel Generator Air Start / Engine Air System; Revision 38

1R05 Fire Protection

Procedure:
- DB-MI-04817; Supervisory and Functional Test of Accessible Detectors for Node 7 C4720; Revisions 5 and 6

Pre-Fire Plans:
- PFP-AB-319; Diesel Generator 1-2 Room, Rooms 319 and 319A, Fire Area J; Revision 7
- PFP-AB-328; Component Cooling Water Heat Exchanger and Pump Room, Room 328, Fire Area T; Revision 4
- PFP-AB-402; No. 1 Electrical Penetration Room, Room 402, Fire Area DG; Revision 5
- PFP-AB-427; No. 2 Electrical Penetration Room, Room 427, Fire Area DF; Revision 4

Drawing:
- A-224F; Fire Protection, General Floor Plan El 603’-0”; Revision 23

Work Orders:
- 200322167; Clean/Inspect Various Drains; 12/22/2010
- 200346584; Performance of DB-MI-04817; 04/30/2011

Other:
- Fire Hazard Analysis Report; Revision 24

1R11 Licensed Operator Requalification Program

Business Practices:
- DBBP-TRAN-0014; License Requirements for Licensed Operators; Revision 9
- DBBP-TRAN-0502; Development of Continuing Training Simulator Evaluation; Revision 7
- NOBP-TR-1112; FENOC Conduct of Simulator Training and Evaluation; Revision 0

Procedures:
- NT-OT-7001; Training and Qualification of Operations Personnel; Revision 12

1R12 Maintenance Effectiveness

Condition Reports:
- 2009-61627; CREATCS/CREVS Failed MR (A)(1) Action Plan Goal
- 2010-72958; CREVS Standby Condenser 2 Damper Control Auto LED Not On
- 2010-78807; CREVS 1 Low Refrigerant Pressure
- 2010-78954; CREVS Train 1 Compressor Tripped
- 2010-86055; CREATCS Train 1 Compressor Trips
- 2010-87485; WW1051 – CREATCS Train 1 Unplanned Unavailability
- 2011-87783; Refrigerant Leaks Indicated On CREATCS Train 2
- 2011-88222; CREATCS Failed MR (A)(1) Goal
- 2011-89041; Issue Identified During (A)(1) Action Plan Review
- 2011-89233; Water Leaking From Above CTRM Restroom
- 2011-94036; Freon Leak On CREATCS Train #2
- 2011-98237; Sump Pump P111B in Service Water Valve Room Coupling Failed
- 2011-98328; Turbine Building and Water Treatment Sumps Identified as Operational Challenge

Procedures:
- DB-OP-06505; Control Room Emergency Ventilation System Procedure; Revision 13
- DB-OP-06511; Control Room Heating, Ventilation and Air Conditioning System Procedure; Revision 12
- NOP-ER-3004; FENOC Maintenance Rule Program; Revision 01

Work Orders:
- 200342580; PM 5419 – BF1186 Test Circuit Breaker
- 200358602; HV5311A – Damaged Damper Seal
- 200400757; PM 10397 – Internal Inspect/Replace HA19 & 21
- 200400774; PM 10398 – Internal Inspect/Replace HA18 & 20
- 200442169; HV5311A Rebuild Actuator

Calculations:
- C-ME-028.01-011; CREVS Capacity Test; Revision 2

Other:
- Davis-Besse Severe Accident Management Guidelines; Revision 00
- MRPM; Maintenance Rule Program Manual; Revision 29
- Maintenance Rule Unavailability Hours Database
- Unit Operating Logs; July 2009 through July 2011
- USAR Section 9.4; Air Conditioning, Heating, Cooling, And Ventilating Systems

1R13 Maintenance Risk Assessments and Emergent Work Control

Condition Reports:
- 2011-00425; NRC: Tools Left in Protected Train Room
- 2011-00889; MFP 2 Control Issue Causes Feedwater Upset and Requires Operators to Enter Abnormal Operating Procedure
- 2011-00925; HICICS36A Indications Erratic When Zeroing Transfer Volts
- 2011-01652; Routine Infrared Inspection Of Control Rod Drive Transfer Fuses Noted Group 4 Rods Have Only One Energized Phase, A Phase Fuses Are at Ambient Temperature and the Indicator Lights Are Out
- 2011-01654; Group 4 Rods Transferred to the Aux Power Supply Unexpectedly
- 2011-01765; Unexpected Light Indications on the Rod Control Panel While Transferring Control Rods
- 2011-02037; ICS Mismatch During I&C Testing
- 2011-02452; Untimely Initiation of a CR for an ICS Mismatch Alarm (14-4-E)

Procedures:
- DB-OP-02526; Primary to Secondary Heat Transfer Upset; Revision 3
- DB-OP-06402; CRD Operating Procedure; Revision 20
- NOP-OP-1007; Risk Management; Revision 9

Business Practices:
- DBBP-OPS-0003; On-Line Risk Management Process; Revision 10
- DBBP-OPS-0011; Protected Equipment Posting; Revision 3

Work Orders:
- 200473760; Clean/Inspect HICICS36A, ICS Hand/Auto Station for MFPT 2
- 200474961; Troubleshoot CRD Anomalies as per PSDM
- 200475826; Troubleshoot HPI 3B Flow Instrument

Calculations:
- C-NSA-099.16-023; Risk Significant Component Matrix – Attachment 7; Revision 7

Other:
- MRPM; Maintenance Rule Program Manual; Revision 29

1R15 Operability Evaluations

Condition Reports:
- 2011-97823; Snubber DB-SNA87 Oil Leak
- 2011-97975; Issues with EDG Room Max Temperature
- 2011-97198; ECP 10-0192-001 Undesirable Test Results
- 2011-96718; SW103 and SW87 Misposition For ECCS Room Coolers 1 and 2

Procedures:
- NOP-LP-2001; Corrective Action Program; Revision 27
- DB-OP-06316, Attachment 13; Conditions Affecting EDG Operability; Revision 48
- NOP-OP-1009; Operability Determinations and Functionality Assessments; Revision 3
- NOP-OP-1014; Plant Status Control; Revision 1
- NOBP-OP-0004; Plant Status Control and Clearance Events; Revision 4
- DB-OP-06261; Service Water System Operating Procedure; Revision 45

- DB-SP-03026; Service Water Valve Verification Monthly Test Train 2; Revision 13

Calculations:
- C-EE-024.01-010; Emergency Diesel Generator Room Electrical Equipment Temperature Evaluation; Revision 00, Addendum A09
- C-ME-024.02-001; HVAC Diesel Generator Room; Revision 00, Addendum A05
- C-NSA-011.01-019; Analysis of Service Water System Online Flow Balance Test Data for Train 2; Revision 1, Addendum 1

Work Orders:
- 200240643; PM 4875, ECCS Piping Clean and Inspect

Other:
- ECP 10-0192-001; Correct Wiring Problems with Service Water Strainer F15-1; Revision 3

1R18 Plant Modifications

Condition Reports:
- 2011-97844; Station Air Compressor 1 Auto Start
- 2011-97844; Temporary Diesel Air Compressor Unloaded
- 2011-00467; Temporary Diesel Air Compressor Trip

Procedures:
- NOP-OP-02528; Loss of Instrument Air; Revision 14
- NOP-OP-06251; Station and Instrument Air System Operating Procedure; Revision 26

1R19 Post Maintenance Testing

Condition Reports:
- 2011-97751; Redundant Voltage Regulator SW#2 On SBODG is Binding
- 2011-97672; SBODG New Cylinder 20 Fuel Injector Rack Stuck
- 2011-97709; Bad Termination Lug Found on SBODG Immersion Heater
- 2011-97585; Damaged Rocker Arm Shaft Supports
- 2011-97671; SBODG Fuel Line Clamp Wearing Into Fuel Line
- 2011-01587; LEFM Failed Causing Group 38 to Become Non-Functional
- 2011-00367; Degrading Trend of LEFM Meter 2, Path 7 Transducer
- 2011-01647; LEFM Meter 2, Paths 7 & 8 Equipment Failure

Procedures:
- DB-SC-04271; SBODG Monthly Test; Revision 19
- DB-OP-06334; Station Blackout Diesel Generator Operating Procedure; Revision 18
- DB-MM-09345; Emergency and Station Blackout Diesel Generator Engine 6 Year Maintenance; Revision 1
- DB-MM-09320; Emergency and Station Blackout Diesel Engine Maintenance; Revision 20
- DB-PF-03074; Component Cooling Water Pump 3 Test; Revision 15
- DB-PF-05064; Electrical Machine Testing Using PdMA Motor Tester; Revision 9
- DB-OP-06407; Non Nuclear Instrumentation System Operating Procedure; Revision 12

Work Orders:
- 200427361; E211 – Inspect Internal 5% Radiator Tubes
- 200390839; PM 10011 DBSCBOP Replace Circuit Boards

- 200377668; SBODG Battery Inspection and Discharge Test
- 200392349; Declining SBODG Crankcase Vacuum Trend
- 200435394; PM 4811 S438 Replace Air Start Train #2
- 200387242; CC Pump 3 Motor Testing
- 200473199; Contingent – LEFM Meter 2, Path 7 Replacement

Drawings:
- M-036A; Piping and Instrument Diagram – Component Cooling Water System; Revision 29
- E-700, Sheet 2; LEFM Connection Wiring Diagram Loop B; Revision 1

1R22 Surveillance Testing

Condition Reports:
- 2011-97969; SBODG Jacket Water Temperature High Pre-Alarm Received During Testing
- 2011-97983; SBODG High Water Temperature Alarmed During Loaded Run Using DB-OP-06334

Procedures:
- DB-OP-02250; SBODG Alarm Panel 250 Annunciators; Revision 4
- DB-OP-06334; Station Blackout Diesel Generator Operating Procedure; Revision 18
- DB-SC-03071; Emergency Diesel Generator 2 Monthly Test; Revision 23
- DB-SC-04271; SBODG Monthly Test; Revision 19
- DB-FP-04042; Bus Tie Transformer AC Deluge Test; Revision 8
- DB-SP-03337; Containment Spray Train 1 Quarterly Pump and Valve Test; Revision 21

Calculations:
- C-ME-099.16-010; Check Valve Design Basis Analysis; Revision 1

Other:
- ISTP3; Third Ten Year Inservice Testing Program; Revision 11
- ASME Operation & Maintenance Code, 1995 Edition, 1996 Addenda
- ISTB1; Pump and Valve Basis Document, Volume I – Valve Basis; Revision 10
- ISTB2; Pump and Valve Basis Document, Volume II – Pump Basis; Revision 12
- ISTB3; Pump and Valve Basis Document, Volume III – Stroke Time Basis; Revision 41
- ECP 04-0271; Containment Spray System Piping Thermal Pressure Relief Flow Path; Revision 0

1EP4 Emergency Action Level and Emergency Plan Changes

10 CFR 50.54(q) Evaluation Packages:
- RA-EP-01500; Emergency Classification; Revision 13
- RA-EP-01600; Unusual Event; Revision 5
- RA-EP-01900; General Emergency; Revision 6
- RA-EP-02010; Emergency Management; Revision 11
- RA-EP-02010; Emergency Management; Revision 12
- RA-EP-02110; Emergency Notification; Revision 10
- RA-EP-02220; Emergency Operations Facility Activation and Response; Revision 8
- RA-EP-02240; Offsite Dose Assessment; Revision 5
- RA-EP-02252; DBAB Radiation Monitoring Team Surveys; Revision 2
- RA-EP-02310; Technical Support Center Activation and Response; Revision 8
- RA-EP-02410; Operations Support Center Activation and Response; Revision 16

- RA-EP-02510; Emergency Security Organization Activation and Response; Revision 10
- RA-EP-02510; Emergency Security Activation and Response; Revision 11
- RA-EP-02530; Evacuation; Revision 4

2RS7 Radiological Environmental Monitoring Program

Condition Reports:
- 2008-44305; MS-C-08-08-02; REMP Sample Audit Result Documentation
- 2008-45481; REMP Air Sample Lost in the Mail
- 2010-72255; Leaking Temporary Underground Line for the Condenser Pit Draining to the South Settling Basin
- 2010-80339; Four Groundwater Wells Sampled in July were Above 2000 pCi/L Tritium
- 2010-80372; Unavailable Environmental Sample for REMP Program
- 2010-80693; REMP Purchase Order Expired Prior to New Purchase Order Being Finalized
- 2010-83341; Four September Groundwater Wells Test Above 200 pCi/L Tritium
- 2010-83622; Containment Air Purge Exhaust Filter Fails In-place Leakage Test
- 2010-86100; Safety Issues While Sampling for REMP Samples
- 2011-00517; Evaluation of Current X/Q and D/Q Values Used in Dose Calculation Manual
- 2011-00532; NRC REMP Inspection Have Resulted in Three Action Items
- 2011-91561; Potential Source of Past Groundwater Tritium
- 2011-91925; Turbine Building Sump Outage Discharge Line Leak to the Settling Pond
- 2011-92208; I-131 Detected in Davis Besse REMP Air Samplers
- 2011-92476; I-131 Detected in Davis Besse REMP Air Sampler for Second Week in a Row
- 2011-92679; Documentation of Iodine-131 in a REMP Samples
- 2011-96420; NRC Requested for Information of Missing from 2010 Annual Report Package
- 2011-97362; Water from East Condenser Pit Sump Pumped to Gravel East of Circ Water Pump House

Procedures:
- DB-CN-00015; REMP Program Administered by Manager Site Chemistry; Revision 2
- DB-CN-03004; Radiological Monitoring Quarterly, Semi-annually and Annual Sampling; Revision 6
- DB-CN-03005; Radiological Monitoring Weekly, Semi-monthly and Monthly Sampling; Revision 3
- CDB-CN-03023; Annual Land Use Census; Revision 1
- DB-CN-10101; REMP Enhancement Sampling; Revision 4
- DB-CN-03023; Surveillance Test Procedure; Annual Land Use Census; Revision 1
- DB-ST20079; Functional Check of 100 Meter Anemometer; Revision 1
- EN-DP-0400; Meteorological Monitoring System Channel Calibration; Revision 1
- NOBP-OP-2012; Nuclear Business Practice; System/Work Practice Prioritization for NEI 0707; Revision 0
- NOP-ER-2007; Underground Piping and Tanks Integrity Program; Revision 1

Other:
- 2010 Annual Radiological Environmental Operating Report
- 2010 Radiological Effluent Release Report
- ATI Environmental Inc., Midwest Laboratory; Reporting Period from January through December 2010
- DB-CN3023-001; Annual Land Use Census; August 10, 2009
- DB-CN3023-001; Annual Land Use Census; August 12, 2011
- EPRI Priority Index Worksheet for Potential Failure Mode of Buried Piping

- Offsite Dose Calculation Manual; Revision 25
- SD-032C; System Description for Meteorological Monitoring System; Revision 2
- DB-MI-04050; Instrument and Control Procedure; Channel Calibration of Temperature for Meteorological Tower (Primary and Backup); April 14, 2010
- DB-ST-1010; Functional Check Wind Speed and Wind Direction Sensors of 10 Meter Anemometer Backup System; 2008, 2009, and 2010

4OA1 Performance Indicator Verification

Forms:
- NOBP-LP-4012-48; MSPI Heat Removal System (AFW); Completed Forms for July 2010 through June 2011
- NOBP-LP-4012-49; MSPI Residual Heat Removal System (LPI); Completed Forms for July 2010 through June 2011
- NOBP-LP-4012-50; MSPI Support Cooling System, Component Cooling Water; Completed Forms for July 2010 through June 2011
- NOBP-LP-4012-51; MSPI Support Cooling System, Service Water; Completed Forms for July 2010 through June 2011

Procedures:
- NOBP-LP-4012; NRC Performance Indicators; Revision 3

Other:
- NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 6
- Select Operator Logs covering the period of July 2010 through June 2011
- Maintenance Rule Unavailability Database covering the period of July 2010 through June 2011
- Occupational Exposure Control Effectiveness; PI Summary of Davis Besse; between January 2010 and June 2011
- RETS/ODCM Radiological Effluent Occurrences; PI Summary of Davis Besse; between January 2010 and June 2011

4OA2 Identification and Resolution of Problems

Condition Reports:
- 2011-00385; NRC: Timeliness of Corrective Actions for a Nuclear Safety Related Component
- 2011-02200; Unscheduled Low Level Deficiencies

Work Orders:
- 200007358; 02-006019-000 SFAS 2 Rework Wire Dressings; 10/02/2002
- 200075182; Replace Missing Fireproofing; 12/17/2003
- 200123961; Rework E22-1 End Bells; 11/30/2004
- 200123962; Rework E22-2 Degraded End Bells; 11/30/2004
- 200123963; Rework E22-3 End Bells; 11/30/2004
- 200186853; MU6423A Replace Valve Due to Leak By; 11/12/2005

Procedures:
- NOP-LP-2001; Corrective Action Program; Revision 29
- NOBP-LP-2010; FENOC Trend Coding; Revision 10

Other:
- FENOC Quality Assurance Program Manual; Revision 15

4OA3 Followup of Events and Notices of Enforcement Discretion

Condition Reports:
- 2010-81824; Inconsistency Within Technical Specification Bases 3.7.16
- 2010-83814; Past Compliance With TS 3.7.16 – SFP Patterns
- 2011-00501; Operation of YAU/YBU and Other Loads Under the NRC Response to Safety-Related Batteries Electrical Separation Design and Licensing Bases
- 2011-00702; Operation of YAU/YBU and Other Loads Under the NRC Response to Safety-Related Batteries Electrical Separation Design and Licensing Bases
- 2011-01902; Extent of condition concerns from CR 2011-98223
- 2011-02037; ICS Mismatch During I&C Testing
- 2011-02452; Untimely Initiation of a CR for an ICS Mismatch Alarm (14-4-E)
- 2011-02622; NRC Discussion Relating to POD 2011-04 (CR 2011-1902)

Procedures:
- DB-NE-00100; Fuel Handling Administration; Revision 12

Engineering/Technical Analyses:
- HI-2002359; Holtec International Report – Criticality Analysis for Storage Racks in the Spent Fuel Pool of the Davis-Besse Nuclear Power Station; 04/03/2000 [PROPRIETARY]
- HI-992329; Holtec International Report – Design and Licensing Report, Davis-Besse Spent Fuel Pool Rerack Project; Undated [PROPRIETARY]

Drawings:
- OS-60, Sheet 1; 250/125V DC and 125V Instrument AC System; Revision 16
- OS-60, Sheet 2; 250/125V DC and 125V Instrument AC System; Revision 14

4OA5 Other Activities

Audit and Surveillance Records:
- Framatome ANP -Surveillance 6020527; SMAW of CRDM Housing J-groove Welds; dated May 9-15, 2003
- Framatome ANP -Surveillance 6020527; SMAW of CRDM Housing J-groove Welds; dated April 11, 2003
- Framatome ANP -Surveillance 6020519; UT of Mating Flange; dated January 21, 2003
- Framatome ANP -Surveillance 6020521; RT of CRDM housing welds; dated March 31, 2003
- FENOC- Surveillance 7084643; NDE and Mechanical Tests of JSW Head Forging; dated August 21 – 27, 2002
- AREVA- Surveillance 6030478; Hydro Test; dated March 1, 2003
- AREVA- Surveillance 6030478; In-Process Welding-Cladding; dated January 14, 2003
- FENOC-Source Surveillance DB120112613-CRDM Stators and Position Indicators; dated May 6, 2011
- FENOC-CRDM QADP Review and Shipping Preparation; dated June 27, 2011

Certified Material Test Reports:
- JQA-02-173; Closure Head Forging-Japan Steel Works, LTD; dated August 27, 2002
- MET-02-083; Closure Head Forging Archive Material; Nikko Inspection Service Co.; dated August 21, 2002
- CC028063; Control Rod Drive Housing Flanges-FOMAS; dated July 17, 2002
- 02/0152; Adaptor Sleeves-Valinox; dated October 9, 2002
- 10-005; SA 312 Pipe Material- Babcock and Wilcox; dated October 7, 1974

- 8002520; SA 276 Bar Material- Babcock and Wilcox; dated July 6, 1976
- CO02888; UNS N06690 Round Bar-Aubert & Duval; January 14, 2002
- 64172-00; SA-182F-304 Motor Tube Extension Cap-McInnes Steel Company, dated June 26, 1976
- 1135/2002; F304L LCO- FORONI (Flanges); dated May 31, 2002
- Y3189T308L; Arcos Corporation (Weld Wire); dated January 9, 1976
- Y3353T309L; Arcos Corporation (Weld Wire); dated June 29, 1976
  D2542T308L; Arcos Corporation (Weld Insert); dated June 27, 1975

Code Forms:
- N-2; Reactor Vessel Replacement Closure Head; dated January 6, 2005
- N-2; Control Rod Drive Housing 1615; dated February 1, 1978

Condition Reports:
- AREVA 2010-5721-CR; Scratch on ID of CRDM 62
- AREVA 2010-6495-CR; Interference Fit UT Anomaly on CRDMs 3 and 16
- AREVA 2010-7034-CR; Interference Fit UT Anomaly on CRDMs 3 and 16
- 2011-00343; RRVCH CRDM Flange Hardness not Performed
- 2011-00344; No Surface Exam of Stud Holes
- 2011-00665; Rounded RT Indication Weld 1 Continuous Vent Line
- 2011-00736; AREVA Welding Activities Working Outside of Procedure Parameters
- 2011-01742; RVCH Nozzle UT Data
- 2011-01739; Surface Examination of Accessible Internal Surfaces RVCH Stud Holes
- 2011-02288; Battery Charger DBC1PN Broken Lifting Lug
- 2011-02290; Cracking Found on Cabinet Welds for New Battery Charger DBC1PN Installed In Plant (Not Connected or Energized)

Contract Variation Approval Requests (CVARs)
- CVAR 87-5037876-00; CRDM holes Nos. 16 and 32 machined oversized; dated December 5, 2003

Drawings:
- AREVA -02-5053158E-00; Replacement Reactor Vessel Closure Head; Revision 6
- AREVA- BUMPDB/NCC4100; Replacement Reactor Vessel Closure Head as-Built Dimensions; Revision B
- JSW - N148620; Closure Head Forging; Revision 2
- Diamond Power -704374-1052; Lower Motor Tube Assembly; Revision A
- Diamond Power -704285-1142; Motor Tube Base; Revision B
- FENOC-ISI-SK-005; Control Rod Drive Housing Welds and Bolting Details; Revision 2
- Fabrication Nondestructive Examination Reports and Procedures
- AREVA- Report Nozzle 21-NDE-620-00- PT-200 Data Sheet-Liquid Penetrant Examination- J-Groove Weld Root Pass; dated August 29, 2011
- AREVA- Report Nozzle 21-NDE-C-90-00- PT-200 Data Sheet-Liquid Penetrant Examination- J-Groove Weld Root Pass-Repair; dated August 30, 2011
- AREVA- Report Nozzle 21-NDE-650-00- PT-200 Data Sheet-Liquid Penetrant Examination- J-Groove Weld ¼ pass; dated August 30, 2011
- AREVA- Report Nozzle 21-NDE-680-00- PT-200 Data Sheet-Liquid Penetrant Examination- J-Groove Weld 1/2 pass; dated August 30, 2011
- AREVA- Report Nozzle 21-NDE-710-00- PT-200 Data Sheet-Liquid Penetrant Examination- J-Groove Weld 3/4 pass; dated August 30, 2011

- JSW -1054-1-16-2; Record of Magnetic Particle Examination -Closure Head Forging; dated June 27, 2002
- JSW 1054-1-18-2; Record of Magnetic Particle Examination -Closure Head Forging; dated August 23, 2002
- JSW -1054-1-16-1; Record of Ultrasonic Examination -Closure Head Forging; dated August 26, 2002
- JSW 1054-1-18-1; Record of Ultrasonic Examination -Closure Head Forging; dated June 29, 2002
- Framatome ANP- cc/DB001-2800-0240; Closure Head Hydrotest Report; dated March 2, 2004
- FOMAS -UT021313; Ultrasonic Examination Certificate; dated June 14, 2002
- Diamond Power Specialty Corporation- Hydrostatic Test of Motor Tube Assembly 1615; dated December 30, 1977
- McInnes Steel Company- Ultrasonic Inspection Report 64172-OB; dated July 7, 1976
- McInnes Steel Company- Dye Penetrant Inspection Report 64172-OB; dated July 7, 1976
- Diamond Power- Nondestructive Examination Certification of Pressure Boundary Components- CRDMs 1601-1640; dated March 17, 1978
- Diamond Power- Nondestructive Examination Certification (Dye Penetrant)- CRDMs part Number 034; dated February 9, 1978
- Diamond Power- Nondestructive Examination Certification (Ultrasonic)- CRDM part Number 034; dated February 9, 1978
- Diamond Power- Nondestructive Examination Certification (Magnetic Particle)- CRDM part Numbers 10-39, 255,256,257; dated February 9, 1978
- Diamond Power- Radiographic Inspection Report - CRDM 1290 Weld No 3; dated May 17, 1976
- Diamond Power- Radiographic Inspection Report - CRDM 1290 Weld No 2; dated February 12, 1976
- Diamond Power- Radiographic Inspection Report - CRDM 1290 Weld No 1; dated April 30, 1976
- Diamond Power- Radiographic Inspection Report - CRDM 1619 weld No 3; dated January 10, 1978
- Diamond Power- Radiographic Inspection Report - CRDM 1619 weld No 2; dated November 22, 1977
- Diamond Power- Radiographic Inspection Report - CRDM 1619 weld No 1; dated November 30, 1977
- Diamond Power- Radiographic Inspection Report - CRDM 1964 weld No 1; dated June 29, 1979
- Diamond Power- Radiographic Inspection Report - CRDM 1941 weld No 2; dated April 10, 1979
- Framatome ANP- Dimensional Checking Report-60 Stud Holes; dated December 16, 2003
- Framatome ANP- Dimensional Checking Report-69 CRDM Holes; dated November 21, 2003
- Framatome ANP- Liquid Penetrant Examination Report-PT CRD Adaptors after Grinding; dated December 20, 2003
- Framatome ANP; Liquid Penetrant Examination Report-PT White of Final CRDM J-groove Welds; dated February 13, 2004
- Framatome ANP- Liquid Penetrant Examination Report-PT of Lifting Lug Welds; dated December 18, 2003
- Framatome ANP- Liquid Penetrant Examination Report-PT of CRDM Housing Holes and Groove Buttering; dated December 18, 2003.
- Framatome ANP- Magnetic Particle Examination Report-MT of the base metal after Hydrotest; dated March 10, 2004

- Framatome ANP; Radiographic Examination of CRDHC; dated April 22, 2003
- Framatome ANP- Radiographic Examination of CRDHC; dated December 10, 2002
- Framatome ANP- Procedure CORSDB/NCC0102; Radiographic Examination of the Flange to Sleeve Weld of CRDM; dated December 19, 2002
- JSW- N-7409-30; Ultrasonic Examination Procedure for Closure Head Forging; Revision B

Miscellaneous Documents:
- AREVA - 08-5015881-008; Certified Design Specification-Reactor Vessel Closure Head Replacement Davis Besse-1; dated June 30, 2010
- Code Case 2142-2; F-Number Grouping for Ni-Cr-Fe Filler Metals Section IX (Applicable to all Sections, Including Section III, Division 1, and Section XI); August 7, 2003
- F.500459; CRDM Nozzle Removable/CV Nozzle Installation Traveler; dated July 15, 2011
- JSW- Certificate of Compliance 1054-1-20-1; Closure Head Forging and Archive Material; dated August 27, 2002
- AREVA -Quality Assurance Plan- Final Machining; Revision A
- AREVA -Quality Assurance Plan-Control Rod Drive Housing Flanges; Revision A
- Valinox -Quality Assurance Plan- Adaptor Sleeves; Revision 01
- FOMAS -Heat Treatment Certificate- CRDM Housing Flanges; dated June 18, 2002
- DMV - Stainless Mechanical Test 216964; dated April 9, 2002
- DMV -Stainless Mechanical Test 216976; dated April 9, 2002
- Framatome ANP Heat Treatment Report CC/DB001-2400-0120; dated October 25, 2003
- Diamond Power Specialty Corporation -Heat Treat Summary CRDM 1614; dated January 12, 1978
- Framatome ANP -Procurement Specification BUHSDB/NCC001; dated January 22, 2002
- JSW- N-7409-10; Technical Manufacturing Program for Closure Head Forging; Revision B
- AREVA- 51-5043387-002; DB-1 Replacement RV Closure Head Reconciliation; dated July 26, 2011
- AREVA -51-9166078-002; Davis Besse CRDM Replacement Reconciliation; dated August 5, 2011
- AREVA -51-9164798-001; Material Reconciliation for Control Rod Drive Mechanism Motor Tube for Davis Besse Unit 1; dated July 22, 2011
- AREVA Certificate of Personnel Qualification; ID No J4369; dated August 18, 2010
- AREVA -Certificate of Personnel Qualification; ID No S7081; dated July 22, 2010

Procedures:
- DBBP-LP-1204; Mode Checklist and Plant Conditions; Revision 04
- NOP-LP-1203; Security Badge Control; Revision 2

Nonconformance Reports (NCRs):
- Framatome ANP -NCR-003.072; Skirt Machining Radius; dated January 12, 2003
- Framatome ANP -NCR-36/02/7; 3 Tool Marks on ID of Flange BR/001; dated July 18, 2002
- Framatome ANP -NCR-03/00203; Removal/Repair Cladding Indications; dated March 25, 2003
- Framatome ANP -NCR-03/00205; Aligned Rounded Indications of 7mm length CRDHC 38.44; dated April 23, 2003
- Framatome ANP -NCR-03/00206; Repair of Nonconforming J-groove Weld Profiles; dated April 30, 2003
- Framatome ANP -NCR-03/00215; Nonconforming CRDM holes Nos. 16 and 32; dated November 26, 2003
- Valinox -NCR-02/006; Boring and Hot Extrusion Defects; dated July 17, 2002
- Valinox -NCR-02/008; ID Diameter out of Specification; dated September 24, 2002

- Preservice Examinations and Procedures
- AREVA- 51-9143642-00; Davis Besse Unit 1, Replacement Reactor Vessel Base Line NDE Inspection Report; dated December 7, 2010
- AREVA -Report DB-PT-2; PT-240 Data Sheet-Liquid Penetrant Examination (CRDM No 2 Nozzle to Flange Dissimilar Metal Weld); dated August 24, 2010
- AREVA -RPV Head Penetration UT Data Sheet-Dual Blade (CRDM No 2); dated August 21, 2010
- AREVA -Procedure 54-ISI-603-05; Automated Ultrasonic Examination of RPV Closure Head Penetrations Containing Thermal Sleeves; dated July 12, 2010
- AREVA -Procedure 54-ISI-460-03; Multi-Frequency Eddy Current Examination of Nozzle Welds and Regions; dated March 17, 2010
- AREVA -Procedure 54-ISI-491-07; Multi-Frequency Rotating Eddy Current Examination Reactor Vessel Head Penetrations; dated October 2, 2006
- AREVA -Procedure 54-ISI-240-44; Visible Solvent Removable Liquid Penetrant Examination Procedure; dated August 3, 2006
- AREVA -Procedure Qualification 54-PQ-02; dated July 29, 2010
- AREVA- Report Noz 21-NDE-780-00- PT-200 Data Sheet-Liquid Penetrant Examination- J-Groove Weld Final Surface; dated August 31, 2011
- AREVA- Report CV-NDE-810;RPV; Penetration 21- Head Penetration UT Data Sheet- Circ Blade; dated September 2, 2011
- AREVA- Report CV-NDE-810; Penetration 21 RPV Head Penetration UT Data Sheet- Axial Blade; dated September 3, 2011
- AREVA- Report CV-NDE-870; RPV Head Penetration DM Weld UT Data Sheet- C; dated September 1, 2011
- AREVA- Reports CV-NDE-840, 850, 860; Inspection Report for the Rotating ID Exam of the Nozzles (10, 21, 25) – Eddy Current; dated September 1, 2011
- AREVA- Report CV-NDE-920, 930, 940; Inspection Report for the J-Groove Exam of the Nozzles (10, 21, 25) – Eddy Current; dated September 1, 2011
- AREVA- Report 91627770-000; Examination Summary Sheet- UT of Weld No. 1 CRDM Tube Housings; dated June 10, 2011
- AREVA- Calibration Sheet CAL-1-1A; Manual Ultrasonic Calibration Data Sheet- 45 Shear Flat; dated May 31, 2011
- AREVA- Calibration Sheet CAL-1-1A; Manual Ultrasonic Calibration Data Sheet- 45 Shear Contoured; dated May 31, 2011
- AREVA- UT Data Sheet DAT-1-01; Weld No. 1 – CRDM 1941, 1942, 1943 and 1944; dated June 1, 2011

Radiographic Film Records:
- CRDH No. 05; dated April 22, 2003
- CRDH No. 33; dated April 22, 2003
- CRDH No. 72; dated December 10, 2002
- CRDM No. 1619; dated November 22, 1977
- CRDM No. 1290; dated May 17, 1976
- CRDM No. 1941; dated April 10, 1979
- CRDM No. 1964; dated June 29, 1979
- Weld Procedures and Qualification Documents (AREVA)
- AREVA 55-GWP01-012; General Welding Procedure – 1 (GWP-1) ASME Code and Safety Related Applications; dated July 20, 2007
- AREVA 55-SPP02-013; Special Process Procedure – 2 (SPP-2) General Procedure for ARC Welding; dated February 12, 2009

- AREVA WP43/43/F43AW1; Manual Welding by GTAW of P-43 Nickel Chromium Iron Base Materials – WPS; dated May 23, 2006
- AREVA PQ7072-004; Manual Welding by GTAW/SMAW of P-43 Nickel Chromium Iron Base Materials PQR; dated June 18, 2007
- AREVA WP8/8F6AW1; Manual Welding by GTAW of P-8 Stainless Steel Base Materials without PWHT – WPS; dated March 25, 2010
- AREVA PQ7037-007; Manual Welding by GTAW/SMAW of P-8 Materials – PQR; dated April 26, 2006
- AREVA PQ7038-006; Manual Welding by GTAW/SMAW of P-8 Materials – PQR; dated April 26, 2006
- AREVA PQ7114-003; Manual Welding by GTAW/SMAW of P-8 Materials – PQR; dated January 15, 2004
- Framatome ANP -SOFSRX/NXX0150; Welding of Stainless (P8 G1) to Nickel Base Alloy (P43) by Gas Tungsten Arc Welding Process (GTAW)-WPS; dated February 21, 2002
- Framatome ANP- SOFSRX/NXX0152; Welding by SMAW Process of Nickel Alloy (P43) on Low Alloy Steel (P3 G3) Buttered with Nickel Alloy by SMAW -WPS; dated April 23, 2002
- Framatome ANP -SOFSRX/NXX0152; Welding by SMAW Process of Nickel Alloy (P43) on Low Alloy Steel (P3 G3) Buttered with Nickel Alloy by SMAW -PQR; dated August 22, 2002
- Framatome ANP- SOPRRX/NXX0150; Welding of Stainless (P8 G1) to Nickel Base Alloy (P43) by Gas Tungsten Arc Welding Process (GTAW)-PQR; dated April 15, 2002
- Framatome ANP -SOQRAS/NXX5022; Welding of Stainless (P8 G1) to Nickel Base Alloy (P43) by Gas Tungsten Arc Welding Process (GTAW)-WPQR; dated May17, 2002
- Framatome ANP -SOQRAS/NXX5378; Welding of Stainless (P8 G1) to Nickel Base Alloy (P43) by Gas Tungsten Arc Welding Process (GTAW)-WPQR; dated March 21, 2002

Weld Data Records (AREVA):
- Framatome ANP- Production Weld Data Sheets Operation 125 (GTAW of CRDH DM weld); March 11, 2003

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
CAC	Containment Air Cooler
CAL	Confirmatory Action Letter
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Housing Mechanism
dc	Direct Current
deg F	Degrees Fahrenheit
DRP	Division of Reactor Projects
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EN	Event Notification
FENOC	FirstEnergy Nuclear Operating Company
HPI	High Pressure Injection
I&C	Instrumentation and Controls
ICS	Integrated Control System
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Testing
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPI	Low Pressure Injection
MSPI	Mitigating Systems Performance Index
MT	Magnetic Particle
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
ODCM	Offsite Dose Calculation Manual
OpESS	Operating Experience Smart Sample
PARS	Publicly Available Records System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post-Maintenance Testing
PT	Dye Penetrant
QA	Quality Assurance

QAPM	Quality Assurance Program Manual
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RT	Radiographic
RVCH	Reactor Vessel Closure Head
SBODG	Station Blackout Diesel Generator
SDP	Significance Determination Process
SFAS	Safety Features Actuation System
SFP	Spent Fuel Pool
SSC	Systems, Structures, and Components
SW	Service Water
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
USAR	Updated Safety Analysis Report
URI	Unresolved Item
WO	Work Order

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Sincerely,

/RA/

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

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Letter to B. Allen from J. Cameron dated October 26, 2011.

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