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

October 24, 2012

Mr. Raymond Lieb
Site Vice President
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 05000346/2012004

Dear Mr. Lieb:

On September 30, 2012, 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 9, 2012, 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, one NRC-identified finding and one self-revealed finding of very low safety significance were identified. These two findings also involved violations of NRC requirements. Additionally, two licensee-identified violations are listed in Section 4OA7 of this report. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of any finding or NCV in this report, 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 Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000346/2012004
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:	05000346/2012004
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	July 1, 2012, through September 30, 2012
Inspectors:	D. Kimble, Senior Resident Inspector A. Wilson, Resident Inspector J. Bozga, Engineering Inspector T. Briley, Reactor Engineer J. Neurauter, Senior Engineering Inspector
Approved by	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS 1

REPORT DETAILS 3

Summary of Plant Status 3

1. REACTOR SAFETY 3
- 1R04 Equipment Alignment (71111.04) 3
- 1R05 Fire Protection (71111.05) 4
- 1R06 Flooding (71111.06) 5
- 1R07 Annual Heat Sink Performance (71111.07) 5
- 1R11 Licensed Operator Requalification Program (71111.11) 6
- 1R12 Maintenance Effectiveness (71111.12) 7
- 1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13) 8
- 1R15 Operability Determinations and Functionality Assessments (71111.15) 11
- 1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17 and 71007) 12
- 1R18 Plant Modifications (71111.18) 12
- 1R19 Post-Maintenance Testing (71111.19) 14
- 1R22 Surveillance Testing (71111.22) 15
- 1EP6 Drill Evaluation (71114.06) 17

4. OTHER ACTIVITIES 17
- 4OA1 Performance Indicator Verification (71151) 17
- 4OA2 Identification and Resolution of Problems (71152) 19
- 4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153) 20
- 4OA5 Other Activities 22
- 4OA6 Management Meetings 27
- 4OA7 Licensee-Identified Violations 28

SUPPLEMENTAL INFORMATION 1

Key Points of Contact 1

List of Items Opened, Closed and Discussed 2

List Of Documents Reviewed 3

List of Acronyms Used 14

SUMMARY OF FINDINGS

Inspection Report (IR) 05000346/2012004; 7/1/2012-9/30/2012; Davis-Besse Nuclear Power Station; Maintenance Risk Assessments and Emergent Work Control; and Other Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings were identified by the inspectors. Both of the findings were dispositioned as non-cited violations (NCVs) 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: Barrier Integrity

- Green. A self-revealed finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," were identified for the licensee's failure to properly implement the procedure for restoring power to motor control center (MCC) E16B. Specifically, the operator repositioned circuit breakers at the incorrect MCC, inadvertently removing power from plant equipment supplied by MCC E16A and causing an unplanned entry into Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.15, Condition A, for an inoperable channel of station vent normal range radiation monitoring. As an immediate corrective action, the operating crew performed steps to restore the unintentionally lost loads associated with MCC E16A and exited LCO 3.3.15 Condition A in a timely manner.

  This finding was associated with the Barrier Integrity Cornerstone because a high radiation level in the station vent, as measured by the radiation monitors, is used to detect a potential threat to control room personnel and automatically isolate the control room normal ventilation system. The inspectors determined that the finding was more than minor because, if left uncorrected, the failure to follow plant procedures and the mispositioning of plant equipment would have the potential to lead to a more significant safety concern. The inspectors evaluated the finding using IMC 0609, Appendix A, the "Significance Determination Process for Findings At-Power." The inspectors used Exhibit 2 – "Barrier Integrity Screening Questions for the Control Room, Auxiliary, Reactor, or Spent Fuel Pool Building." The finding screened as very low safety significance (Green) because it only represented a degradation of the radiological barrier function provided for the control room. The finding had a cross-cutting aspect in the area of human performance, work practices component, because personnel failed to use human error prevention techniques to ensure that work was performed safely. (H.4(a)) (Section 1R13.1)

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to use material specified minimum yield stress in accordance with American Institute for Steel Construction design standards in evaluations of safety-related structural components. The licensee entered this issue into their corrective action

- program (CAP) as condition reports (CRs) 2011-98333 and 2012-13249 and initiated corrective actions to resolve identified design standard non-conformance.

  The finding was determined to be more than minor because the finding was associated with the Barrier Integrity Cornerstone attribute of design control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, compliance with the design standards ensured safety-related structures would function as designed during accident and maximum seismic conditions. The finding was considered to be of very low safety significance since this was a design deficiency confirmed to not result in a loss of operability or functionality. The inspectors determined there was no cross-cutting aspect associated with this finding because the cause of the performance deficiency was the licensee's revision to the Updated Safety Analysis Report (USAR) that allowed certified material test report yield strength in structural design calculations which was not reflective of current licensee performance due to the age of the revision. (Section 4OA5.1)

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's CAP. These violations 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. During the entire course of the inspection period the unit remained operating at or near full power, with only slight periodic power reductions incurred to support the testing of certain equipment and components.

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:

- Auxiliary Feedwater (AFW) Train 1 when AFW Train 2 was unavailable for planned maintenance activities during the week ending July 21, 2012;
- Emergency Diesel Generator (EDG) 1 when EDG 2 was unavailable for a planned maintenance window during the week ending September 15, 2012; and
- The Station Blackout Diesel Generator (SBODG) when EDG 2 was unavailable for a planned maintenance window during the week ending September 15, 2012.

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 three 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 Resident Inspector Quarterly Fire Zone Inspections

a. Inspection Scope

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

- High Voltage Switchgear Rooms A and B (Rooms 323 and 325, Fire Areas Q and S);
- EDG No.1 (Rooms 318 and 318UL, Fire Area K);
- EDG No. 2 (Rooms 319 and 319A, Fire Area J); and
- Radwaste and Fuel Handling Areas, Radwaste and Main Station Exhaust Fan Room (Rooms 500 and 501, Fire Area EE).

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 to this report, 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 zone inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding Review

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the USAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant areas to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Areas of the auxiliary building, including the emergency core cooling system (ECCS) pump rooms, that could be affected by internal flooding from feedwater and fire suppression water.

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

This review by the inspectors constituted a single internal flooding inspection sample as defined in IP 71111.06-05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

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

This review by the inspectors constituted a single annual heat sink performance inspection sample as defined in IP 71111.07-05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Simulator Training

a. Inspection Scope

On July 31, 2012, the inspectors observed a crew of licensed operators in the plant's simulator during a graded simulator evaluation scenario to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and that training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance;
- The clarity and formality of communications;
- The ability of the crew to take timely and conservative actions;
- The crew's prioritization, interpretation, and verification of annunciator alarms;
- The correct use and implementation of abnormal and emergency procedures by the crew;
- Control board manipulations;
- The oversight and direction provided by licensed senior reactor operators (SROs); and
- The ability of the crew to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

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

This inspection constituted a single quarterly licensed operator requalification program sample as defined in IP 71111.11-05.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Control Room Activities

a. Inspection Scope

During the course of the inspection period, the inspectors performed numerous observations of licensed operator performance in the plant's control room to verify that operator performance was adequate and that plant evolutions were being conducted in accordance with approved plant procedures. Specific activities observed that involved a heightened tempo of activities or periods of elevated risk included, but were not limited to:

- Daily reactivity manipulations involving Reactor Coolant System (RCS) dilutions during the week ending September 8, 2012;
- Safety Features Actuation System (SFAS) periodic functional testing during the week ending September 8, 2012;
- Steam and Feedwater Rupture Control System (SFRCS) channel functional testing and calibrations during the week ending September 21, 2012; and
- Testing of fire protection (Appendix 'R') electrical circuits during the week ending September 21, 2012.

The inspectors evaluated the following areas during the course of the control room observations:

- Licensed operator performance;
- The clarity and formality of communications;
- The ability of the crew to take timely and conservative actions;
- The crew's prioritization, interpretation, and verification of annunciator alarms;
- The correct use and implementation of normal operating, annunciator alarm response, and abnormal operating procedures by the crew;
- Control board manipulations;
- The oversight and direction provided by on-watch SROs and plant management personnel; and
- The ability of the crew to identify and implement appropriate TS actions and notifications.

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

This inspection constituted a single quarterly observation sample of operator performance in the plant's control room as defined in IP 71111.11-05.

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 risk-significant systems:

- The station and instrument air systems; and
- The reactor shield building and external safety-related structures.

The inspectors reviewed systems where ineffective equipment maintenance had resulted or could result in valid or invalid automatic actuations of engineered safeguards systems 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.

The inspectors' reviews constituted two quarterly maintenance effectiveness inspection 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 Quarterly Reviews of 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 during the week ending July 7, 2012, following several days of elevated outside ambient air temperature that challenged numerous plant systems. The emergent work included response to a trip of station air compressor (SAC) No.1 resulting in its unavailability, high service water flow conditions that challenged secondary header pressure control, elevated turbine plant cooling water temperatures, a failure of the letdown system radiation monitor requiring compensatory actions, and a plant computer failure;
- Emergent work associated with failures in the station and instrument air systems during the week ending July 28, 2012;
- Emergent work and associated response and repair to a steam leak past the valve cap for a feedwater loop 1 drain valve during the week ending September 8, 2012; and
- Emergent work during the week ending September 22, 2012, after an equipment operator inadvertently removed power from plant equipment supplied by essential motor control center (MCC) E16A.

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.

Specific documents reviewed during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted four samples as defined in IP 71111.13-05.

b. Findings

Operator Error Restoring Essential Motor Control Center to Service Renders Technical Specification Equipment Inoperable

Introduction

A self-revealed finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified for the licensee’s failure to properly implement the procedure for restoring power to essential MCC E16B. Specifically, the operator repositioned circuit breakers at the incorrect MCC, inadvertently removing power from plant equipment supplied by MCC E16A and caused an unplanned entry into TS Limiting Condition for Operation (LCO) 3.3.15, Condition A, for an inoperable channel of station vent normal range radiation monitoring.

Description

On September 19, 2012, during the restoration of power to essential 480 Vac MCC E16B following planned maintenance, an equipment operator mistakenly opened all the load circuit breakers on essential 480 Vac MCC E16A. The mispositioning resulted in the loss of the following loads:

- Station Vent “A” Channel normal range RE4598AA sample pump;
- Station Vent “A” Channel emergency range RE4598AB sample pump;
- Central alarm station and computer room air conditioning;
- Constant Voltage Transformer XY1; and
- Constant Voltage Transformer XY3.

On-watch operators in the control room immediately entered TS LCO 3.3.15, Condition A, for an inoperable channel of station vent normal range radiation monitoring at 8:35 p.m., following receipt of control room alarms. In addition, the operators referenced plant documents and procedures to ensure that no other TS equipment was rendered inoperable. The operating crew performed actions to restore the unintentionally lost loads associated with MCC E16A. Limiting Condition for Operation 3.3.15, Condition A, was exited at 3:10 a.m. on September 20, 2012. Power was also eventually restored to MCC E16B.

The licensee's initial investigation into the event revealed that the operator did not correctly match the MCC panel identifier at the location with the specified MCC indicated in the procedure step. The operator had an incorrect mindset that he was to perform action on MCC E16A due to a previously completed step which verified a source breaker was energized at MCC E16A. The operator failed to exercise several human performance error prevention tools/techniques at the job site, and failed to adequately communicate his actions with the control room prior to manipulation of components. Additionally, the pre-job brief for the evolution was determined to not have been sufficiently thorough to detect and assess the worker's readiness for the job. An immediate corrective action was taken to require senior reactor operator oversight during all in-plant equipment manipulations until further notice in order to verify that the expected human performance tools were being utilized when field activities were being performed.

Analysis

The inspectors reviewed this finding using the guidance contained in Appendix B, Issue Screening, of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." The inspectors determined that the licensee's failure to properly implement the procedure for restoring power to MCC E16B was a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented. This finding was associated with the Barrier Integrity Cornerstone because a high radiation level in the station vent, as measured by the radiation monitors, is used to detect a potential threat to control room personnel and automatically isolate the control room normal ventilation system. The inspectors determined that the finding was of more than minor significance because, if left uncorrected, the failure to follow plant procedures and the mispositioning of plant equipment would have the potential to lead to a more significant safety concern.

The inspectors evaluated the finding using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors used Exhibit 2 – "Barrier Integrity Screening Questions for the Control Room, Auxiliary, Reactor, or Spent Fuel Pool (SFP) Building." The finding screened as very low safety significance (Green) because it only represented a degradation of the radiological barrier function provided for the control room, and nothing more.

This finding had a cross-cutting aspect in the area of human performance, work practices component, because personnel failed to use human error prevention techniques to ensure that work was performed safely. (H.4(a))

Enforcement

Appendix B of 10 CFR Part 50, 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 this requirement, on September 19, 2012, the licensee failed to correctly perform Step 3.2.6 of approved procedure DB-OP-06318, "120/240 Volt and 480 Volt MCC Switching," which states, "Verify all load breakers on MCC E16B are tripped." Instead, the operator repositioned circuit breakers at the incorrect MCC, inadvertently removing power from plant equipment supplied by MCC E16A and causing an unplanned entry into TS LCO 3.3.15, Condition A. The licensee included this issue in their CAP as CR 2012-14440. Because

this violation was of very low safety significance and it was entered into the licensee's corrective action program, it is being treated as an NCV, consistent with the NRC's Enforcement Policy. (NCV 05000346/2012004-01)

1R15 Operability Determinations and Functionality Assessments (71111.15)

.1 Quarterly Resident Inspector Review of Operability and Functionality Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- The operability and availability of incore thermocouple M7, as documented in CR 2012-11595;
- The operability of safety-related equipment located within the station's low voltage switchgear rooms when considering a high energy line break (HELB) in the turbine building, as documented in CR 2012-12992;
- The impact on the operability of safety-related structures due to water intrusion into the EDG building, as documented in CR 2012-12829; and
- The operability and availability of Reactor Coolant Pump (RCP) 1-1 due to degrading conditions with the pump's mechanical seal, as documented in CR 2012-12414.

The inspectors selected these potential operability and/or functionality 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 and/or functional. 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 appropriately identifying and correcting any deficiencies associated with operability and/or functionality evaluations. Documents reviewed are listed in the Attachment to this report.

The inspectors' reviews of these operability and functionality evaluations constituted four inspection samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17 and 71007)

.1 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

From July 18, 2011, through August 30, 2012, the inspectors reviewed two safety evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors reviewed these documents to determine if:

- The changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- The safety issue requiring the change, tests or experiment was resolved;
- The licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- The design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations, and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC IMC, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

The inspectors' reviews of these safety evaluations constituted two inspection samples, and completed the required sample size as defined in IP 71111.17-04.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modification

a. Inspection Scope

The inspectors reviewed the following temporary modification to the facility:

- Installation of temporary cooling to station air compressor (SAC) No. 1 from the station's domestic water system.

The inspectors reviewed the configuration changes and associated 10 CFR Part 50.59 safety evaluation documents against the design basis, the USAR, and the TS, as applicable, to verify that the temporary modification did not affect the operability or availability of any safety-related systems, or systems important to safety. The inspectors conducted a field walkdown of the completed work activities to ensure that the

modification was installed as directed and consistent with the design control documents; that the modification operated as expected; that post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modification did not impact the operability, availability, or functionality of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modifications with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the temporary modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

The inspectors' review of this temporary plant modification constituted a single inspection sample as defined in IP 71111.18-05.

b. Findings

No findings were identified.

.2 Permanent Plant Modification

a. Inspection Scope

The inspectors reviewed the following permanent modification to the facility:

- Application of an exterior water-resistant coating to the containment shield building walls and dome.

During a mid-cycle outage to replace the reactor vessel closure head in late 2011, the licensee identified laminar cracking in the safety-related shield building of the containment system while performing hydrodemolition operations to create a shield building maintenance access opening. Based on an evaluation of the licensee's extent-of-condition and technical analysis of the shield building laminar cracking, the NRC staff concluded that the licensee had provided reasonable assurance that the shield building was capable of performing its safety functions. In order to provide continued long-term confidence, the licensee agreed to several follow-on actions. Chief amongst these follow-on actions was the licensee's commitment to perform an investigation into the root cause of the cracking.

The licensee submitted its root cause report (ADAMS Accession No. ML120600056) to the NRC on February 27, 2012. The licensee identified the direct cause as the integrated effect of moisture content, wind speed, temperature, and duration from a severe winter blizzard that occurred in 1978, and the root cause as the design specification for construction of the shield building not specifying application of an exterior sealant from moisture. The licensee also identified three contributing causes involving specific design features of the building. The root cause report also identified planned corrective actions as well as associated due dates, and acknowledged that the shield building, although operable, did not conform to the licensing basis in its current condition.

The NRC completed an inspection of the licensee's root cause efforts and planned corrective actions on May 9, 2012 (NRC IR 05000346/2012009; ADAMS Accession No. ML12173A023). The NRC inspection team concluded that the licensee had a sufficient

basis for the causes of the shield building laminar cracking related to the environmental factors associated with the 1978 blizzard, the lack of an exterior moisture barrier, and the structural design elements of the shield building. The team did, however, identify minor weaknesses in the licensee's root cause report associated with the level of detail in the documentation provided. These weaknesses did not constitute performance deficiencies or findings because they did not adversely affect the outcome of the root cause process.

The licensee submitted a revised root cause report (ADAMS Accession No. ML12142A053) on May 16, 2012, with changes to address the minor weaknesses identified during the NRC inspection. NRC follow-up inspection plans to this issue are focused on verification and evaluation of licensee corrective action implementation, and a principal corrective action is the licensee's permanent modification to coat the containment shield building walls and dome with an exterior water-resistant sealant.

The inspectors reviewed the permanent modification plans and associated 10 CFR 50.59 safety evaluation documents against the design basis, the USAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of any safety-related systems, or systems important to safety. The inspectors observed ongoing sealant application work activities to ensure that the modification was installed as directed and consistent with the design control documents; that the sealant performed as expected; that applicable post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that the application of the sealant did not adversely impact the operability, availability, or functionality of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the permanent plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the sealant could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

The inspectors' review of this permanent plant modification constituted a single inspection sample as defined in IP 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Quarterly Resident Inspector Observation and Review of Post-Maintenance Testing Activities

a. Inspection Scope

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

- Observation of the SBODG test run and review of the test results following a planned maintenance window during the week ending July 14, 2012;

- Observation of the SAC No. 2 test run and review of the test results following troubleshooting and corrective maintenance during the week ending July 28, 2012; and
- Observation of EDG No. 2 post-maintenance test runs and review of the test results following a planned maintenance outage during the week ending September 15, 2012.

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 inspectors' reviews of these activities constituted three PMT inspection 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-MI-03246; "Channel Functional Test and Device Calibration of SFRCS Steam Generator Level Inputs to Actuation Channel 2," during the week ending August 18, 2012 (routine);
- DB-ME-03000; "Station Battery and Charger Weekly Surveillance," during the week ending August 25, 2012 (routine);
- DB-MI-03011; "Channel Functional Test of Reactor Trip Breaker B, RPS Channel 1 Reactor Trip Module Logic, and ARTS Channel 1 Output Logic," during the week ending August 25, 2012 (routine);

- DB-OP-01101; “Containment Entry,” during the week ending September 22, 2012 (routine); and
- DB-SP-04150; “Auxiliary Feedwater Pump (AFP) 1 Monthly Test,” during the week ending September 29, 2012 (routine).

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency 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, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

The inspectors’ reviews of these activities constituted five routine surveillance testing inspection samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of an integrated licensee emergency drill on September 13, 2012, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room simulator, technical support center, and emergency operations facility to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

The inspectors' review of this emergency preparedness (EP) drill constituted a single inspection sample as defined in IP 71114.06-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Occupational Radiation Safety, Public 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 for the period from the third quarter of 2011 through the second quarter of 2012. 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, event reports, MSPI derivation reports, and NRC Integrated Inspection Reports for the period of July 2011 through June 2012 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 issue report database to determine if any problems had been identified with the performance indicator (PI) data

collected or transmitted for this indicator, and none were identified. Documents reviewed are listed in the Attachment to this report.

The inspectors' review of this performance indicator data constituted a single MSPI – Heat Removal System inspection 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 the third quarter of 2011 through the second quarter of 2012. 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 of July 2011 through June 2012 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 issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified. Documents reviewed are listed in the Attachment to this report.

The inspectors' review of this performance indicator data constituted a single MSPI – Residual Heat Removal System inspection 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 indicator for the period from the third quarter of 2011 through the second quarter of 2012. 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 of July 2011 through June 2012 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 issue report database to determine if any problems had been identified

with the PI data collected or transmitted for this indicator, and none were identified. Documents reviewed are listed in the Attachment to this report.

The inspectors' review of this performance indicator data constituted a single MSPI – Cooling Water Systems inspection 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 condition report 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 Annual Sample: Review of Operator Workarounds

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the operator workarounds (OWAs) on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment to this report were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an initiating event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of mitigating systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workarounds.

This review constituted a single operator workaround annual inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

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

.1 (Closed) Licensee Event Report 05000346/2012-001-00: Direct Current Source for Diesel Generator Transferred to Inoperable Source During Fuel Movement

On the morning of May 19, 2012, the unit was offline and in a refueling (Mode 6) condition, with reactor core offload activities in progress. Emergency Diesel Generator (EDG) No. 2 was inoperable and unavailable while undergoing extensive planned modifications to its exhaust piping. Emergency Diesel Generator No. 1 was operable, and was being supported by Direct Current (DC) Train No. 2. The cross-connection of the DC essential distribution panels for maintenance purposes while in Modes 5 or 6 is typically utilized in each refueling outage during the battery preventive maintenance testing, and is permitted by plant TS.

At approximately 8:03 p.m., with maintenance activities completed on DC Train No. 1, plant operators conducted a briefing to swap DC loads from DC Train No. 2 back to Train No. 1. The live swap of essential DC loads, including DC support for EDG No. 1, was completed at approximately 10:31 p.m. Movement of irradiated fuel assemblies to facilitate reactor core offload continued throughout this timeframe, and was completed on May 20, 2012, at approximately 3:48 p.m. At approximately 11:30 p.m., with the reactor defueled and all movement of irradiated fuel in the spent fuel pool (SFP) suspended, the licensee declared EDG No. 1 inoperable to facilitate a planned swap of its supporting cooling equipment trains. This work was completed by approximately 2:07 a.m. on the morning of May 21, 2012, and EDG No. 1 was declared operable. Movement of irradiated fuel in the SFP recommenced at approximately 2:22 a.m.

On the morning of May 22, 2012, at approximately 9:56 a.m., plant operators identified that required review and approvals of several WOs associated with the DC Train No. 1 maintenance had not yet been completed. While both fully functional and available, DC Train No. 1 was not operable per TS. As a result, EDG No. 1 was also not operable per TS during the timeframe when it was being supported by DC Train No.1. A hold was immediately placed on the movement of irradiated fuel assemblies in the SFP, and plant operators initiated a formal hold on entry back into Mode 6 to preclude core refueling until the issue could be resolved. At approximately 11:44 a.m., the licensee completed the requisite reviews for the DC Train No. 1 maintenance, conducted an inspection of DC Train No. 1 equipment to verify proper operation, and declared DC Train No. 1 and EDG No. 1 operable per TS requirements.

In response to this issue, the licensee conducted an evaluation into the root cause. This was determined to be less than adequate administrative controls for maintaining DC system power source operability with the distribution network cross-tied while in cold shutdown. In addition to immediately performing the reviews necessary to support TS operability for DC Train No. 1, other corrective actions performed by the licensee included a revision to the station DC switching procedure to ensure operability prior to distribution panel transfers, revision to applicable pre-job briefings, enhancements to the outage schedule to address DC power source availability, and a case study on DC system restoration during outages.

The inspectors' review of this event determined that the licensee's failure to maintain DC Train No. 1 and EDG No.1 operable during the period of movement of irradiated fuel assemblies constituted a licensee-identified violation of TS 3.8.2, which was of very low safety significance. Further details of this licensee-identified violation are discussed in Section 4OA7.1 of this report. The licensee had entered this event into their CAP as CR 2012-08422. Documents reviewed as part of this inspection are listed in the Attachment. This Licensee Event Report (LER) is closed.

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

.2 <u>(Closed) Licensee Event Report 05000346/2012-002-00: Leak From Reactor Coolant Pump Seal Piping Socket Weld Due to High Cycle Fatigue</u>

On June 6, 2012, the unit was offline and in a hot standby (Mode 3) condition with the reactor coolant system (RCS) at normal operating pressure and temperature. Plant personnel were in the process of conducting scheduled visual inspections of RCS components for leakage as part of the regular sequence of events required to return the

plant to operation following reactor refueling activities. During the course of these inspections, plant engineering personnel identified a small pinhole leak on a socket weld on the first stage seal cavity vent line (¾ inch diameter) for reactor coolant pump (RCP) 1-2. The leak was estimated to be approximately 0.1 gpm, and due to its location could not be isolated from the RCS.

In order to meet the requirements of TS 3.4.13 for RCS pressure boundary leakage, the licensee commenced a plant cooldown. The unit entered a cold shutdown (Mode 5) condition on June 7, 2012. Utilizing a freeze seal to isolate the pinhole leak from the RCS, the licensee effected repairs by grinding out the weld defect and then restoring the socket weld to its original design on June 11, 2012. Plant restart followed at that point without any further issues.

In response to this issue, the licensee conducted an evaluation into the cause. While a definitive cause could not be established due to the fact that all forensic evidence related to the defect was eliminated by the nature of the repair technique (i.e., grinding out the weld defect and performing a re-weld, etc.), the licensee established that the most probable cause for the leak had been a high-cycle fatigue failure. Using operating experience from the industry, the licensee postulated that the leak resulted from a combination of a less than adequate design for the RCP vibration conditions in combination with a discontinuity that was most probably induced during the initial weld's root pass. These conditions have existed since 1990, when the licensee modified their RCP seal cavity vent lines to accommodate a new style of RCP seal package.

In addition to the repairs made to the leak on the RCP 1-2 first stage seal cavity vent line, other corrective actions performed by the licensee included inspections of all similar RCP seal cavity vent lines for any signs of leakage. Ultimately, the licensee has plans to replace all of the current RCP seal cavity vent lines with flex hose connections during the next refueling outage in 2014.

The inspectors' review of this event determined that the licensee's actions in response to the event were appropriate, and that no violations of any NRC requirements were involved. However, the inspectors' review did note a licensee-identified violation of very low safety significance involving the prior modifications in 1990 to the plant's RCP seal cavity vent lines. Further details of this licensee-identified violation are discussed in Section 4OA7.2 of this report. The licensee entered this event into their CAP as CR 2012-09381. Documents reviewed as part of this inspection are listed in the Attachment. This LER is closed.

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

4OA5 Other Activities

.1 Reactor Vessel Head Replacement – Plant Modifications (71007)

The original reactor vessel closure head (RVCH) penetration nozzles were fabricated from Inconel Alloy 600 material. These nozzles were welded to the RVCH with a partial penetration weld fabricated from Inconel Alloy 182 weld filler metal. Pressurized water reactors have experienced pressure boundary leakage caused by primary water stress corrosion cracking (PWSCC) of these materials. In 2002, the licensee replaced the original RVCH with a RVCH of similar design with materials that were susceptible to

PWSCC. As documented in IR 05000346/2010-008 dated October 22, 2010 (ADAMS Accession No. ML102930380), it was concluded that reactor coolant pressure boundary leakage had occurred due to cracks in the reactor vessel head control rod drive penetration nozzles and J-groove welds, and the licensee repaired 24 control rod drive mechanism (CRDM) nozzle locations with PWSCC indications in the J-groove weld or nozzle base material prior to restarting the plant.

a. Inspection Scope

From July 18, 2011 through August 30, 2012, the inspectors performed a review of modifications and activities related to replacement of the RVCH and CRDM housings and installation of a new integrated head assembly (IHA) in accordance with Sections 02.02 and 02.04 of IP 71007, "Reactor Vessel Head Replacement Inspection." This review was performed to determine if the replacement RVCH was designed in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, the IHA was designed in accordance with established standards, the containment vessel temporary construction opening was in accordance with the design assumptions and restoration maintained required strength and integrity, and lay-down areas were determined to have sufficient structural capacity to handle the loads by calculations. Specifically, the inspectors reviewed records associated with the following activities:

(1) Reactor Vessel Closure Head and Control Rod Drive Mechanism Housing Replacement

The configuration of the new replacement RVCH was similar to the existing RVCH, with several notable differences in the configuration, components, or materials that included:

- The replacement RVCH was a one-piece forging, including both the flange and domed hemisphere;
- The replacement RVCH o-ring groove dimensions have changed consistent with replacement o-ring dimensional changes;
- The replacement CRDM nozzle was formed from SB-167 Alloy 690 material. The replacement CRDM nozzle was attached to the ERNiCrFe-7 weld butter inside of the replacement RVCH with an ERNiCrFe-7 or ENiCrFe-7 weld. The new materials were considered to be an improvement because they were designed to be resistant to cracking due to PWSCC;
- The new continuous vent line was relocated from existing RVCH penetration 14 to replacement RVCH penetration 21;
- The seven unused CRDM adapter flanges were capped with new SA-182 F304L blind flanges. The use of new CRDM blind flanges in lieu of the vented blind flange and associated piping was considered a design improvement since removal of the components reduced the potential for component failure or leakage, and the potential for failure at the CRDM flange was reduced; and
- The IHA support skirt flange (segmented skirt) was bolted directly to the new IHA lower shroud. The CRDM upper service structure support skirt was replaced by the IHA lower shroud.

The inspectors reviewed certified design specifications, certified design reports, ASME Code reconciliation reports, fabrication deviation notices, non-conformance reports, and design calculations to confirm that the replacement RVCH and CRDM housings were in compliance with the requirements of ASME Boiler and Pressure Vessel Code,

Section III, Subsection NB (1989 Edition). The inspectors confirmed that the design specifications and design reports were certified by registered professional engineers competent in ASME Code requirements. The inspectors confirmed that adequate documentation existed to demonstrate the certifying registered professional engineers were qualified in accordance with the requirements of the ASME Code Section III. The inspectors also confirmed that the replacement RVCH was procured as a Code 'NPT' stamped component.

As part of the RVCH and CRDM housing replacements, the inspectors also reviewed the licensee's engineering change that provided a temporary construction opening in the steel containment vessel and subsequent containment vessel restoration that facilitated removal of the existing RVCH and installation of the replacement RVCH. In addition, the inspectors reviewed calculations that demonstrated lay-down areas supporting transport of the existing and replacement RVCHs had sufficient structural capacity to handle the loads.

(2) Integrated Head Assembly

During the fall 2011 mid-cycle outage, the licensee installed a reactor IHA that incorporated various plant components and structures into the reactor head assembly design. This integration involved the reuse of some plant components and the complete replacement of others. Major component enhancements/changes included:

- Integral ventilation system for CRDMs;
- Integral radiation shielding;
- Retractable cable bridges (batwings);
- IHA lower shroud openings and inspection doors;
- Replacement of RVCH continuous vent line;
- Re-route of plant side component cooling water piping; and
- The addition of an integrated service air manifold.

The inspectors reviewed the licensee's design documentation associated with the installation of the IHA. Specifically, the inspectors reviewed the IHA equipment design specification, the IHA design report, and a representative sample of design documents to confirm that IHA structures and components were designed in accordance with the codes and standards specified in the IHA equipment design specification.

Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection," is an infrequently performed inspection procedure specified by IMC 2515, Appendix C; as such, these reactor vessel head replacement reviews performed by the inspectors did not represent any baseline inspection program samples. The records reviewed by the inspectors are identified in the Attachment to this report.

b. Findings

Failure to Use Material Specified Minimum Yield Stress in Structural Design

Introduction

A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the

licensee's failure to establish adequate measures to assure that appropriate quality standards were specified and included in the design documents.

Description

The requirements for seismic design of Class I and Class II structures were specified in Section 3.8 of the licensee's USAR. For the Shield Building, Section 3.8.2.2.3 of the USAR, referenced structural steel conformance to the American Institute for Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," Sixth Edition. For containment vessel internal structures, Section 3.8.2.3.3 of the USAR specified structural steel conformance to the AISC "Manual of Steel Construction," Seventh Edition.

The inspectors identified that the licensee used certified material test report (CMTR) yield stress data instead of material specified minimum yield stress in accordance with the AISC design standards referenced in the USAR for the design of structural steel. Specifically, the inspectors identified that design calculation C-CSS-099.11-022, "Evaluation of Containment Elevation 603 (Floor Loading) for 17M RPVH Replacement," Revision 0, approved on May 26, 2011, used yield stress values based on CMTR data instead of yield stress as defined in the AISC standard. This calculation used the AISC Manual of Steel Construction, Allowable Stress Design, Ninth Edition, to determine the allowable bending stress based on material yield stress. This standard defined yield stress in its definition of a symbol section: Fy: "Specified minimum yield stress of the type of steel being used, ksi. As used in this Specification, 'yield stress' denotes either the specified minimum yield point (for those steels with a yield point) or specified minimum yield strength (for those steel without a yield point)." The inspectors concluded that the use of actual material yield strength from CMTR data to determine allowable bending stress was not in conformance with the AISC design standard. The licensee entered the concern into their CAP as CR 2011-98333. As part of CR 2011-98333 corrective actions, the licensee revised calculation C-CSS-099.11-022 to demonstrate conformance with the AISC design standard prior to placement of RVCHs onto the floor at Containment Elevation 603.

As documented in CR 2011-98333, the licensee indicated that the Seventh Edition of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" was used to design the plant's structural steel and that this code defined yield stress, Fy, as the "specified minimum yield stress of the type of steel being used." In addition, the inspectors verified that the AISC Manual of Steel Construction, Eighth Edition, General Nomenclature, defined yield stress, Fy, consistent with the Ninth Edition.

The inspectors further identified that for accident and maximum seismic design conditions, Section 3.8.1.3.1.b of the USAR stated, in-part, that "in backfit re-analysis situations, the actual certified mill test report yield strength of the structural member may be used to determine the AISC code allowable stresses." The inspectors questioned if the licensee limited use of CMTR yield strength to establish structural component "operability or functionality" (the component has sufficient structural strength to perform its design safety function). As indicated in CR 2011-98333, the licensee identified CMTR yield strength was used in additional structural design calculations instead of the material specified minimum yield stress. Therefore, the inspectors concluded that some design calculations for the licensee's safety-related structural components were not in

conformance with the AISC design standards. However, the inspectors further concluded that the use of CMTR yield strength provided reasonable assurance that an affected structural component had sufficient structural strength to perform its design basis safety function (i.e., operable or functional, but non-conforming to the AISC design standards). The licensee initiated corrective actions to address AISC non-compliance as part of CR 2011-98333. In addition, the licensee initiated CR 2012-13249 to modify the USAR to remove use of CMTR data in design basis structural evaluations.

Analysis

The inspectors determined that failure to establish adequate measures to assure that appropriate quality standards are specified and included in the design documents was contrary to 10 CFR Part 50, Appendix B, Criterion III, and constituted a licensee performance deficiency.

The finding was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of design control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, compliance with AISC requirements ensured safety-related structures would function as designed during accident and maximum seismic conditions. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process" (issued June 2, 2011), Attachment 0609.04, "Phase I – Initial Screening and Characterization of Findings," Table 4a, for the Barrier Integrity Cornerstone. The finding was screened as having very low safety significance (Green) because it was a design deficiency of the physical integrity of safety-related structures. The inspectors answered "no" to all the questions in the Containment Barrier column based on the fact that structural components designed using CMTR yield strength remained operable or functional.

This finding did not have a cross-cutting aspect because the inspectors concluded that the cause of the performance deficiency was the licensee's revision to the USAR that allowed CMTR yield strength in structural design calculations which was not reflective of current licensee performance due to the age of the revision.

Enforcement

Criterion III of 10 CFR Part 50, Appendix B, "Design Control," requires, in-part, that: "measures to assure that appropriate quality standards are specified and included in the design documents."

Contrary to this requirement, as of July 28, 2011, the licensee had not established adequate measures to ensure compliance with design standards specified for evaluation of safety-related structures. Specifically, calculation C-CSS-099.11-022, Revision 0, approved on May 26, 2011, used CMTR yield stress data instead of material specified minimum yield stress as defined in the AISC design standard for evaluation of structural components. In addition, the licensee identified other Davis-Besse Nuclear Power Station safety-related structural components that were designed using CMTR yield stress data instead of material specified minimum yield stress. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CRs 2011-98333 and 2012-13249, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000346/2012004-02)

.2 (Discussed) NRC Temporary Instruction 2515/187, Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns, and NRC Temporary Instruction 2515/188, Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

Inspectors accompanied the licensee on a sampling basis, during their flooding and seismic walkdowns, to verify that the licensee's walkdown activities were conducted using the methodology endorsed by the NRC. These walkdowns are being performed at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 3 of the March 12, 2012, letter requested licensees to perform seismic walkdowns using an NRC-endorsed walkdown methodology. Electric Power Research Institute (EPRI) document 1025286 titled, "Seismic Walkdown Guidance," (ADAMS Accession No. ML12188A031) provided the NRC-endorsed methodology for performing seismic walkdowns to verify that plant features, credited in the current licensing basis (CLB) for seismic events, are available, functional, and properly maintained.

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, are available, functional, and properly maintained. The inspectors accompanied the licensee on their walkdown of the wave protection dike to confirm flood protection features were in place.

b. Findings

The licensee's flooding walkdowns were not fully complete at the time the inspection period ended; inspectors' review of licensee's seismic review activities was not completed at the end of the inspection procedure. Findings or violations associated with the flooding and seismic walkdowns, if any, will be documented in the fourth quarter integrated inspection report.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 9, 2012, the inspectors presented the inspection results to the Site Vice President, Mr. Raymond Lieb, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection of modifications associated with the reactor vessel head replacement (IP 71007), which were discussed with the outgoing Site Vice President, Mr. Barry Allen, and other members of the licensee staff on August 30, 2012. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee, or destroyed using an approved method of destruction for sensitive material.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy for being dispositioned as NCVs.

.1 Inadequate Administrative Controls Result in Inoperable Emergency Diesel Generator and Essential Direct Current Distribution Equipment During Movement of Irradiated Fuel Assemblies

Technical Specification 3.8.2, "AC Sources – Shutdown," LCO 3.8.2(b) requires that one EDG capable of supplying one train of the onsite Class 1E ac electrical power distribution subsystems required by LCO 3.8.10 be maintained operable in Modes 5 and 6, and during the movement of irradiated fuel assemblies.

As discussed in Section 4OA3.1 of this report, contrary to this requirement, licensee personnel failed to maintain EDG No.1 operable during the movement of irradiated fuel assemblies from approximately 10:31 p.m. on May 19, 2012, until all movement of irradiated fuel was completed at approximately 6:00 p.m. on May 20, 2012, and then again during the movement of irradiated fuel assemblies in the SFP during various periods on May 21 – 22, 2012. A licensee causal evaluation team concluded that this error resulted from less than adequate administrative controls for maintaining dc system power source operability with the distribution network cross-tied while in the cold shutdown and refueling plant conditions.

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 configuration control, and specifically, control of operating and shutdown equipment alignment. 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 maintain adequate administrative control over the operability of DC Train No. 1, such that it was allowed to be realigned to support EDG No. 1 operability without first being declared operable itself, caused EDG No. 1 to be rendered inoperable when it was required by TS to be operable during periods of irradiated fuel movement.

As discussed in Section 4OA3.1 of this report, the licensee had entered this issue into their CAP as CR 2012-08422. Immediate corrective actions taken by the licensee included placing a hold on the movement of irradiated fuel assemblies in the SFP, and

initiation of a Mode 6 entry hold to preclude core refueling until the issue could be resolved. In addition, the licensee immediately began performing the reviews necessary to support TS operability for DC Train No. 1. Other subsequent corrective actions performed by the licensee included a revision to the station DC switching procedure to ensure operability prior to distribution panel transfers, revision to applicable pre-job briefings, enhancements to the outage schedule to address DC power source availability, and a case study on DC system restoration during outages.

.2 Inadequate Design Control Results in Reactor Coolant System Pressure Boundary Leakage From Reactor Coolant Pump Seal Piping Socket Weld

Appendix B to 10 CFR Part 50, Criterion III, "Design Control," requires, in part, that the licensee establish measures for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of SSCs. Criterion III further requires that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design.

As discussed in Section 4OA3.2 of this report, contrary to this requirement, licensee personnel failed to properly review the suitability of a modification that was performed to their RCP seal cavity vent lines to accommodate a new style of RCP seal package in 1990. Specifically, the RCP seal cavity vent lines were lengthened by approximately five inches, and the licensee's engineering design personnel failed to consider what the impact of changing the small bore (i.e., ¾ inch diameter) piping length would have on the piping resonance frequencies and the piping socket welded connections. Industry operating experience has shown that minor changes to small bore piping can result in higher amplitude vibrations, potentially resulting in high-cycle fatigue failure. A licensee causal evaluation team concluded that a pinhole leak through a socket weld on the RCP 1-2 first stage seal cavity vent line that occurred on June 6, 2012, was most probably this kind of high-cycle fatigue failure.

The objective of the Barrier Integrity Cornerstone of Reactor Safety is to provide reasonable assurance that physical design barriers (fuel cladding, RCS, and containment) protect the public from radionuclide releases caused by accidents or events. Key attributes of this objective are design control, and specifically plant modifications, and RCS equipment and barrier performance. 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 consider what the impact of changing the small bore piping length would have on the piping resonance frequencies and the piping socket welded connections resulted in a pinhole failure of the RCS pressure boundary, and compromised the RCS barrier performance. The inspectors also determined that since the licensee's performance deficiency had occurred in 1990, the licensee-identified violation constituted an Old Design Issue, as defined by the NRC Enforcement Policy, which was not indicative of current licensee performance.

As discussed in Section 4OA3.2 of this report, the licensee had entered this issue into their CAP as CR 2012-09381. Immediate corrective actions taken by the licensee included repair of the leak on the RCP 1-2 first stage seal cavity vent line, as well as inspections of all similar RCP seal cavity vent lines for any signs of leakage. The

licensee has plans to replace all of the current RCP seal cavity vent lines with flex hose connections during the next refuel outage in 2014.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Allen, Site Vice President (Outgoing)
B. Boles, Director, Site Operations
K. Byrd, Director, Site Engineering
T. Chowdhary, NRC Liaison
J. Cuff, Manager, Training
A. Dawson, Manager, Chemistry
J. Dominy, Director, Site Maintenance
D. Gerren, Manager, Steam Generator Replacement Project
J. Hook, Manager, Design Engineering
D. Imlay, Director, Site Performance Improvement
G. Kendrick, Manager, Site Outage Management
B. Kremer, Manager, Plant Engineering
R. Lieb, Site Vice President (Incoming)
P. McCloskey, Manager, Site Regulatory Compliance
D. Noble, Manager, Radiation Protection
W. O'Malley, Manager, Nuclear Oversight
R. Oesterle, Superintendent, Nuclear Operations
M. Parker, Manager, Site Protection
R. Patrick, Manager, Site Work Management
D. Petro, Manager, Steam Generator Replacement Project
T. Summers, Manager, Site Operations
C. Price, Director, Special Projects (Outgoing)
M. Roelant, Manager, Site Projects
L. Rushing, Director, Special Projects (Incoming)
D. Saltz, Manager, Site Maintenance
C. Steenbergen, Superintendent, Operations Training
J. Sturdavant, Regulatory Compliance
L. Thomas, Manager, Nuclear Supply Chain
M. Travis, Superintendent, Radiation Protection
J. Vetter, Manager, Emergency Response
A. Wise, Manager, Technical Services
G. Wolf, Supervisor, Regulatory Compliance
K. Zellers, Supervisor, Reactor Engineering

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000346/2012004-01	NCV	Operator Error Restoring Essential MCC to Service Renders TS Equipment Inoperable (Section 1R13.1)
05000346/2012004-02	NCV	Failure to Use Material Specified Minimum Yield Stress in Structural Design (Section 4OA5.1)

Closed

05000346/2012004-01	NCV	Operator Error Restoring Essential MCC to Service Renders TS Equipment Inoperable (Section 1R13.1)
05000346/2012004-02	NCV	Failure to Use Material Specified Minimum Yield Stress in Structural Design (Section 4OA5.1)
05000346/2012-001-00	LER	Direct Current Source for Diesel Generator Transferred to Inoperable Source During Fuel Movement (Sections 4OA3.1 and 4OA7.1)
05000346/2012-002-00	LER	Leak From Reactor Coolant Pump Seal Piping Socket Weld Due to High Cycle Fatigue (Sections 4OA3.2 and 4OA7.2)

Discussed

2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns (Section 4OA5.2)
2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns (Section 4OA5.2)

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-06233; Auxiliary Feedwater System; Revision 32
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 52
- DB-OP-06334; Station Blackout Diesel Generator Operating Procedure; Revision 19

Drawings:
- OS-017A, Sheet 1; Auxiliary Feedwater System; Revision 26

1R05 Fire Protection

Procedures:
- PFP-AB-318; Diesel Generator 1-1 Room, Fire Area K; Revision 7
- PFP-AB-319; Diesel Generator 1-2 Room, Fire Area J; Revision 7
- PFP-AB-320A; Diesel Fuel Oil Day Tank 1-2 Room, Fire Area J; Revision 4
- PFP-AB-321A; Diesel Fuel Oil Day Tank 1-1 Room, Fire Area K; Revision 4
- PFP-AB-323; High Voltage Switchgear Room B, Room 323, Fire Area Q; Revision 5
- PFP-AB-325; High Voltage Switchgear Room A, Room 325, Fire Area S; Revision 5
- PFP-AB-500; Radwaste and Fuel Handling Areas, Room 500, Fire Area EE; Revision 4
- PFP-AB-501; Radwaste Exhaust Equipment and Main Station Exhaust Fan Room, Room 501, Fire Area EE; Revision 4

Drawings:
- A-0223F; Fire Protection General Floor Plan El. 585’ - 0”; Revision 22
- A-0225F; Fire Protection General Floor Plan El. 623’ - 0”; Revision 18

Other:
- Fire Hazard Analysis Report; Revision 24

1R06 Flood Protection Measures

Condition Reports:
- 2007-22619; CDBI – HPI Pump 1 ECCS Sump Flowrate

Procedures:
- RA-EP-02880; Internal Flooding; Revision 3

Calculations:
- VF14/B000-082; Concrete Adequacy Check of Rooms 200 & 201 Due to Flooding – Auxiliary Building
- 05.039; Feedwater Flooding in the Auxiliary Building; Revision 0

- 15.50; Evaluation of Fire Suppression System Impact on Auxiliary Building and Intake Structure Flooding; Revisions 0 and 1
- 58.020; Flooding in the ECCS Rooms Due to a FWLB; Revision 1, Addendum 1
- 63.013; Possible Flooding Levels for Various Rooms; Revision 1

1R07 Heat Sink Performance

Condition Reports:
- 2010-79719; SBODG High Water Temperature Alarm During Monthly Run
- 2011-97983; SBODG High Water Temperature Alarmed During Loaded Run Using DB-OP-06334
- 2012-11584; Continuing SBODG High Jacket Water Temperature

Procedures:
- DB-OP-06334; SBODG Operating Procedure; Revision 18

Work Orders:
- 200451295; Engine Analysis on SBODG
- 200473121; SBODG Radiator Chemical Cleaning
- 200475172; Replace SBODG Test Cock Valves and Seals

Other:
- NALCO Chemical Company; SBODG Closed Loop Cooling Water Cleaning Report; 7/13/2012

1R11 Licensed Operator Requalification Program and Licensed Operator Performance

Condition Reports:
- 2012-14402; DB-SC-03003 Revision 5 Limited Use Change for Testing of Appendix R Circuits

Procedures:
- NT-OT-7001; Training and Qualification of Operations Personnel; Revision 12
- B-MI-03203; Channel Functional Test and Calibration of SFRCS Actuation Channel 1, Steam Generator Differential Pressure Inputs PDS-2686A, PDS-2686B, PDS-2685C, PDS-2685D; Revision 10
- DB-OP-06902; Power Operations; Revision 37
- DB-SC-03003; Testing of Appendix R Circuits for AC1CE11, AC1CE12, BCE11, BCE12; Revision 5

Business Practices:
- DBBP-TRAN-0014; License Requirements for Licensed Operators; Revision 9
- DBBP-TRAN-0021; Simulator Configuration Control; Revision 3
- DBBP-TRAN-0502; Development of Continuing Training Simulator Evaluation; Revision 7
- NOBP-TR-1112; FENOC Conduct of Simulator Training and Evaluation; Revision 1

1R12 Maintenance Effectiveness

Condition Reports:
- 2012-12455; Temporary Diesel Air Compressor No. 1 Surging
- 2012-11541; Training Not Received for Temporary Diesel Air Compressor Operation
- 2012-11519; Station Air Compressor 2 Would Not Load
- 2012-10650; SAC 1 Tripped

- 2012-11166; SAC 1 Trip Due to High Discharge Temperature
- 2012-10803; TPCW ODMI Revision
- 2012-11303; SAC 1 Auto Started When it was Supposed to be Tripped
- 2012-11132; Auto Start of both Station Air Compressors 1 and 2 Due to Trip of Diesel Driven Air Compressor
- 2012-11298; NOBP-TR-1122, Operating Crew Performance Critique for Station Air Compressor 1 Trip
- 2012-09985; Need ODMI for Limited TPCW Flow Margin
- 2012-00048; TPCW System Loss of Operating Margin
- 2011-97930; Unable to Maintain Requested MVAR Output Due to Limitations on the TPCW System
- 2011-97881; SAC 1 Trip During High Ambient Temperature Condition
- 2012-00504; Guidance for Use of Temporary Diesel Air Compressor

Work Orders:
- 600716444; SAC 1 Interim High Temperatures Guidance

Engineering Change Packages (ECPs):
- 12-0208; Exterior Coating Systems For the Shield Building Walls and Dome; Revisions 0 and 1

Other:
- Aggreko PREMAIR Diesel Air Compressor Maintenance & Operating Manual
- Maintenance Rule Program Manual; Revision 30

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

Condition Reports:
- 2012-10591; TPCW System Flow Readings
- 2012-10656; Main Steam Line Room Temperatures Above 120 F
- 2012-10569; Main Transformer Doble IDD Issues
- 2012-10609; The Indicating Reading for RE1998 in Analyzer Mode is Trending Lower
- 2012-10650; SAC 1 Tripped
- 2012-10704; Plant Computer Failure
- 2012-10497; Breaker HBBD Noise
- 2012-10810; SW Secondary Header Pressure Challenged by High Flow Conditions
- 2012-10721; Review DBRM-EMER-5003 Comp Action for RE1998 Being Out of Service
- 2012-12455; Temporary Diesel Air Compressor No. 1 Surging
- 2012-11670; AC212 (Startup Feedwater Pump) Will Not Rack IN
- 2012-11132; Auto Start of Both Station Air Compressors 1 and 2 Due to Trip of Diesel Driven Air Compressor
- 2012-11391; Service Water Pump #1 Strainer Excessive Mechanical Seal Leakage
- 2012-11502; SA23 Possible Rework
- 2012-11519; Station Air Compressor 2 Would Not Load
- 2012-11673; SAC 2 Failed to Load During Confidence Run Per DB-OP-06251
- 2012-13513; Steam Leak at FW26
- 2012-13423; CC1467 Regulator Failure
- 2012-14440; Mispositioning Breakers on E16A

Procedures:
- NOP-OP-1007; Risk Management; Revision 9
- DB-MM-09067; Temporary Leak Sealing; Revision 12
- DB-OP-02525; Steam Leaks; Revision 10
- DB-OP-06318; 120/240 Volt and 480 Volt MCC Switching; Revision 15
- NOBP-OP-0004; Plant Status Control and Clearance Events; Revision 6
- NOBP-LP-3002; Incident/Near Miss Response and Reporting; Revision 8
- NOBP-LP-2602; Human Performance Success Clocks and Quick Human Error Response; Revision 9

Work Orders:
- 200529329; FW26 Online Leak Seal TM ECP 12-0607

Engineering Change Packages:
- 12-0607; TM Installation Related Steam Leak at Cap Near FW26; Revision 0

Other:
- Davis-Besse Weekly Maintenance Risk Summary, Week of July 1, 2012; Revision 0
- Davis-Besse Weekly Maintenance Risk Summary, Week of July 23, 2012; Revisions 0,1,2
- Davis-Besse Weekly Maintenance Risk Summary, Week of September 2, 2012; Revision 0
- Night Shift Narrative Unit Log; September 20, 2012

<u>1R15 Operability Determinations and Functionality Assessments</u>

Condition Reports:
- 2010-80584; RCP 1-2 3rd stage seal degradation
- 2012-07520; Decay Heat Pump No. 2 Seal Leakage
- 2012-07993; VT-2 Examination of RCP 1-1 Class 1 Bolting Test Zone RC13
- 2012-08170; Indications of Water Intrusion in Emergency Diesel Generator Room No. 1, Room 318
- 2012-09554; Incore Detector String 13 Thermocouple
- 2012-10674; RCP 1-1 increased demin water flush flow and increased seal leakage trend
- 2012-11595; Review of Technical Specification Bases for Currently Inoperable Incore Thermocouple M7 Identified Concern
- 2012-11737; RCP 1-1 Seal Leakage Rate is rising
- 2012-12414; ODMI: Reactor Coolant Pump 1-1 has indications of seal degradation Operation Decision Making Issue (ODMI): Recommendations for operating with degraded RCP 1-1 third stage seal performance
- 2012-12789; Tracking CR: Repair Emergency Diesel Generator Roof Leak per Order 200449943
- 2012-12829; Water Intrusion Into EDG Room No. 1
- 2012-12992; NRCCDBI2012 – Switchgear Room Ventilation During HELBs
- 2012-14004; NRCCDBI2012 – Turbine Building HELB Impact on Switchgear Rooms

Procedures:
- DB-MI-04272; Incore Detector and Cable Assembly Post Installation Test; Revision 9
- DB-OP-02515; Reactor Coolant Pump and Motor Abnormal Operation; Revision 6

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications Evaluations

10 CFR 50.59 Safety Evaluations:
- 10-05488; SGR-17M-Containment Vessel Opening; 06/09/2011
- 11-02493; SGR-17M-New RV Service Structure; 11/02/2011

1R18 Plant Modifications

Engineering Change Packages (ECPs):
- 12-0208; Exterior Coating Systems For the Shield Building Walls and Dome; Revisions 0 and 1
- 12-0519; Temporary Modification – Install Alternate Cooling to Station Air Compressor 1 (SAC 1), C101-1; Revisions 0 and 1

1R19 Post Maintenance Testing

Condition Reports:
- 2010-79719; SBODG High Water Temperature Alarm During Monthly Run
- 2011-97983; SBODG High Water Temperature Alarmed During Loaded Run Using DB-OP-06334
- 2012-11584; Continuing SBODG High Jacket Water Temperature
- 2012-11502; SA23 Possible Rework
- 2012-11519; Station Air Compressor 2 Would Not Load
- 2012-11673; SAC 2 Failed to Load During Confidence Run Per DB-OP-06251
- 2012-14216; PA-DB-12-03: Duty Team Requirement for Orange Risk to Generation Was Not Met

Procedures:
- DB-OP-06334; SBODG Operating Procedure; Revision 18
- DB-OP-06251; Station and Instrument Air Operating Procedure; Revision 28
- DB-SC-03077; Emergency Diesel Generator 2 184 Day Test; Revision 26
- DB-SC-03081; Emergency Diesel Generator 2 Overspeed Trip Test; Revision 7
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 52
- DB-MM-09320; Emergency and Station Blackout Diesel Engine Maintenance; Revision 25

Work Orders:
- 200451295; Engine Analysis on SBODG
- 200473121; SBODG Radiator Chemical Cleaning
- 200475172; Replace SBODG Test Cock Valves and Seals
- 200514220; Station Air 23 – Inspect/Replace
- 200355987; Power Pack Assembly Replacement
- 200341248; PM 1353 EDG 2 6 Year PMs
- 200352054; EDG 2 J398 KW Reading Low
- 200391238; PM 2908 PS5157 & PSL5166 Calibration EDG#2 Air Receiver

Other:
- NALCO Chemical Company; SBODG Closed Loop Cooling Water Cleaning Report; 7/13/2012

1R22 Surveillance Testing

Condition Reports:
- 2012-12528; High Voltage Reading Across 1P Battery Shunt
- 2012-14411; September 2012 Quarterly Containment Inspection; Debris Found in CTMT
- 2012-14415; September 2012 Quarterly Containment Inspection; Green Dust in Containment
- 2012-14423; September 2012 Quarterly Containment Inspection; Oil Noted on Floor of 585' Elevation
- 2012-14425; September 2012 Quarterly Containment Inspection; Oil on Floor 603' Elevation

Procedures:
- DB-MI-03246; Channel Functional Test and Device Calibration of SFRCS Steam Generator Level Inputs 83C-ISLSP9A8, A9, B6 and B7 to Actuation Channel 2; Revision 13
- DB-MI-03011; Channel Functional Test of Reactor Trip Breaker B, RPS Channel 1 Reactor Trip Module Logic, and ARTS Channel 1 Output Logic; Revision 27
- DB-ME-03000; Station Battery and Charger Weekly Surveillance; Revision 22
- DB-OP-01101; Containment Entry; Revision 10
- DB-SP-04150; AFP 1 Monthly Test; Revision 11

Calculations:
- C-ICE-083.03-001; SFRCS Low and High Level Setpoints; Revision 17

Other:
- ALARA Plan #2012-2005; Containment Entry at Power; Revision 0

1EP6 Drill Evaluation

Condition Reports:
- 2012-14053; EP Drill – DBAB Emergency Response Facility South Badging Station Door Key Card Access Required for NRC Resident Inspectors
- 2012-14056; EP Drill – Additional Guidance is Required in DBRM-EMER-1500A for the Use of the "F" Curves
- 2012-14109; EP Drill – Area for Improvement – Accumulative Dose Tracking for Personnel Exposure
- 2012-14114; EP Drill – Failed to Successfully Demonstrate ERO Callout Within 10 Minutes of the Event Declaration
- 2012-14115; EP Drill – EAPM Was Not Informed of ATWS Event Upon Arrival in Simulator Control Room

Reference Manuals:
- DBRM-EMER-1500A; Davis-Besse Emergency Action Level Basis Document; Revision 3

Drawings and Charts:
- DBRM-EMER-1500B; Hot EAL Wall Board, Revision 1
- DBRM-EMER-1500B; Cold EAL Wall Board, Revision 1

Calculations:
- EP-EALCALC-DB-0701; Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss and CT2 Potential Loss); Revision 0

Other:
- Nuclear Energy Institute (NEI) 99-01; Methodology for Development of Emergency Action Levels; Revision 5
- Emergency Preparedness Integrated Drill Manual, September 13, 2012; Revision 0

4OA1 Performance Indicator Verification

Condition Reports:
- 2012-13678; Unavailability Data Reported for Auxiliary Feedwater (AFW) Train 2 in the January 2012 and February 2012 Mitigating System Performance Index (MSPI) is Incorrect
- 2012-00982; Possible Inaccurate Reporting of Mitigating System Performance Index (MSPI) Data

Forms:
- NOBP-LP-4012-48; MSPI Heat Removal System (AFW); Completed Forms for July 2011 through June 2012
- NOBP-LP-4012-49; MSPI Residual Heat Removal System (LPI); Completed Forms for July 2011 through June 2012
- NOBP-LP-4012-50; MSPI Support Cooling System, Component Cooling Water; Completed Forms for July 2011 through June 2012
- NOBP-LP-4012-51; MSPI Support Cooling System, Service Water; Completed Forms for July 2011 through June 2012

Procedures:
- DB-SP-03161; AFW Train 2 Level Control, Interlock, and Flow Transmitter Test; Revision 28
- 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 2011 through June 2012
- Maintenance Rule Unavailability Database covering the period of July 2011 through June 2012
- Mitigating System Performance Index Basis Document, Davis-Besse Nuclear Power Station; Revision 4

4OA2 Problem Identification and Resolution

Condition Reports:
- 2012-13654; D-RPO-17: Operator Burdens Performance Indicator Yellow Monthly and 12-Month Average Performance
- 2012-13656; D-RPO-15: Control Room Deficiencies Performance Indicator Yellow Monthly Performance

Business Practices and Procedures:
- NOBP-OP-0012; Operator Work-Arounds, Burdens and Control Room Deficiencies; Revision 01

Quarterly Aggregate Impact Reports for Operator Workarounds, Burdens, and Control Room Deficiencies:
- Second Quarter 2012 Report; 6/29/2012
- Third Quarter 2012 Report; 9/25/2012

4OA3 Followup of Events and Notices of Enforcement Discretion

Condition Reports:
- 2012-08422; DC MCC Busses 1P and 1N Not Supplied by Operable DC Sources
- 2012-09381; During DB-PF-03010 NOP/NOT: Active Leak on RCP 1-2 1st Seal Cavity Vent Line
- 2012-09452; Failure to Establish a Root Cause Evaluation Method for CR 2012-09381

Procedures:
- DB-PF-03010; Reactor Coolant System Leakage Test; Revision 11

NRC Event Notification Worksheet:
- EN # 48000; Degraded Condition Due to Discovery of Pressure Boundary Leakage

4OA5 Other Activities

Condition Reports:
- 2011-96511; Reactor Vessel Closure Head Temperature Affect on Transients
- 2011-98333; NRC RRCH 2011 Use of CMTR Material Properties for Design Basis Structural Calculations
- 2011-03647; Referenced Calculations C-CSS-099.20-50 and C-CSS-099.20-51 Incorrectly Referenced in ECP 10-0459
- 2012-09209; Reactor Head IHA Calculation Failed to Address All ASME Fillet Weld Allowable Stress Criteria
- 2012-04213; NRC RVH Inspection Questions RVH 2011-0214 and RVH 2011-0222
- 2012-04173; NRC RVCH: Areva Calculation 32-9034634-003
- 2012-04487; NRC RVCH Inspection: Question 2011-0214 Continuous Vent Line Design Requirements
- 2012-05192; NRC RVH Inspection Question RVH 2011-0224-Temporary Lead Shielding
- 2012-05193; NRC RVH Inspection Questions RVH 2011-00226 and RVG 2011-00227 Continuous Vent Line Terminal Line Breaks
- 2012-13249; NRCRRCH2011 Failure to Revise USAR to Remove Use of CMTR Yield Strength Values for Design Basis Structural Evaluations

Procedures:
- DB-MM-04010; Periodic Test Procedure for Special Lifting Devices; Revision 9

Engineering Change Packages (ECPs):
- 10-0459-000; SGR-17M-Containment Vessel Opening; Revision 0
- 10-0469-000; Replacement RVCH and CDRMs and New RVCH Vent Nozzle; Revision 3
- 10-0470-000; SGR-17M-New Reactor Vessel Service Structure; Revision 3

10 CFR 50.59 Screenings and Safety Evaluations:
- Screening 11-02492; Replacement RVCH and CDRMs and New RVCH Vent Nozzle; Revision 0
- Screening 10-05488; SGR-17M-Containment Vessel Opening; Revision 0
- Screening 11-02492; SGR-17M-New RVCH and CRDMs; 10/17/2011
- Screening 11-02493; SGR-17M-New RV Service Structure; 11/02/2011
- Evaluation 11-02493; SGR-17M-New RV Service Structure; 11/02/2011
- Evaluation 93-0023; Safety Evaluation for Modification 92-004 Supplement 1; 03/30/1993

Drawings:
- 02-146047B; Flange Ring; Revision 2
- 02-5015884E; Specification Drawing for Replacement Reactor Vessel Closure Head Davis-Besse Unit 1; Revision 6
- 02-5019935D; CRDM Nozzle Weld Preparations and Welding 177 FA Heads; Revision 1
- 02-9140840C; Bottom Ring Beam Assembly; Revision 1
- 02-9140843C; Radiation Shield Assembly; Revision 3
- 02-9141037B; CRDM Blind Flange Assembly; Revision 1
- 02-9144531D; Design Drawing for RRCVH Modification for Installation of a Continuous Vent Piping Nozzle for Davis-Besse 1; Revision 2
- 02-9150411D; Specification Drawing for RRVCH Modifications for Installation of a Continuous Vent Piping Nozzle for Davis-Besse 1; Revision 1
- 02-9151530C; Shroud Assembly to RVCH, Connection Details; Revision 3

Calculations:
- 32-5026263; Calculation: DB-1 RV Head CRDM Housing Nozzle; Revision 2
- 32-5027094; Calculation: DB-1 RV Closure Analysis with Replacement Head; Revision 0
- 32-5027094; Calculation: DB-1 RV Closure Analysis with Replacement Head; Revision 2
- 32-5027513; Calculation: Davis Bessse-1 Reactor Vessel Closure Head Sizing Calculation; Revision 3
- 32-5046469; Calculation: DB-1 Section III Appendix G Closure Head (Chalon) P-T Limits; Revision 2
- 32-9037975; Calculation: DB-1 Section XII RV Closure Head (Chalon) Limits; Revision 2
- 32-9146083; Calculation: DB-1 Replacement CRDM Blind Flange Structural Analysis; Revision 0
- 32-9147091; Calculation: DB-1 Replacement RVCH Continuous Vent Nozzle Analysis; Revision 2
- 32-9149238; Calculation: Segmented Skirt Weld Analysis; Revision 0
- 32-9149862; Calculation: DB Continuous Vent Line Piping Analysis; Revision 4
- 33-1179616; Davis Besse Unit 1 Type “C” Replacement CRDMs; Revision 3
- 33-5046434; ASME Design Report Certification Document: Davis Besse – Unit 1 Replacement Reactor Closure Head; Revision 2
- 51-5043387; DB-1 Replacement RV Closure Head Reconciliation; Revision 2
- 51-9146694; Davis-Besse Chalon Replacement RVCH/IHA 630 Degree Fahrenheit Fluid Temperature Reconciliation; Revision 0
- 51-9147329; Davis-Bess Continuous Vent Piping – Code Reconciliation; Revision 2
- 51-9147331; Davis-Besse RRVCH with Replacement Vent Nozzle – Code Reconciliation; Revision 1
- 51-9164798; Material Reconciliation for Control Rod Drive Mechanism Motor Tube for Davis-Besse Unit-1; Revision 1
- 51-9166078; Davis-Besse CDM Replacement Reconciliation; Revision 2
- 51-9166279; DB-1 Replacement CRDM Blind Flange Reconciliation; Revision 0
- 87-5037876; CVAR for Oversize CRDM Holes; Revision 0
- 87-5038383; CAR for Oversize CARDS Support; Revision 0
- 87-5053804; CVAR – DB-1 Ferrite Number Violation; Revision 0
- 87-5064209; CVAR – Davis Besse RVCH Hardware Reclassification; Revision 0
- Addendum 1 to C-CSS-059.01-003; Revision 0; Maximum External Pressure Capacity of Containment Using ASME Code Subsection NE; May 20, 2011
- Addendum 1 to C-CSS-059.01-004; Revision 0; Maximum External Pressure Capacity of Containment Using ASME Code Case N-184; 05/20/2011

- Addendum 1 to C-CSS-059.01-005; Revision 0; Containment Vessel – External Pressure; 05/20/2011
- Addendum 1 to C-CSS-059.01-008; Revision 5; Containment Vessel Analysis; 05/20/2011
- C-CSS-059.01-017; Monorail System for Rigging and Storing Containment Vessel for Construction Opening; Revision 0
- Addendum 1 to C-CSS-059.01-017; Revision 0; Monorail System for Rigging and Storing Containment Vessel Opening Door Sheet; 10/21/2011
- C-CSS-059.01-018; Evaluation of Containment Vessel for Construction Opening; Revision 0
- C-CSS-059.01-019; ASME Evaluation of the Containment Vessel for RVCH Replacement Outage Permanent Attachments; Revision 0
- C-CSS-059.01-020; Permanent Stress Change in the Containment Vessel Due to Creation of the Temporary Construction Openings During 17 Mid Cycle and 18 RFO Outages; Revision 0
- Addendum 1 to C-CSS-059.01-020; Revision 0; Permanent Stress Change in the Containment Vessel Due to Creation of the Temporary Construction Openings During 17 Mid Cycle and 18 RFO Outages; 09/20/2011
- C-CSS-059.01-021; Potential Tornado Missile Impact on Containment Vessel with Construction Opening in Shield Building; Revision 0
- Addendum 1 to C-CSS-059.01-021; Revision 0; Potential Tornado Missile Impact on Containment Vessel with Construction Opening in Shield Building; 10/20/2011
- 32-1151112; Calculation: TED Handling Fixture; Revision 7
- 32-1151112; Calculation: TED Handling Fixture; Revision 8
- 32-9034634; Calculation: DB-1 Lifting Pendant, Turnbuckle, Clevis, Pin, Tee Lug, and Cover Plate Analysis; Revision 4
- 32-9037953; Calculation: DB-1 Internal Handling Adapters Re-Analysis; Revision 3
- 32-9037953; Calculation: DB-1 Internal Handling Adapters Re-Analysis; Revision 4
- 32-9149862; Calculation: DB Continuous Vent Piping Analysis; Revision 4
- 32-9150029; Calculation: Davis-Besse Unit 1 Continuous Vent Piping Support Analysis; Revision 0
- 51-9151802; Calculation: Continuous Vent Piping Supports for Davis-Besse Unit 1; Revision 4
- 32-9150029; Calculation: Davis-Besse Unit 1 Continuous Vent Piping Support Analysis; Revision 0
- 32-9150029; Calculation: Davis-Besse Unit 1 Continuous Vent Piping Support Analysis; Revision 1
- C-CSS-099.11-022; Evaluation of Containment Elevation 603 (Floor Loading) for 17M RPVH Replacement; Revision 0
- C-CSS-099.11-022; Evaluation of Containment Elevation 603 (Floor Loading) for 17M RPVH Replacement; Revision 1

Specifications:
- 08-5015881; Certified Design Specification: Reactor Vessel Closure Replacement Davis-Besse-1; Revision 8
- 08-9140479; Certified Design Specification: Continuous Vent Piping, Davis-Besse Unit 1; Revision 4
- 08-9144523; Certified Design Specification: Continuous Vent Piping Nozzle, Davis-Besse Unit 1; Revision 4
- 08-9146243; Design Specification: CRDM Blind Flange for Davis-Besse Unit 1; Revision 1
- 18-1149327; Functional Specification: Reactor Coolant System for Davis-Besse; Revision 3
- 18-5015883; Load Specification: DB-1 Replacement Reactor Vessel Head; Revision 1
- 08-8041356; Equipment Design Specification: Integrated Head Assembly for FENOC, Davis Besse Unit 1; Revision 2
- 51-9030308; DB-1 Design Requirements for RV Head Fixed Lifting Pendants; Revision 2

- 51-9151802; Technical Requirements for Continuous Vent Piping Supports For Davis-Besse Unit 1; Revision 4
- 33-9133574; Davis-Besse Unit 1 Reactor Vessel Head Replacement Project Integrated Head Assembly (IHA) Design Report; Revision 0; 07/11/2011
- 33-9133574; Davis-Besse Unit 1 Reactor Vessel Head Replacement Project Integrated Head Assembly (IHA) Design Report; Revision 1; 10/20/2011
- TR-6108; Pipe Stress Analysis Report for Reactor Head to Hot Leg Vent Line; Revision 4; Addendum A01
- TR-6108; Pipe Stress Analysis Report for Reactor Head to Hot Leg Vent Line; Revision 4; Addendum A02

Other:
- 17-VT-297; Visual Examination System Leakage (VT-2): Reactor Coolant System In-Service Leak Test; 12/05/2011
- OE30898; Operating Experience: Indications on Multiple Control Rod Drive Mechanism Nozzles (Davis-Besse); 04/07/2010
- OE31664; Operating Experience: (Update to OE30893) Indications on Multiple Control Rod Drive Mechanism Nozzles (Davis-Besse); 07/29/2010
- Prompt Operability Determination 2012-04487; 03/24/2012
- Toledo Edison Company Letter to NRC; Subject: Use of ASME Code Case N-411; 03/14/1985
- NRC letter to Toledo Edison Company; Subject: Use of ASME Code Case N-411; 04/15/1985

4OA7 Licensee-Identified Violations

Condition Reports:
- 2012-08422; DC MCC Busses 1P and 1N Not Supplied by Operable DC Sources
- 2012-09381; During DB-PF-03010 NOP/NOT: Active Leak on RCP 1-2 1st Seal Cavity Vent Line
- 2012-09452; Failure to Establish a Root Cause Evaluation Method for CR 2012-09381

Procedures:
- DB-PF-03010; Reactor Coolant System Leakage Test; Revision 11

NRC Event Notification Worksheet:
- EN # 48000; Degraded Condition Due to Discovery of Pressure Boundary Leakage

LIST OF ACRONYMS USED

ac	Alternating Current
ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
AISC	American Institute for Steel Construction
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CMTR	Certified Material Test Report
CR	Condition Report
CRDM	Control Rod Drive Mechanism
dc	Direct Current
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
HELB	High Energy Line Break
IHA	Integrated Head Assembly
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
ksi	Thousands of Pounds Per Square Inch
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MCC	Motor Control Center
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OWA	Operator Workaround
PARS	Publicly Available Records System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Post Maintenance
PMT	Post-Maintenance Testing
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RVCH	Reactor Vessel Closure Head
SAC	Station Air Compressor
SBODG	Station Blackout Diesel Generator
SDP	Significance Determination Process
SFAS	Safety Features Actuation System
SFP	Spent Fuel Pool
SRO	Senior Reactor Operator
SSC	Structures, Systems, and Components
SW	Service Water

TS	Technical Specification
USAR	Updated Safety Analysis Report
WO	Work Order

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 Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000346/2012004
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

DOCUMENT NAME: G:\DRPIII\DAVI\DAVI 2012 004.docx
☐ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☐ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII					
NAME	JRutkowski:dtp		JCameron					
DATE	10/23/12		10/24/12					

OFFICIAL RECORD COPY

Letter to R. Lieb from J. Cameron dated October 24, 2012.

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION INTEGRATED INSPECTION REPORT 05000346/2012004

DISTRIBUTION:
Cayetano Santos
RidsNrrDorlLpl3-2 Resource
RidsNrrPMDavisBesse Resource
RidsNrrDirsIrib Resource
Chuck Casto
Cynthia Pederson
Steven Orth
Jared Heck
Allan Barker
Christine Lipa
Carole Ariano
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports.Resource@nrc.gov