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

May 5, 2008

EA-03-0214
EA-07-0199

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

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2008002

Dear Mr. Allen:

On March 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Davis-Besse Nuclear Power Station. The enclosed inspection report documents the inspection findings which were discussed on April 14, 2008, with you and other members of your staff. Additionally, this inspection report documents special inspection activities associated with your compliance with the requirements of Confirmatory Order (EA-03-0214) and of Confirmatory Order (EA-07-0199).

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.

The report documents one NRC-identified finding and two self-revealing findings of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements. Additionally, licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Davis-Besse Nuclear Power Station.

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

Sincerely,

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

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

Enclosure: Inspection Report 05000346/2008002
w/Attachment: Supplemental Information

cc w/encl: The Honorable Dennis Kucinich
J. Hagan, President and Chief
Nuclear Officer - FENOC
J. Lash, Senior Vice President of
Operations and Chief Operating Officer - FENOC
Manager - Site Regulatory Compliance - FENOC
D. Pace, Senior Vice President of
Fleet Engineering - FENOC
J. Rinckel, Vice President, Fleet Oversight - FENOC
D. Jenkins, Attorney, FirstEnergy Corp.
Director, Fleet Regulatory Affairs - FENOC
Manager - Fleet Licensing - FENOC
C. O'Claire, Chief, Ohio Emergency Management Agency
R. Owen, Administrator, Ohio Department of Health
Public Utilities Commission of Ohio
President, Lucas County Board of Commissioners
President, Ottawa County Board of Commissioners

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Letter to B. Allen from J. Cameron dated May 5, 2008

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2008002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	January 1, 2008, through March 31, 2008
Inspectors:	J. Rutkowski, Senior Resident Inspector R. Smith, Resident Inspector T. Bilik, Reactor Inspector T. Go, Health Physicist J. Jacobson, Senior Reactor Inspector J. Jandovitz, Region III Reactor Inspector R. Jickling, Senior Emergency Preparedness Analyst M. Phalen, Health Physicist T. Taylor, Reactor Engineer P. Voss, Reactor Engineer A. Wilson, Reactor Engineer G. Wright, Project Engineer
Observer:	J. Draper, Reactor Engineer
Approved by:	J. Cameron, Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000346/2008002; 1/01/08 – 3/31/08; Davis-Besse Nuclear Power Station; Inservice Inspection Activities, Identification and Resolution of Problems, Event Followup

This report covers a three-month period of inspection by resident inspectors and announced baseline and supplemental inspections by regional inspectors. Three Green findings, of which two were non-cited violations (NCVs) were identified. 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-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure of two contractor welders to adhere to welding procedures for the weld overlay (WOL) repairs on two pressurizer safety relief valve (SRV) nozzles. Specifically, both welders failed to use calculated relative travel speed settings as required by procedure in order to ensure that correct heat input values (an essential variable) were maintained. The licensee entered the issue into their corrective action program and suspended welding activities on the two SRVs until it was determined that the travel speeds used resulted in a heat input that was bounded by the procedure as qualified.

  This finding is greater than minor because if left uncorrected it would have become a more significant safety concern in that the failure to control heat input could have reduced the impact toughness of the WOL such that it would be susceptible to brittle fracture. The finding is of very low safety significance because calculations determined that the resulting heat inputs were bound by the welding procedure specifications' (WPS) parameters. As a result, assuming worst case degradation, it is unlikely that there would be reactor coolant system leakage or the loss of safety function of any mitigating system. The cause of the finding is related to the cross-cutting aspect of Human Performance, Work Practices, (Item H.4.(c)) because licensee personnel failed to ensure supervisory and management oversight activities of their contractors such that nuclear safety was ensured. (Section 1R08)

- Green. A self-revealing finding was identified for the failure of operators to maintain configuration control of valves during an air pressure test of a repair of a feedwater heater. Specifically, the operators left valve RD198 open during a pressure test of the extraction steam, or shell side, of feedwater heater 1-5 of the Main Feedwater System. This loss of configuration control gave testing air a path to the main condensers and led to degradation of the condenser vacuum, which then caused the Integrated Control System to raise reactor power unexpectedly. No violation occurred. Once the issue was identified, the licensee stopped the air pressure test and entered the finding into their corrective action program.

The finding is greater than minor since it was associated with the configuration control-operating equipment lineup attribute of the Initiating Events Cornerstone and because it affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. The finding is of very low safety significance since it did not contribute to the likelihood of a primary or secondary system loss of coolant accident, did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available, and did not increase the likelihood of a fire or internal/external flood. The finding was associated with the cross-cutting area of human performance in that work control and specifically the coordination of work activities did not properly record or assess the status of a valve in the test boundary and created a condition that had an operational impact (H.3(b)). (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. A self-revealing NCV of TS 4.5.2b was identified for failure to properly fill and vent a portion of decay heat/low pressure injection train 1 after maintenance which resulted in an approximate 15 cubic foot air void in the discharge piping of the train 1 decay heat/low pressure injection system for approximately 59 days of plant operation. Work packages and procedures used in restoration and refilling of the system did not adequately identify and provide for filling of drained high points in the piping. Upon identification with a 6 inch step decrease in pressurizer level while aligning the decay heat system for refueling operations, the licensee filled and then vented the system from high point vents located in system's discharge piping in the plant's containment. The licensee entered the failure to properly fill and vent the system after maintenance in their corrective action program.

  The finding is greater than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone, and affected the cornerstone objective in that permitting an air void in the train's discharge piping affected the reliability and capability of the system. The finding is of very low safety significance because it did not result in a loss of function per Part 9900, "Technical Guidance – Operability Determinations and Functionality Assessments," did not represent an actual loss of safety function, and is not potentially risk significant due to external events. The finding is associated with the cross-cutting area of human performance in that the resources and specifically work packages and procedures were not adequate to ensure that the train 1 decay heat/low pressure injection system was restored to a filled and vented system condition (H.2(c)) after completion of maintenance activities. (Section 4OA2)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period the plant was in mode 5 with preparations ongoing to drain the reactor coolant system to the level of the reactor vessel flange as part of scheduled activities for the plant's fifthteenth refueling outage. On January 31, 2008, the reactor was made critical as part of normal plant startup activities. On February 1, 2008, because of high vibrations on the turbine generator, the licensee was not able to connect the generator with the electrical grid.

On February 5, 2008, the reactor was placed in mode 3 while activities were conducted to correct the vibration problems with the main generator. On February 14, 2008, after replacing amortisseur windings in the main generator, the reactor was taken critical with generator synchronization and connection to the electrical grid occurring the same day. The unit attained full operating power on February 19, 2008.

On March 7, 2008, after an unexpected small power increase due to a feedwater heater tube rupture, the licensee reduced power to approximately 95 percent. After repairs to the degraded feedwater heater, the plant was returned to 100 percent power on March 11, 2008.

At the end of the inspection period the plant was operating at approximately 100 percent power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness For Impending Adverse Weather Condition – Heavy Rainfall/External Flooding Conditions

a. Inspection Scope

The inspectors conducted an external flooding walkdown of plant external areas and rooftops during a period of heavy rainfall February 5-7, 2008, and verified plant operations were in accordance with their flooding procedures given the environmental conditions. The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Final Safety Analysis Report (UFSAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constitutes one readiness for impending adverse weather condition sample as defined in Inspection Procedure 71111.01-05

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Motor driven feedwater system on February 19 and 20, 2008, after performance of the systems quarterly functional test; and
- High pressure injection system train 2 during train 1 maintenance on March 18, 2008.

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 Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, Administrative TS, outstanding work orders, condition reports, 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 with the appropriate significance characterization. Documents reviewed are listed in the attachment.

These activities constituted two partial system walkdown samples as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Emergency diesel generator 1 room (Fire Zone K, Room 318);
- High voltage switchgear room 1 (Fire Zone S, Room 325);
- Electrical Penetration Room 1 (Fire Zone DG, Room 402);
- Mechanical Penetration Room 4 (Fire Zone A, Room 314); and
- Containment Elevations 565' and 585' (Fire Zone D, Rooms 215, 220, and 317).

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 had 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 with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

These activities constituted five quarterly fire protection inspection sample as defined by Inspection Procedure 71111.05-05.

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On March 26, 2008, the inspectors observed a fire brigade activation for a simulated fire from main turbine bearing 3. The observation evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and availability of fire hoses; (3) employment of appropriate fire fighting techniques; (4) firefighting equipment to be brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) need for a search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

These activities constituted one annual fire protection inspection sample as defined by Inspection Procedure 71111.05-05.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

The inspections described in Sections 1R08, 1R08.2, 1R08.3, and 1R08.4 below count as one inspection sample.

.1 Piping Systems ISI

a. Inspection Scope

From January 7, 2008, through January 14, 2008, the inspectors conducted a review of the implementation of the licensee's Risk-Informed Inservice Inspection Program (RI-ISI) for monitoring degradation of the reactor coolant system boundary, and the risk significant piping system boundaries. The inspectors selected the licensee's RI-ISI Program components, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, required components in order of risk priority as identified in Section 711111.08-03 of the inspection procedure, based upon the ISI activities available for review during the on-site inspection period.

The inspectors observed the following four types of nondestructive examination (NDE) activities to evaluate compliance with the ASME Code, Section XI, and Section V requirements, and to verify that the indications and defects, (if present) were dispositioned in accordance with the ASME Code, Section XI requirements, or a NRC approved alternative (e.g., relief requests).

- Ultrasonic examination (UT) of Decay Heat weld DH-33B-CCB-6-5-SWA (elbow-to-pipe weld);
- Dye Penetrant (PT) examination of Decay Heat weld DH-33B-CCB-6-5-SWA (elbow-to-pipe weld);
- Visual (VT-3) examination of Diesel Generator support DG-JKT WTR HTXCHR-1-1-SUPPORTS; and
- Visual (VT-1) examination of Diesel Generator welded attachment DG-JKT WTR HTXCHR-1-1-AW.

The inspectors requested examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service to verify that the licensee's acceptance was in accordance with the Section XI of the ASME Code. Specifically, the inspector reviewed the following records:

- CR 06-01091 and CR 06-01151 concerning an axial indication on the Reactor Coolant Pump (RCP) drain line Nozzle to Elbow Weld. The indication was determined to be a crack of unknown depth and therefore could not be accepted to ASME XI standards and was repaired prior to plant restart.

The inspectors reviewed two ASME Section XI Code repair/replacement activities to determine if the welding and weld procedure qualification tensile tests were performed in accordance with ASME Code Sections IX and XI requirements, and those of approved relief requests. The inspectors reviewed pressure boundary welds for Class 1 Systems with the pre-emptive weld overlays of the alloy 600 welds. Specifically, the inspectors reviewed welds associated with the following work activities:

- Repair/replacement (welding) of ASME Class 1, weld overlay of safety relief nozzle to-safe-end weld (RC-PZR-WP-91-Z/W-2 ½" X/W Axis); and
- Repair/replacement (welding) of ASME Class 1, weld overlay of safety relief nozzle to-safe-end weld (RC-PZR-WP-91-Y/Z- 3" Y/Z Axis).

b. Findings

Failure to Follow Welding Procedures

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure of two contractor welders to use Weld Head Travel Speed Calculation sheets to set travel speed as required by procedure while performing weld overlay (WOL) repairs on two pressurizer safety relief valves (SRVs).

Description: On January 10, 2008, the inspectors observed contract welders performing machine gas tungsten arc welding (GTAW) on layer two of structural WOL’s on pressurizer SRVs “1” and “3” as part of the licensee’s Alloy 600 mitigation program.

The welding procedure specification (WPS) used for the pressurizer SRV WOL’s was “WPS 01-08-T-894-Top.” The WPS obtained its heat input (an essential variable) from “WPS Technique Data Sheets” attached to the WPS. The Data Sheets indicated that the travel speed (one component of heat input) was to be the speed as measured at the tungsten. In order for the speed at the tungsten to remain constant, as required to control heat input, the settings the welders input to control weld head speed for each layer were calculated on an excel spreadsheet and provided to the welders in the form of Travel Speed Calculation Sheets. The Travel Speed Calculation Sheets were required to be present and followed by the welders at the work location per Step 5.1.3 of the Weld Traveler (WSI Traveler No. 103571-TR-001).

The inspectors observed that while the welders were welding on layer two of the WOL’s for the SRVs, the Travel Speed Calculation Sheets for layer two were not at the work location. Without the Travel Speed Calculation Sheets to provide the correct travel speed, heat input to the WOL was not controlled in accordance with the procedure.

The inspectors notified the welding supervisor who immediately stopped the welding activities on the two SRVs. It was determined that the Travel Speed Calculation Sheets for either welder had not been prepared or provided to the welders. When questioned about welding without the Travel Speed Calculation Sheets generated specifically for the second layer, the welders stated that they did not realize the significance of the difference in travel speed between the first and second layer. Instead of obtaining the sheets as required by procedure, the welders took it upon themselves to calculate their own travel speeds based upon values taken from the layer one sheets.

The licensee put welding activities on the two SRVs on hold for several days until it was determined the travel speeds used by the welders resulted in a heat input which was bound by the procedure as qualified. While neither welder could recall the travel speeds they had used, conservative values were used in calculating the heat inputs. The review determined that heat inputs for the most severe differences between the required relative travel speed settings and the actual relative travel speed settings were 33 KJ/in. and 38 KJ/in., which were bound by the 32 - 39 KJ/in. heat input range qualified.

The contractor issued NCR 08-147 to address the issue. They conducted re-training for both the day and night shift personnel. The contractor also stated a CAR will be issued to address the issue at other nuclear site projects. The packages for the completed WOLs and in-progress WOL were reviewed and verified by the contractor Quality Control staff, that a similar issue did not exist on the other WOL nozzle packages. Notations will be added to the comments column of the bead logs to provide a permanent record of the variances. The licensee entered the issue into their corrective action program in CR 08-33133.

<u>Analysis</u>: The inspectors determined that the failure of two contractor welders to use Weld Head Travel Speed Calculation sheets to set travel speed as required by procedure was a performance deficiency warranting a significance determination. This finding is greater than minor because if left uncorrected it would have become a more significant safety concern in that the failure to control heat input could, among other detrimental affects, have reduced the impact toughness of the pressurizer weldment such that it would be susceptible to brittle fracture. The finding is of very low safety significance because it was subsequently determined through calculations that the resulting heat inputs were bound by the welding procedure specifications' (WPS) parameters. As a result, assuming worst case degradation, it is unlikely that there would be reactor coolant system leakage or the loss of safety function of any mitigating system.

The cause of the finding is related to the cross-cutting aspect of Human Performance, Work Practices, (Item H.4.(c) of IMC 0305) because licensee personnel failed to ensure supervisory and management oversight activities of their contractors such that nuclear safety was supported.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

The application of a WOL on the pressurizer SRV nozzles to form a new pressure boundary is an activity affecting the quality of the component's safety related function to serve as part of the reactor coolant pressure boundary.

The weld overlay procedure for the pressurizer SRV nozzle, "Welding Procedure Specification (WPS) 01-08-T-894-Top," an activity affecting quality, states that control of heat input (an essential variable) is mandatory and that all weld layers shall be completed in accordance with direction and parameters provided in the WPS Technique Data Sheets.

The WPS Technique Data Sheet stated that the travel speed shall be the number of inches per minute as measured on the tungsten and that any changes affecting heat input must be approved.

The weld overlay procedure included Traveler (WSI Traveler No. 103571-TR-001). Section 6.6 of the Traveler (WSI Traveler No. 103571-TR-001) stated that the Weld Head Travel Speed Calculation Sheets shall be utilized when welding over an area where the diameters of the underlying material changes. The Sheet was to be located at the work site per Step 5.1.3 of the Weld Traveler. This calculation is used to determine the correct travel speed settings and shall be signed by the preparer and on independent checker prior to implementation.

Contrary to the above, on January 10, 2008, while welding over an area where the diameters of the underlying material changed, the welders were not utilizing Weld Head Travel Speed Calculation Sheets as required by procedure. As a result of not following procedures, the welders did not control the heat inputs.

Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (CR 08-33133), it is being treated as a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000346/2008002-01).

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

At the end of Cycle 15, the Reactor Vessel Head's effective degradation years (EDY) was calculated as 4.48, which places the Unit in the low susceptibility ranking category. The inspectors performed the following through direct observation:

- Verified that the activities were performed in accordance with the requirements of NRC Order EA-03-009; and
- Verified that indications and defects, if detected, were dispositioned in accordance with the ASME Code or an NRC approved alternative (e.g., approved relief request).

In keeping with the Order, a visual examination (VT-2) was performed. The inspectors reviewed visual recordings of a minimum of 20 percent of the head penetrations, and confirmed visual examination quality to ensure required examination coverage.

The inspectors reviewed the NDE examination procedures and confirmed that the resolution requirements were met.

There were no examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service.

There were no welding repairs on the upper head penetrations completed since the beginning of the previous refueling outage.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From December 30, 2007, through January 16, 2008, the inspectors reviewed the BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05 “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.” The inspectors conducted direct observation of BACC visual examination activities and reviewed records to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” requirements. Specifically:

- On December 30, 2007, following shutdown, the inspectors walked down portions of the reactor coolant system (RCS) at elevated pressure and temperature. The inspectors conducted containment walkdowns after the BACC examinations to verify that there were no BA leaks that the licensee had not identified; and
- The inspectors also reviewed the visual examination procedures and examination records for the BACC examination to determine if degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed the engineering evaluations performed for the following components to ensure that ASME Code wall thickness requirements were maintained:

- CR 06-00669; BACC: RC14CB Pressurizer Level Transmitter Isolation Valve BA Deposits; Ensure the boric acid deposits are removed and an as-left inspection performed for RC14CB; and
- CR 07-16861; BACC: Boric acid on CF3A1 Found During Containment Entry; Perform an As Found BACC inspection on CF3A1 and determine if any remedial actions are necessary.

The inspectors also reviewed three boric acid leak corrective actions to determine if they were consistent with the requirements of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI.

- CR 06-00707; BACC: HP81 High Pressure Injection (HPI) Line Vent Valve Boric Acid Deposits;
- CR 06-00774; BACC: RC33 Loop 2-1 Cold Leg Drain Valve Boric Acid Deposits; and
- CR 06-02402; MU32 Minor Packing Leakage Dry White BA Residue – BACC.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

From January 9, 2008, through January 15, 2008, the inspectors performed an on-site review of SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements. The NRC inspectors observed acquisition of eddy current (ET) data, interviewed ET data analysts, and reviewed documentation related to the SG ISI program to determine if:

- In-Situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria were consistent with the Electric Power Research Institute (EPRI) TR-107620, A Steam Generator In-Situ Pressure Test Guidelines;

- the numbers and sizes of SG tube flaws/degradation identified was bound by the licensee’s previous outage Operational Assessment predictions;

- the SG tube ET examination scope and expansion criteria were sufficient to identify the degradation based onsite and industry operating experience by confirming that the ET scope completed was consistent with the licensee’s procedures, plant TS requirements and EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6";

- the licensee identified new tube degradation mechanisms;

- the SG tube ET examination scope included tube areas which represent ET challenges such as the tube sheet regions, expansion transitions, and support plates;

- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements;

- the required repair criteria are being adhered to;

- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle;

- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6;

- retrieval attempts of foreign objects were made where practicable. For those objects that were unable to be retrieved, evaluations were performed for the potential detrimental effects of the objects and appropriate repairs of the affected tubes were planned/taken; and

- the licensee identified deviations from ET data acquisition or analysis procedures.

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

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff and reviewed licensee corrective action records to determine if;

- the licensee had described the scope of the ISI/SG related problems;
- the licensee had established an appropriate threshold for identifying issues; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

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

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On March 7, 2008, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constitutes one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

c. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated the performance of the following risk significant systems:

- Condensate and condenser systems.

The inspectors reviewed events associated with the system listed above and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues 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 system performance with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

This inspection constitutes one quarterly maintenance effectiveness sample as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Orange risk drain of reactor coolant system (RCS), without an adequate vent path for ‘feed and bleed’ cooling, from filled and vented to greater than or equal to 80 inches above the RCS hot leg centerline on January 1, 2008;
- Orange risk drain of the RCS to below 80 inches of the RCS hot leg centerline on January 3, 2008, to permit installation of steam generator nozzle dams;
- Continuation of reactor core off load after discovery of weepage from the decay heat drop line during weld overlay work while the line was in service to remove core decay heat on January 4, 2008;
- Reactor vessel head move from the reactor vessel to the reactor head stand on January 5, 2008, including differences in readings between load cells;
- Orange risk during reactor coolant system fill on January 22, 2008, after removal of the steam generator nozzle dams and with a restricted vent path if needed for ‘feed and bleed’ cooling operation; and
- Reactivity plan review and use during the approach to and subsequent criticality of the new reactor core on January 31, 2008.

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

These activities constituted six samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- CR 08-32521; operability of low pressure injection flow path and availability of decay heat cooling flow path after discovery of weepage from a pipe crack in the decay heat line attached to the reactor coolant system hot leg;
- CR 08-33923; axial power shaping rod D-10 found with at least one broken male coupling tang;
- CR 08-35173; main steam line 2 remained pressurized after MS100 and MS100-1 were shut due to leakage past MS100;

- CR 08-35702; acceptability of the method specified to refill decay heat discharge piping after scheduled maintenance that required a partial drain of the low pressure injection system; and
- CR 08-36648; measured boron concentrations in the core flood tanks, the borated water storage tank, and the boric acid addition tanks do not include a provision for accounting for sample accuracy.

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

This inspection constitutes five samples as defined in Inspection Procedure 71111.15-05

b. Findings

No findings of significance were identified.

1R18 Permanent Plant Modifications (71111.18)

a. Inspection Scope

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

- ECP 06-0143-00 and ECP 06-0143-07; Alloy 600 Mitigation; Revision 0 and 1.

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and to ensure that relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The modification provided for mitigation of Alloy 600 component items and dissimilar metal welds located within the reactor coolant system. The modification included weld overlays for drained systems and for welding on the decay heat line while it was in service for removing reactor decay heat.

This inspection constitutes one sample of a permanent plant modification as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (PMT) (71111.19)

a. Inspection Scope

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

- Emergency diesel generator 2 semi-annual start and load test on January 17, 2008, (DB-SC-3077, Emergency Diesel Generator 2 184 Day Test) after maintenance on the diesel's governor controls;
- VT-2/PM test of the RCS and other components inside containment at reactor coolant system normal zero power operating temperature and normal operating pressure on January 30, 2008, (DB-PF-03010, RCS Leakage Test), after reassembly of the reactor vessel;
- Rod drop insertion time testing on January 30, 2008, (DB-SC-03270, Control Rod Assembly Insertion Time Test) after core fuel reload; and
- Time response testing valve TV1357 [Containment Air Cooler 2 Service Water Outlet Valve] on February 14, 2008, after preventive maintenance on components of the valve.

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 TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the attachment.

This inspection constitutes four samples as defined in Inspection Procedure 71111.19.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the refueling outage, conducted December 30, 2007, to February 14, 2008, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the attachment.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out of service;
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- Controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities;
- Monitoring of decay heat removal processes, systems, and components;
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- Controls over activities that could affect reactivity;
- Maintenance of secondary containment as required by TS;
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the primary containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- Licensee identification and resolution of problems related to refueling outage activities.

This inspection constitutes one refueling outage sample as defined in Inspection Procedure 71111.20-05. The sample does include the completion of NRC's Operating Experience Smart Sample (OpESS) FY2007-03, Revision 0 and Revision 1, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20." The specifics of that inspection activity were reported in Davis-Besse Nuclear Power Station NRC Integrated Inspection Report 05000346/2007005.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine 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-OP-03013, Containment Daily Inspection and Containment Closeout Inspection, on January 24, 2008;
- DB-SC-04119, RPS Channel 3 Flow Scaling Factor Determination, on January 31, 2008; and
- DB-SC-03071 Emergency Diesel Generator Train 2 Monthly Surveillance Test for DA31 Air Start Side, on March 6, 2008.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as left setpoints were within required ranges; the calibration frequency was in accordance with TS, 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 the safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the attachment.

This inspection constitutes three routine surveillance testing samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.2 In Service Testing Surveillance

a. Inspection Scope

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

- DB-SP-03219; HPI Train 2 Pump and Valve Test on February 8, 2008; and
- DB-SP-03160; Auxiliary Feed Pump Train 2 Quarterly Pump and Valve Test, on March 5, 2008.

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

This inspection constitutes two inservice inspection samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.3 Containment Isolation Valve Testing

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-PF-03008; local leakrate test of containment penetration 41 [pressurizer quench tank circulation inlet line] on January 11, 2008.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; 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; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the attachment.

This inspection constitutes one containment isolation valve inspection sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Evaluation (71114.02)

a. Inspection Scope

The inspectors reviewed and discussed with Emergency Preparedness (EP) staff and telecommunication personnel the operation, maintenance, and periodic testing of the ANS in the Davis-Besse Station’s plume pathway Emergency Planning Zone to determine whether the ANS equipment was adequately maintained and tested in accordance with Emergency Plan commitments and procedures. Additionally, the inspectors observed a siren test to evaluate procedure usage and interaction between licensee staff and county officials. Also, the inspectors reviewed records of March 2006 through January 2008 monthly trend reports and siren test failures, as well as 2006 and 2007 maintenance documents.

This inspection constitutes one sample as defined in Inspection Procedure 71114.02-05.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors reviewed and discussed with plant EP staff the emergency plan commitments and procedures that addressed the primary and alternate methods of initiating an ERO activation to augment the on-shift ERO as well as the provisions for maintaining the station's ERO call-out roster. The inspectors also reviewed reports and a sample of corrective action program records of unannounced off-hour augmentation tests, which were conducted semi-annually from February 2006 through January 2008, to determine the adequacy of the drills' critiques and associated corrective actions. The inspectors also reviewed the EP training records of a sample of approximately 34 Davis-Besse Station ERO personnel, who were assigned to key and support positions, to determine whether they were currently trained for their assigned ERO positions. Also, the inspectors conducted a walk-down of emergency response facilities to evaluate material condition and readiness of the facilities.

This inspection constitutes one sample as defined in Inspection Procedure 71114.03-05.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed a sample of Nuclear Oversight staff's 2006 and 2007 audits of the Davis-Besse Station emergency preparedness program to verify that these independent assessments met the requirements of 10 CFR 50.54(t). The inspectors also reviewed critique reports and samples of corrective action program records associated with the 2007 biennial exercise, as well as various EP drills conducted in 2006 and 2007 in order to verify that the licensee fulfilled its drill commitments and to evaluate the licensee's efforts to identify, track, and resolve concerns identified during these activities. Additionally, the inspectors reviewed and discussed with station EP staff the corrective action program, a sample of EP items, and corrective actions related to the facility's EP program and activities to determine whether corrective actions were acceptably completed.

This inspection constitutes one sample as defined in Inspection Procedure 71114.05-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's occupational exposure control cornerstone performance indicators (PIs) to determine whether or not the conditions surrounding the PIs had been evaluated, and whether the licensee entered the identified problems into the corrective action program for resolution.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys for the following five radiologically significant work activities within radiation areas, high radiation areas, and airborne radioactivity areas in the plant, and reviewed work packages, which included associated licensee controls and surveys of these areas to determine if radiological controls including surveys, postings, and barricades were acceptable:

- Replacement and change out of spent fuel pool filter;
- Steam generator platform work that included eddy current testing;
- Cutting of incores and placing incores into transfer casks;
- Pressurizer weld overlay and alloy 600 decay heat suction line overlay; and
- Scaffolding, insulation, and shielding in Containment Building.

The inspectors reviewed the RWPs and work packages used to access these five areas and other high radiation work areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to verify that they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed.

The inspectors walked down and surveyed (using an NRC survey meter) these five areas to verify that the prescribed RWP, procedure, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers were properly located.

The inspectors reviewed RWPs for the following three airborne radioactivity producing work activities to verify barrier integrity and engineering controls performance (e.g., HEPA ventilation system operation) and to determine if there was a potential for individual worker internal exposures of >50 millirem committed effective dose equivalent:

- Replacement and change out of spent fuel pool filter;
- Steam generator platform work that included eddy current testing; and
- Pressurizer weld overlay and alloy 600 decay heat suction line overlay.

Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection.

These inspections constitute four samples as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed 15 corrective action reports related to access controls and three high radiation area radiological incidents (non-PIs identified by the licensee in high radiation areas <1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of Non-Cited Violations (NCVs) tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

The inspectors evaluated the licensee's process for problem identification, characterization, prioritization, and verified that problems were entered into the corrective action program and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies.

These inspections constitute two samples as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following five jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- Replacement and change out of spent fuel pool filter;
- Steam generator platform work that included eddy current testing and equipment trouble shooting and repair;
- The cutting of incores and placing incores into transfer casks;
- Pressurizer weld overlay and alloy 600 decay heat suction line overlay; and
- Containment Building support activities including scaffolding, insulation, and shielding.

The inspectors reviewed radiological job requirements for these activities including RWP requirements and work procedure requirements, and attended ALARA job briefings.

Job performance was observed with respect to these requirements to determine whether radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors also determined if radiological controls were adequate including required radiation, contamination, and airborne surveys for system breaches; radiation protection job coverage which included audio and visual surveillance for remote coverage; and contamination controls.

These inspections constitute two samples as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate High Radiation Area (HRA) and Very High Radiation Area (VHRA) Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager and the Outage Manager during the refueling outage (RFO-15) concerning high dose rate/high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to determine whether any procedure modifications could substantially reduce the effectiveness and level of worker protection.

The inspectors discussed with RP supervisors the controls that were in place for special areas that had the potential to become very high radiation areas during certain plant operations, to determine if these plant operations required communication beforehand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

The inspectors conducted plant walkdowns to determine the adequacy of the posting and locking of entrances to high dose rate HRAs, and very high radiation areas.

These reviews constitute three samples as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements and evaluated whether workers were aware of the significant radiological conditions in their workplace, the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present.

The inspectors reviewed radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. These problems, along with planned and taken corrective actions, were discussed with the Radiation Protection Manager.

These reviews constitute two samples as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician (RPT) Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated RPT performance with respect to radiation protection work requirements and evaluated whether they were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed ten radiological problem reports which found that the cause of the event was radiation protection technician error to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

These reviews constitute two samples as defined by Inspection Procedure 71121.01.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably Achievable (ALARA) Planning And Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, site specific trends in collective exposure and source term measurements, current exposure trends, ongoing and planned activities in order to assess current performance and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to help establish resource allocations and to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed the outage work scheduled during the inspection period and associated work activity exposure estimates for the RFO-15 which were likely to result in the highest personnel collective exposures. The work activities reviewed included but were not limited to the following:

- Replacement of spent fuel pool filter;
- Steam generator platform activities;
- Cutting of incores detectors, and loading into transfer casks;
- Pressurizer weld and decay heat suction line overlays;
- Containment support activities including scaffolding, insulation, and shielding;
- Pressurizer nozzle replacement and support activities;
- Reactor head disassembly activities;
- Work on spent fuel pool and reactor refuel bridge; and
- Transfer tube cover removal.

The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and processes used to estimate and track work activity specific exposures.

These reviews constitute four samples as defined by Inspection Procedure 71121.02.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The interfaces between operations, radiation protection, maintenance planning, scheduling and engineering groups were evaluated to identify interface problems or missing program elements that could impact work planning.

The inspectors compared the person-hour estimates, provided by maintenance planning and other groups to the radiation protection group, with the actual work activity time requirements in order to evaluate the accuracy of these time estimates.

The inspectors evaluated if work activity planning included consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components/piping, job scheduling, and shielding and scaffolding installation and removal activities.

The licensee's post-job (work activity) reviews were evaluated to determine if identified problems were entered into the licensee's corrective action program.

These reviews constitute four optional samples as defined by Inspection Procedure 71121.02.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the assumptions and bases for the current annual collective exposure estimate including procedures, in order to evaluate the licensee's methodology for estimating work activity-specific exposures and the intended dose outcome. Dose rate and man-hour estimates were evaluated for reasonable accuracy.

The licensee's exposure tracking system was evaluated to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of collective exposures. RWPs were reviewed to determine if they covered too many work activities to allow work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased beyond exposure estimates.

These reviews constitute two required samples and one optional sample as defined by Inspection Procedure 71121.02.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the work activities described in Section 2OS1.4 and assessed the licensee's use of ALARA controls for these work activities including:

The licensee's use of engineering controls to achieve dose reductions and whether procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

Job sites were observed to determine if workers were utilizing the low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays.

These reviews constitute one required sample and one optional sample as defined by Inspection Procedure 71121.02.

b. <u>Findings</u>

No findings of significance were identified.

.5 <u>Source-Term Reduction and Control</u>

a. <u>Inspection Scope</u>

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms and to determine if the licensee was making allowances and had developing contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry.

The inspectors determined whether the licensee had developed an understanding of the plant source-term, that this included knowledge of input mechanisms to reduce the source term and that the licensee had a source-term control strategy in place that included a cobalt reduction strategy and shutdown ramping and operating chemistry plan which was designed to minimize the source-term external to the core. Other methods used by the licensee to control the source term including component and system decontamination, and use of shielding were evaluated.

These reviews constitute one required sample and one optional sample as defined by Inspection Procedure 71121.02.

b. <u>Findings</u>

No findings of significance were identified.

.6 <u>Radiation Worker Performance</u>

a. <u>Inspection Scope</u>

Radiation worker and RPT performance was observed during work activities being performed in radiation areas, airborne radioactivity areas, and high radiation areas that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas and that work activity controls were being complied with. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved.

These reviews constitute one sample as defined by Inspection Procedure 71121.02.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03).

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the plant USAR to identify applicable radiation monitors associated with transient high and very high radiation areas including those used in remote emergency assessment.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

The inspectors identified the types of portable radiation detection instrumentation used for job coverage of high radiation area work, other temporary area radiation monitors currently used in the plant, continuous air monitors associated with jobs with the potential for workers to receive 50 mrem committed effective dose equivalent (CEDE), whole body counters, and the types of radiation detection instruments utilized for personnel release from the radiologically controlled area.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

The inspectors reviewed calibration documentation observed instrument calibrator use and assessed alarm setpoint determinations for the following:

- J. L. Shepherd portable instrument calibrator Model 78-2M;
- Small Article Contamination Monitors (SAMs);
- Eberline Radiation Detection Device Model RM-14s;
- Main steam lines monitors RE600 and RE609;
- Containment post accident monitors RE4597AA, RE4597AB, RE4597BA, and RE4597BB;
- Station vent normal range RE4598AA, station vent accident range RE4598AB; and
- Station vent normal range RE4598BA.

The inspectors determined what actions were taken when, during calibration or source checks, an instrument was found significantly out of calibration (>50 percent), determined possible consequences of instrument use since last successful calibration or source check, and determined if the out of calibration result was entered into the corrective action program. The inspectors also reviewed the licensee’s 10 CFR Part 61 source term reviews to determine if the calibration sources used are representative of the plant source term.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports, and Special Reports that involved personnel contamination monitor alarms due to personnel internal exposures to verify that identified problems were entered into the corrective action program for resolution. All event reports involving internal exposures >50 mrem CEDE were reviewed to determine if the affected personnel were properly monitored utilizing calibrated equipment and if the data was analyzed and internal exposures properly assessed in accordance with licensee procedures.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

The inspectors reviewed corrective action program reports related to exposure significant radiological incidents that involved radiation monitoring instrument deficiencies since the last inspection in this area. Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of NCVs tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

The inspectors determined if the licensee's self-assessment activities were identifying and addressing repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

b. Findings

No findings of significance were identified.

.3 Radiation Protection Technician Instrument Use

a. Inspection Scope

The inspectors verified the calibration expiration and source response check on radiation detection instruments staged for use and observed radiation protection technicians for appropriate instrument selection and self-verification of instrument operability prior to use.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

b. Findings

No findings of significance were identified.

.4 Self-Contained Breathing Apparatus (SCBA) Maintenance and User Training

a. Inspection Scope

The inspectors reviewed the status and surveillance records of SCBAs staged and ready for use in the plant and inspected the licensee's capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions. The inspectors determined if control room operators and other emergency response and radiation protection personnel were trained and qualified in the use of SCBAs (including personal bottle change-out). The inspectors verified that five individuals on each control room shift crew, and radiation protection fire and operation individuals from each designated department were currently assigned emergency duties (e.g., onsite search and rescue duties).

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

The inspectors reviewed the qualification documentation for at least 50 percent of the onsite personnel designated to perform maintenance on the vendor-designated vital components, and the vital component maintenance records over the past five years for three SCBA units currently designated as "ready for service." The inspectors also ensured that the required, periodic air cylinder hydrostatic testing was documented and up to date, and that the Department of Transportation (DOT) required retest air cylinder markings were in place for these three units. The inspectors reviewed the onsite maintenance procedures governing vital component work including those for the low-pressure alarm and pressure-demand air regulator and licensee procedures and the SCBA manufacturer's recommended practices to determine if there were inconsistencies between them.

This inspection constitutes one sample as defined by Inspection Procedure 71121.03-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the fourth quarter 2007 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator for the period from the first quarter of 2007 through the fourth quarter of 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection reports for the period of the first quarter of 2007 through the fourth quarter of 2007 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one unplanned scrams per 7000 critical hours sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System Leakage performance indicator for the period from the first quarter of 2007 through the fourth quarter of 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator

Guideline," were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection reports for the period of the first quarter of 2007 through the fourth quarter of 2007 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one reactor coolant system leakage sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.4 Emergency Preparedness Strategic Areas

a. Inspection Scope

The inspectors reviewed the licensee's records associated with the three EP performance indicators (PIs) listed below. The inspectors verified that the licensee accurately reported these indicators in accordance with relevant procedures and Nuclear Energy Institute guidance endorsed by NRC. Specifically, the inspectors reviewed licensee records associated with PI data reported to the NRC for the period April 2007 through December 2007. Reviewed records and processes discussed included: procedural guidance on assessing opportunities for the three PIs; assessments of PI opportunities during predesignated Control Room Simulator training sessions, the 2007 biennial exercise, and other drills; revisions of the roster of personnel assigned to key ERO positions; and results of periodic ANS operability tests. The following PIs were reviewed:

- Alert and Notification System;
- ERO Drill Participation; and
- Drill and Exercise Performance.

This inspection constitutes one alert and notification system sample, one ERO drill participation sample, and one drill/exercise performance sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.5 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's PI submittals for the periods listed below. The inspectors used PI definitions and guidance contained in Revision 5 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to determine if the PI data were accurate. The following PI was reviewed:

- Occupational Exposure Control Effectiveness

The inspectors reviewed the licensee's determination of the Performance Indicator (PI) for the occupational radiation safety cornerstone to determine if the licensee accurately assessed the performance indicator and had identified all occurrences. Specifically, the inspectors reviewed the licensee's condition reports (CRs) for 2007/2008 and associated occupational exposure performance indicator data to ensure that there were no PI occurrences that were not identified by the licensee. Additionally, as part of plant walkdowns, the inspectors selectively examined the adequacy of posting and controls for locked HRAs. The inspectors interviewed members of the licensee's staff who were responsible for performance indicator data acquisition, verification and reporting, to determine if their review and assessment of the data was adequate.

This review constitutes one sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance 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 corrective action program (CAP) at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached list of documents reviewed.

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

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

.3 Annual Sample: Review of CR-07-32112; Pressurizer Level Decrease While Placing Decay Heat Train 1 In Standby

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting an unexpected lowering of the pressurizer level on December 30, 2007, when decay heat train 1 was aligned for decay heat cooling of the reactor coolant system. The pressurizer level decreased approximately 6 inches which was indicative of an air void in the decay heat system of about 17 cubic feet. The inspectors reviewed the licensee's investigation into the cause of the event, the thoroughness of the investigation, and the appropriateness of any corrective or preventive action items. The event and licensee's investigations were documented in CR 07-32112.

The above constitutes completion of one in-depth problem identification and resolution sample.

b. Findings

Introduction: A Green self-revealing NCV was identified for failure to properly vent a portion of decay heat/low pressure injection train 1 after maintenance in accordance with TS 4.5.2b., which resulted in a void in the pump discharge piping of the train 1 decay heat/low pressure injection system for approximately 59 days of plant operation.

Description: On December 30, 2007, with the plant in Mode 5, a step decrease of approximately 6 inches in the reactor coolant system (RCS) pressurizer water level was observed when the decay heat (DH) removal train 1 pump suction was being realigned from the low pressure injection mode to the DH removal mode. The re-alignment was suspended and an investigation commenced to determine if there was an air void in the DH system that caused an outflow of RCS water to the DH system. A void was discovered in the discharge piping of DH pump 1. The void was vented using a high point vent valve inside containment and DH train 1 alignment to DH removal mode was completed.

The licensee documented and investigated the issue via CR 07-32112. The licensee's investigation concluded that a void was created in the pump discharge piping during maintenance on train 1 during the period of October 29, 2007, through October 31, 2007. A portion of the train was drained during that period of time for scheduled maintenance. The system was refilled and vented using the existing system operating procedure. However, the train restoration instructions and the system operating procedure did not require venting from the high point of the discharge piping that had been drained. The licensee concluded that a void of approximately 15 cubic feet was left in the discharge piping that had been isolated. The air void then grew to about 17 cubic feet when a discharge valve (DH1B), used as a work isolation point, was opened and the void moved to a higher portion of the piping. This void existed undiscovered from October 31, 2007, until December 30, 2007, when the train was being aligned to the DH removal mode. This alignment activity permitted RCS inventory to flow to and through the pump to the voided discharge piping and to compress the void and potentially sweep part of the void into the RCS. The RCS, at the time of the re-alignment, was at a pressure of approximately 200 psig; prior to the alignment the DH system train 1 pressure was due to the head of water in the borated water storage tank with the maximum DH train 1 system pressure being less than about 35 psig.

In addition to the post-event activity of venting the system from the high point vent inside containment, the licensee reviewed the affect of the void on the ability of the DH/low pressure injection system train 1 to perform its function in the event of a condition requiring its use. The licensee's evaluation concluded that the DH/low pressure injection train 1 would have been able to perform its safety function, and the system would not have exceeded any design stress limits if low pressure injection was actuated. Initial water delivery to the reactor vessel times would have been slightly increased but, per the licensee's evaluation, would have been within times assumed in design calculations.

The licensee's investigation stated that the root cause of the event was inadequate procedural guidance for refilling the DH system after draining portions of the system downstream the of the DH coolers. During the October 2007 maintenance, piping was drained downstream of the DH coolers. The investigation, among other items and causes, also listed that there was no specific requirement to review isometric drawings when modifying venting procedures and that a mindset existed that in-place operating procedures were adequate for post maintenance recovery. The investigation further mentioned that there were incorrect assumptions by personnel about what had been reviewed by others, and that there was less than adequate use of human performance tools in reviewing procedure revisions.

The licensee's immediate corrective action restored the DH/low pressure injection system to required operability. The licensee also developed corrective and preventive actions to preclude recurrence of the event and minimize the potential for similar events. Inspectors concluded that the intended actions appeared to address the root and contributing causes. A significant portion of the actions were scheduled for completion after the end of the inspection period.

<u>Analysis</u>: The performance deficiency with this event is that the licensee did not adequately review the scope of the work and piping elevations to properly plan for restoration of the DH system after maintenance that required draining of the system. The finding is greater than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone in that permitting an air void

in the train's discharge piping affected the reliability and capability of the system. The finding is of very low safety significance because it did not result in a loss of function per Part 9900, "Technical Guidance – Operability Determinations and Functionality Assessments," did not represent an actual loss of safety function, and is not potentially risk significant due to external events. The finding is associated with the cross-cutting area of human performance in that the resources and specifically work packages and procedures were not adequate to ensure that the train 1 DH/Low pressure injection system was restored to a filled and vented system condition (H.2(c)) after completion of maintenance activities.

Enforcement: TS 4.5.2b requires that prior to operation after emergency core cooling system (ECCS) piping has been drained, the ECCS piping is to be verified full of water by venting the ECCS pump casings and discharge high points. Contrary to the above, the licensee failed to verify that the DH/low pressure injection train 1 (an ECCS system) discharge piping was full of water by venting from the high point vents prior to returning the system to an operational status. The system was declared operable even though it was partially drained and remained in that condition for approximately 59 days. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as CR 07-32112, this violation (NCV 05000346/2008002-02) is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.

.4 Annual Sample: Review of CREVS Issue

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized three condition reports documenting recent issues with the Control Room Emergency Ventilation System (CREVS). CR 07-29963 "CREVS Train 2 Compressor Tripped during DB-SS-03042 Monthly Test" and CR 08-34173 "CREVS Train 1 Low Refrigerant Charge" described two maintenance preventable functional failures that occurred during testing. A more recent CREVS issue was documented in CR 08-36684 "CREVS 1 Compressor Trip during Monthly Test DB-SS-03041." The inspectors evaluated the completeness and accuracy of identification of the problem, the extent of condition, classification and resolution of the issue commensurate with its safety significance, the identification of the causes of the problem, and the appropriateness of the licensee's actions to address the problem. Additionally, the inspectors reviewed the causal analysis and related corrective action assignment, maintenance rule evaluation, and issues the system has experienced since the initial documented failures.

The above constitutes completion of one in-depth problem identification and resolution sample.

b. Assessments and Observations

The inspectors' identified several items that were not clear from the available documentation and required inspector followup with licensee personnel. On November 9, 2007, the CREVS train 2 compressor tripped on low oil pressure (CR 07-29963). The apparent cause was determined to be "Mechanical Failure - Other" because the failure analysis of the compressor performed by the vendor yielded inconclusive results. This CR also originally listed a secondary apparent cause of incorrect diagnosis of two earlier incidents of CREVS low oil pressure compressor failures. However, this secondary apparent cause did not have any defined corrective actions within the CR. On March 17, 2008, the licensee's Corrective Action Review Board (CARB) directed that this secondary cause be removed from the CR because it had no associated corrective action assigned to it. Licensee procedures required that all apparent causes have a corrective action assigned to them. The CARB members noted that having a secondary apparent cause was not required. However, the inspectors noted that this meeting occurred 5 days after another compressor trip on low oil pressure occurred on Train 1.

On January 24, 2008, the CREVS train 1 compressor tripped during a monthly Safety Feature Actuation System (SFAS) test (CR 08-34173). The apparent cause of the trip was low refrigerant charge. The low refrigerant charge was attributed to a leak or leaks that occurred during scheduled maintenance that involved disconnecting and reconnecting a line that contained refrigerant gas. The line was disconnected for about 10 days and during this time CREVS train 1 was considered inoperable. The licensee's work package for this work's restoration only required a visual inspection of the line as a post maintenance test and did not specify any specific checks for leakage or for compressor operability. It was also subsequently determined that the work had been performed by personnel not qualified to perform the work under the licensee's programs (CR 08-37010). One licensee proposed corrective action was revision of their Post Maintenance Test Manual to include leakage verification after work on the refrigerant line connection involved in this work and similar connections.

On January 28, 2008, maintenance crews performed work to repair the major refrigerant leaks on CREVS train 1 following the compressor trip on January 24, 2008. To make up the lost refrigerant, the maintenance crew recharged the compressor from a cylinder of refrigerant. However, the CREVS train 1 monthly test was run for the purpose of demonstrating operability with the charging cylinder still connected and not insolated from the system. Licensee's review of the issue provided recommendations that existing procedures be changed to ensure that any charging cylinder is isolated or disconnected from the system prior to recording operating parameters for operability considerations. This work and the compressor trip occurred during a plant Mode that did not require CREVS operability.

On March 12, 2008, the CREVS train 1 compressor tripped on low oil pressure during its monthly test making it inoperable and unavailable (CR 08-36684). Workers were able to restart the compressor with a heat load and completed the monthly test but did treat the test as a failed test. The licensee determined that the failure was caused by low oil level as well as the initial lack of a heat load on the system. Licensee personnel stated that had there been an actual need for the CREVS system with the accompanying heat load, they believed that the unit would have not tripped on low oil pressure. Licensee personnel did add oil to the system. The low oil level was initially attributed to the refrigerant leaks that occurred during previous work activities and that compressor oil level was not required to be regularly monitored. It was also identified that maintenance personnel had a practice of recharging for lost refrigerant while not accounting for any oil lost with the refrigerant.

The inspectors also noted that the licensee documented at least two instances where small lines associated with the compressor packages were damaged or improperly connected during maintenance activities. Those lines did contain refrigerant and in both instances there was some refrigerant loss.

c. Conclusions

The inspectors did not identify any findings of significance. Many of the identified issues occurred in a plant mode that did not require CREVS operability. However, the inspectors noted several instances of workers' action contributing to degraded equipment performance. The licensee placed the system into a maintenance rule a(1) status and initiated the process of developing an associated action plan. Identified issues were entered into their corrective action program and corrective actions were either developed or were scheduled to be developed. The inspectors concluded that the developed plans to improve CREVS performance and CREVS maintenance processes appeared reasonable. However, there were corrective actions that were still open and plans not yet developed at the end of the inspection period.

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 Main Turbine Generator Vibrations Sufficient to Preclude Bringing the Equipment to Operating Speed

a. Inspection Scope

During the plant startup from its refueling outage the plant's main turbine generator experienced vibration levels that prohibited bringing the equipment beyond approximately 900 rpm. The inspectors reviewed the plant's response and specifically the operators' response to the issue. This included the actions taken to protect the equipment during startup and the actions taken to diagnose the issue and resolve the issue. The inspectors also looked at the appropriateness of balancing the generator and the appropriateness of the licensee's decision to replace all four components of the amortisseur winding that was determined to have different material in one of the four segments of the winding. Documents reviewed in this inspection are listed in the Attachment.

This inspection constitutes one sample as defined in Inspection Procedure 71153-05.

b. Findings

No findings of significance were identified.

.2 Increased Reactor Coolant Radioisotope Concentrations

a. Inspection Scope

After the plant startup from its refueling outage, the reactor coolant system monitored radionuclide concentrations quickly rose to levels higher than was consistent with previous operating cycles. Specifically radioactive iodine and xenon levels rose sufficiently to cause the licensee to enter into their procedures for monitoring for fuel failures. The licensee believed that they experienced a “tight crack” failure in a fuel pin in a new or a once-burned fuel element. The licensee imposed additional limits on normal maneuvering power change rates and continued to monitor coolant system radionuclide concentrations. The inspectors reviewed the licensee's response to the identified conditions and the licensee's adherence to applicable procedures. Documents reviewed in this inspection are listed in the Attachment.

This inspection constitutes one sample as defined in Inspection Procedure 71153-05.

b. Findings

No findings of significance were identified.

.3 Tube Rupture in High Pressure Feedwater Heater

a. Inspection Scope

On March 7, 2008, the licensee observed that power has increased unexpectedly to a level just above 100 percent. The licensee reduced power and identified an apparent feedwater tube leak in a high pressure feedwater heater in train 1. The inspectors reviewed the licensee response to the transient and the subsequent repairs and testing of the feedwater train. The inspectors also reviewed the cause of a March 10, 2008, unexpected test-induced condenser vacuum increase and a consequential power increase. Documents reviewed in this inspection are listed in the Attachment.

This inspection constitutes one sample as defined in Inspection Procedure 71153-05.

b. Findings

Introduction: A Green, self-revealing finding that did not result in a violation was identified because the licensee failed to retain valve configuration control during a leak test of the high pressure feedwater heater 5 in train 1 of the main feedwater system. Because moisture separator reheater drain valve RD198 was left open during the leak test, station air used to pressurize the shell side of the feedwater heater was introduced into the condenser which ultimately resulted in an unplanned and unexpected reactivity event.

Description: On March 10, 2008, during repairs of a high pressure feedwater heater 1-5 tube rupture (CR08-36528), operators were performing a leak test of the extraction steam side, or shell side, of feedwater heater 1-5. The plant was at 94.5 percent power and the low and high pressure condenser pressures were 1.413 and 2.572 inches Hg(a), respectively.

The operators had filled the shell side of heater 1-5 with demineralized water and then added station air to it to pressurize it. According to the control room log, at 0216 hours, the control room "received a phone call from Shift Manager asking for the condenser pressure value and trend . . .." The Command Senior Reactor Operator observed that condenser pressure had risen to almost 4 inches Hg(a) and was rising rapidly. The Balance-of-Plant Reactor Operator identified that a non-licensed operator was adding air to feedwater heater 1-5. The valve allowing air into the feedwater heater was subsequently closed. This stopped the flow of air into the condenser which allowed the mechanical hogger to recover vacuum in the condensers. The low and high pressure condenser pressures peaked at 4.527 and 4.766 inches Hg(a), respectively. Because the pressure in the condensers had been increasing, the integrated control system raised the power of the reactor to overcome the decreased electrical generation efficiency and maintain electrical output to the grid. After the condenser pressures peaked, the core power peaked to 97.279 percent, about 2.7 percent greater than the power expected. When appropriate vacuum in the condensers was achieved, reactor power lowered to its previous power without operator action. It was determined that reheater drain valve RD198 was the valve that was not closed that provided a path for air from feedwater heater 1-5 to the high pressure condenser, and consequently to the low pressure condenser.

The inspectors determined that the momentary increase in reactor power did not violate any technical specification limits. This issue was entered into FENOC's corrective action program (CRs 08-36573 and 08-36574). The licensee preliminarily determined that the cause of the performance deficiency was not properly scoping and preparing for the repair and test along with not recognizing the position of valve RD198 and its potential impact on the testing activity.

Analysis: The performance deficiency associated with this finding was that the operators did not maintain valve configuration control and account for all flowpaths prior to the pressure test. The finding is greater than minor since it is associated with the configuration control-operating equipment lineup attribute of the Initiating Events Cornerstone and because it affects the associated Cornerstone Objective to limit the likelihood of those events that upset plant stability during power operations. Specifically, the operators did not close reheater drain valve RD198 to prevent a flowpath from feedwater heater 1-5 to the high pressure condenser. This ultimately resulted in a degradation of vacuum in the high pressure and low pressure condenser and an unplanned and unexpected reactivity event. The inspectors reviewed IMC 0609, Appendix A, "Significance Determination of Reactor Findings for At-Power Situations," dated January 10, 2008, and conducted an SDP Phase 1 screening. The inspectors determined that the finding is of very low safety significance (Green) since it does not contribute to the likelihood of a primary or secondary system loss of coolant accident, does not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available, and does not increase the likelihood of a fire or internal/external flood.

The finding was associated with the cross-cutting area of human performance in that work control and specifically the coordination of work activities did not properly record or assess the status of a valve in the test boundary and created a condition that had an operational impact (H.3(b)).

Enforcement: No violation of NRC regulatory requirements occurred. (FIN 05000346/2008002-03) The licensee entered this finding into its corrective action program under CRs 08-36573 and 08-36574.

4OA5 Other Activities

.1 World Association of Nuclear Operators (WANO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the WANO plant assessment conducted in August 2007. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

.2 Licensee Activities and Meetings

The inspectors observed select portions of licensee activities and meetings and met with licensee personnel to discuss various topics. The activities that were sampled included:

- Davis-Besse Daily Shift Outage Management Team meetings several days each week during the refueling outage that occurred from the beginning of the inspection period through February 14, 2008;
- Several days each week of the licensee's daily shift turnover meetings and management oversight meetings;
- The FENOC/NRC Senior Management Briefing combined meeting, on January 9, 2008;
- Outage Restart Readiness Process Final Meeting on January 24, 2008;
- Monthly Performance Review meeting on February 22, 2008;
- Corrective Action Review Board Meeting on March 4, 2008, and March 17, 2008;
- Exit Meeting of the Corporate Nuclear Review Board on March 14, 2008;
- Davis-Besse supervisor briefing on March 31, 2008, which involved discussion of site and fleet radiation protection activities and goals;
- Davis-Besse All Hands Meeting on March 28, 2008;
- Corrective Action Review Board Meeting on March 17, 2008, which discussed Control Room Emergency Ventilation System issues; and
- Plant Health Committee Meeting on March 26, 2008.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Dissimilar Metal Butt Welds (TI 2515/172, Revision 0)

a. Inspection Scope

The inspectors conducted a review of the licensee's activities regarding licensee dissimilar metal butt weld (DMBW) mitigation and inspection implemented in accordance with the industry self-imposed mandatory requirements of Materials Reliability Program (MRP) - 139, "Primary System Piping Butt Weld Inspection, and Evaluation Guidelines." To support the evaluation of the licensees' implementation of MRP - 139, TI 2515/172, the "Reactor Coolant System Dissimilar Metal Butt Welds," was issued February 21, 2008.

From January 7, 2008, through January 17, 2009, the inspectors performed a review in accordance with a draft version of TI-172 of licensee procedures, equipment, and personnel used for installation of full structural weld overlays on Alloy 82/182 components equal to, or greater than hot leg temperatures (TI Section 03.03). The scope of this inspection did not include any of the programmatic reviews of the in-service inspection program, (TI Sections 03.01 and 03.05) or weld ultrasonic examinations (TI Section 03.02). These will be inspected later. TI Section 03.04 for mechanical stress improvement activities was not done since this activity was not planned or conducted by the licensee.

This review included:

- Observation and interviews with contractor personnel, equipment conducting automated welding of 2 weld overlays on the pressurizer;
- Welding procedures and specifications implemented for full structural overlays;
- Interviews with contractor personnel responsible for oversight of the implementation of the weld overlays;
- Welder qualification records; and
- Contractor and licensee corrective actions implemented for the weld overlays.

The documents reviewed by the inspector for this inspection are listed in the Attachment to this report.

b. Observations

Summary: Davis-Besse committed to fulfill the examination or mitigation schedule contained in MRP-139 for the applicable pressurizer welds by the end of 2007. To meet these requirements, Davis-Besse scheduled installation of full structural overlays on all welds containing 82/182 material in the scope of MRP-139 operating at or exceeding hot leg temperatures during the refueling outage starting on December 31, 2007.

No ultrasonic examinations were planned to be conducted on these welds prior to the weld overlays. However, due to a through wall leak that resulted during one of the overlays, an ultrasonic examination was conducted on that weld without the overlay to characterize the flaw. Based on the flaw characterization this weld was then overlayed as planned.

Overall, the inspectors concluded that the licensee completed the installation and fabrication of the full structural overlays consistent with the requirements of MRP-139 and associated relief requests. The inspectors documented conclusions in response to specific questions related to commitments for design, fabrication and installation of the weld overlays in c. below.

In accordance with requirements of TI 2515/172, Revision 0, the inspectors evaluated and answered the following questions:

a. For MRP-139 baseline inspections…

   This portion of the TI was not inspected during the period of this report. This will be inspected later.

b. For each examination inspected, was the activity…

   This inspection did not include review of the in-service or pre-service ultrasonic examinations conducted on welds or weld overlays. This will be inspected later.

c. For each weld overlay inspected, was the activity:

1. Performed in accordance with ASME Code welding requirements and consistent with NRC staff relief request authorizations?

   - No. As stipulated in the applicable relief request, the welding process was to be controlled by the requirements of the welding specification and procedure (ASME Section IX). During observation of the welding process on two of the overlays, the inspectors identified the welders were not complying with the weld procedure specifications relevant to heat input. This violation was determined to be a green finding and is discussed in Section 1R08 of this report. By engineering calculation, the licensee determined the overlays had not exceeded the fabrication requirements contained in the approved relief requests.

2. Has the licensee submitted a relief request and obtained NRR staff authorization to install the weld overlays?

   - Yes. The inspectors found the licensee had obtained authorization to install weld overlays based on relief requests RR-A30 and RR-A31 which were approved by Safety Evaluations (SER's) issued on December 20, 2007, and December 14, 2007, respectively. The weld overlays were performed to meet the requirements contained in the relief requests for design, fabrication, pressure testing, and examination.

3. Performed by qualified personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   - Yes, the welders were qualified in accordance with ASME Section IX criteria documented on Welder Performer Qualification reports.

4. Performed such that deficiencies were identified, dispositioned, and resolved?

   - No. As discussed in Question 1 above, during observation of two in-process weld overlays, the inspector identified a performance deficiency involving control of travel speed. A cross cutting aspect of the resultant finding concerned a lack of oversight of the contractor programs.

   - Once identified by the inspector, the inspectors determined the contractor and licensee adequately dispositioned and resolved the issue.

d. For each mechanical stress improvement used by the licensee during the outage, was the activity performed in accordance with a documented qualification report for stress improvement processes and in accordance with demonstrated procedures? Specifically…

   Not applicable; there were no mechanical stress improvement activities performed or planned by this licensee to comply with their MRP-139 commitments.

e. For the inservice inspection program…

This portion of the TI was not inspected during the period of this report. This will be inspected later.

c. Findings

One finding of significance was identified and is documented in section 1R08 of this report.

.4 Evaluation of the Independent Engineering Assessment Report

a. Inspection Scope

As part of the inspection activities performed to verify the licensee's compliance with the requirements for independent assessments, as described in the March 8, 2004, Confirmatory Order Modifying License No. NPF-3, the inspectors reviewed the Confirmatory Order required Independent Assessment of Engineering Programs Effectiveness at the Davis-Besse Nuclear Power Station, final report, submitted to the NRC on November 8, 2007. As part of the Order related inspection activities, the inspectors reviewed the report to ensure that it provided an overall assessment of Engineering performance and that the Team's inspection activities supported the report's conclusions.

b. Observations and Findings

The fourth annual Davis-Besse Independent Engineering Assessment required by the Order was performed during the time period of September 10, 2007, to September 21, 2007. The NRC inspectors reviewed and documented their evaluation of the Independent Assessment Plan in inspection report 05000346/2007003, and implementation of the assessment in inspection report 05000346/2007004. During the time period that the assessment team was on site, the NRC inspectors observed many of the assessment activities in progress. On November 8, 2007 the licensee submitted the Independent Assessment of the Engineering Programs Effectiveness at the Davis-Besse Nuclear Power Station, final report to the NRC. This report documented the findings of that assessment.

- Overall, the team rated the modification process as effective. This was based on the quality of engineering change packages, interviews with engineers and managers, and Engineering Assessment Board performance indicator trends. Further improvement in the reduction of the backlog of open modifications was noted. The milestone for the issuance of all modifications one year prior to the fifteenth refueling outage was met;
- The team rated the calculation area as Effective based on the quality of work products reviewed and the continuing progress being made. Backlogs have been lowered near target levels in plant engineering and technical services engineering, and continue to improve in design engineering. Calculations reviewed by the team met the station's standards and expectations for technical rigor;
- System Engineering Programs and Practices were rated by the Team as effective. System Engineering was found to be responsive to plant problems and

supportive to operations and maintenance. Maintenance Rule systems overall health was Green at the time of the assessment. This was improved from that at the time of the 2006 assessment. At the time of the 2006 assessment, eight systems were designated as in red health condition. At the time of the 2007 assessment, two systems remained in the red system health designation status. This was attributable to completion of significant system health improvement related work as well as a change in the calculation of individual and overall system health. The change in the overall calculation of system health was stimulated by information obtained through benchmarking and discussions with industry groups;

- Corrective Action Program Implementation was rated by the Team as effective overall. Progress continued to be made on corrective action backlogs. Engineering's implementation of the Corrective Action Program was found to be very good to excellent. Condition Reports were found to be promptly initiated as appropriate. Some weakness was identified by the Team in their review of a Limited Apparent Cause Evaluation and in the review of a Root Cause Evaluation; and
- Overall, the team rated the self-assessment process as effective. This is based on the quality of self-assessments, interviews with engineers and managers, and the receptivity and responsiveness management exhibits toward the self-assessment process.

The Team reviewed engineering products in a number of areas and did not identify any discrepancies that were considered significant in terms of the validity of the work product, or indicative of a systematic deficiency in engineering work performance or management. The Team identified five "areas in need of attention." An area in need of attention was defined as an identified performance, program, or process element within an area of assessment that, although sufficient to meet its basic intent, management attention is required to achieve full effectiveness and consistency. These "areas in need of attention" are not required to be addressed by formal Action Plans submitted to the NRC, but are considered for entry into the Corrective Action Program by the licensee. The Team also reviewed the licensee's response to areas in need of attention identified during the 2006 independent assessment.

Overall the effectiveness of engineering programs was rated as effective. The Team noted the following observations:

- Davis-Besse (DB) Engineering Programs continue to be effective in both technical and organizational aspects;
- Improvements have been noted over the four annual assessments. DB Engineering has addressed problems, often self-identified, and has improved performance in technical quality of engineering work products and in throughput of engineering work processes; and
- The nature of work observed has transitioned from post-restart backlog reduction and steady state plant support, to predominantly steady state plant support. The resources made available from reduction of backlog work have been utilized to reduce the amount of engineering work contracted outside the Company and to increase the effort for improvement items.

c. Conclusions

The inspectors determined that the Team's inspection activities were in accordance with the Inspection Plan and were of sufficient depth and scope to develop an adequate assessment of Engineering performance.

.5 In-Process Observation of Corrective Actions Associated with the NRC's August 15, 2007, Confirmatory Order

a. Inspection Scope

By letter dated August 15, 2007, the NRC issued an immediately effective Confirmatory Order EA-07-199 (Order) that formalized commitments made by the FirstEnergy Nuclear Operating Company (FENOC). FirstEnergy Nuclear Operating Company's commitments were documented in its July 16, 2007, letter responding to the NRC's May 14, 2007, Demand for Information (DFI).

In addition to implementing interim corrective actions, the Order required in part that the licensee (refer to IR05000346/2007005 for a list of all required commitments):

- Conduct regulatory sensitivity training for selected FENOC and non-FENOC First Energy employees to ensure those employees identified and communicate information that has the potential for regulatory impact either at FENOC sites or within the nuclear industry to the NRC. The licensee was to provide the population to be trained, the training methodology and materials, and the training objective at least 30 days prior to conducting the training. All training was to be conducted by November 30, 2007. This requirement was inspected and documented in IR 05000346/2007005; and
- Conduct effectiveness review to determine if an appropriate level of regulatory sensitivity was evident among First Energy employees including those who received regulatory sensitivity training in January 2008 and 2009.

To assess the licensee's activities associated with effectiveness reviews, the inspectors observed the independent assessment team's activities during the week of January 15, 2008, at FirstEnergy Headquarters in Akron, Ohio. The observations included review of the standard questions being asked of FirstEnergy individuals, observations of the team members conducting interviews, and observation of the team's internal meetings assessing the results from the interviews.

b. Observations and Findings

Based on the documentation reviews and observations, the inspectors concluded that:

- The effectiveness review was conducted by an independent team of qualified individuals. The team was comprised of three experienced individuals: an independent contractor, a manager from a non-FENOC nuclear facility, and an individual from Nuclear Energy Institute (NEI). The team conducted approximately 100 interviews covering FirstEnergy Nuclear Operating Company individuals at Davis-Besse, Perry, and Beaver Valley and individuals from FirstEnergy and FirstEnergy Nuclear Operating Company in Akron, Ohio. The interview sample included individuals who had received the sensitivity training and individuals whom had not. The objective of interviewing individuals who had not received the training was to determine the level to which the training subject matter had filtered down through the organization;

- The inspectors determined that the questions asked of each FirstEnergy/FirstEnergy Nuclear Operating Company individual were appropriate and designed to elicit the interviewee's knowledge and understanding of the material presented during the sensitivity training. The inspector also determined that the interviews were conducted in a manner that allowed the interviewees to express their understanding of the subject matter and to provide examples of how the information affected their daily activities. The interviews were also designed to assess the level to which individuals understood the concepts discussed in the training, for example, safety conscious work environment; and

- Based on the reviews, observations, and interview responses from individuals, the inspectors concluded that the licensee had met the Order required effectiveness review in 2008 to determine if an appropriate level of regulatory sensitivity was evident among FirstEnergy employees. Further, the inspectors concluded that the training provided had been effective in increasing the sensitivity of the organization as a whole.

.6 <u>Quarterly Resident Inspector Observations of Security Personnel and Activities</u>

a. <u>Inspection Scope</u>

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

- Multiple tours of operations within the Central and Secondary Security Alarm Stations;
- Tours of selected security towers/security officer response posts;
- Direct observation of personnel entry screening operations within the plant's Main Access Facility; and
- Security force shift turnover activities.

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

b. <u>Findings</u>

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 14, 2008, the inspectors presented the inspection results to Mr. B. Allen and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meeting

Interim exits were conducted for:

- Access control to radiologically significant areas and the ALARA planning and controls program with Mr. Vito A. Kaminskas, Director of Site Operation on January 11, 2008;
- Inservice Inspection Activities and a portion of Reactor Coolant System Dissimilar Metal Butt Welds (TI 2515/172,) with Mr. V. Kaminskas on January 17, 2008;
- Emergency Preparedness inspection with Mr. C. Price on February 29, 2008; and
- Radiation monitoring instrumentation protective equipment and the close-out of the licensee identified violations were discussed with Mr. Vito A. Kaminskas, Director Site Operations on March 7, 2008.

The ISI inspection activities involved reviewing some proprietary material. The inspectors returned proprietary information reviewed during the inspection prior to leaving the site. All inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following two violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Occupational Radiation Safety

- Technical Specification 6.12.2.a.2. requires a high radiation area with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or from any surface penetrated by the radiation, to be conspicuously posted as a high radiation area and be provided with a locked door, gate, or other barrier that prevents unauthorized entry. In addition, these doors, gates, or other barriers shall remain locked except during periods of personnel or equipment entry or exit. Contrary to this requirement, on February 23, 2008, a chain used at the containment airlock to lock both hand-wheels together, preventing either the inner or outer doors of the airlock from opening, was not locked in a manner that would prevent opening of the door. This issue was entered into the licensee’s Corrective Action Program as CR 08-35812 and radiation protection staff secured the hand-wheels to the airlock door with a more robust locking steel clamp. The issue is of very low safety significance because it did not involve As-Low-As-Reasonably-Available (ALARA) planning or work controls, an overexposure, substantial potential for overexposure, or limit the ability to assess radiation dose.

- Technical Specification 3.7.6.1 states "control room emergency ventilation system requires two independent control room emergency ventilation systems (monitors) be operable while plant is operating in modes 1, 2, 3, and 4. With both channels of station vent normal range radiation monitoring instrumentation inoperable, TS LCO 3.7.6.1 Action C requires that within one hour the control room normal ventilation system be isolated and at least one control room emergency ventilation train placed in operation. Contrary to this requirement, on October 16 and 22, 2007, both trains of Station Vent Radiation Monitors were inoperable for more that one hour without entering into TS 3.7.6.1 Action C. This issue was entered into the licensee's Corrective Action Program as CR 07-29410. The issue is of very low safety significance because it did not involve ALARA planning or work controls, an overexposure, substantial potential for overexposure, or the ability to assess radiation dose.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Bezilla, Site Vice President
B. Boles, Director, Maintenance
J. Grabnar, Director, Engineering
L. Harder, Radiation Protection Manager
P. Hoppe, ALARA Supervisor
R. Hovland, Manager, Technical Services
R. Hruby, Regulatory Compliance Manager
V. Kaminskas, Director, Plant Operation
S. Plymale, Manager, Plant Engineering
C. Price, Director of Performance Improvement
J. Powers, Director, Fleet Engineering
J. Rinckel, Vice-President, Fleet Oversight
R. Hruby, Regulatory Compliance Manager
C. Steagall, Nuclear Oversight Manager
S. Trickett, Superintendent of Radiation Protection Support
J. Vetter, Performance Improvement and Emergency Response Manager
D. Wuokko, Acting Manager, Regulatory Compliance

Ohio Emergency Management Agency

M. White, Resident Radiological Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000346/2008002-01	NCV	Failure to follow welding procedures
05000346/2008002-02	NCV	Air Void in Decay Heat/Low Pressure Injection System Due to Inadequate Venting After Maintenance
05000346/2008002-03	FIN	Unexpected Reactivity Excursion Due to Unidentified Valve Position During Post Repair Air Pressure Testing

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

Procedures:
- RA-EP-02830; Emergency Planning Off-Normal Occurrence Procedure: Flooding; Revision 2

Other:
-USAR Section 2.4; Hydrology; Revision 24
-USAR Section 3.4; Water Level (Flood) Design Criteria; Revision 2

1R04 Equipment Alignment

Procedures:
- DB-OP-06011; High Pressure Injection System; Revision 19
- DB-SS-03091; Motor Driven Feed Pump Quarterly Test; Revision 10

Drawings:
- OS-3; High Pressure Injection System; Revision 29
- OS-12A, Sheet 1; Main Feedwater System; Revision 23
- OS-17A, Sheet 1; Auxiliary Feedwater System; Revision 20

1R05 Fire Protection

Procedures:
- Davis-Besse Nuclear Power Station Fire Hazard Analysis Report
- DB-FP-00005, Attachment 1; Fire Brigade Drill Record; Revision 5
- DB-OP-02529; Fire Procedure; Revision 5
- PFP-TB-517; Pre-Fire Plan, Turbine Building Elevation 623'; Revision 5

Drawings:
- Drawing A-222F; Fire Protection General Floor Plan EL 585'; Revision 15
- Drawing A-223F; Fire Protection General Floor Plan EL 585'; Revision 20
- Drawing A-224F; Fire Protection General Floor Plan EL 603'; Revision 22
- Drawing A-2225; Barrier Penetration Drawing, Barrier 318-E1; Revision 0
- Drawing A-2236; Barrier Penetration Drawing, Barrier 325-E; Revision 1

1R08 Inservice Inspection Activities

NDE Procedures:
-54-ISI-835, Ultrasonic Inspection of Ferritic Piping Welds; Revision 12
-54-ISI-270, Wet or Dry Magnetic Particle Examination Procedure; Revision 42
-54-ISI-240, Visible Solvent Removable Liquid Penetrant Examination Procedure; Revision 40

- 54-ISI-369, VT-1, VT-3 and General Visual Examinations; Revision 00
- 54-ISI-367, Visual Examination for Leakage of Reactor Head Penetrations dated July 6, 2007

NDE Exam Documents:
-Data sheet 15-UT-021, Ultrasonic examination of DH-33B-CCB-6-5-SWA; dated January 10, 2008
-Data sheet 15-PT-038, Dye Penetrant examination of DH-33B-CCB-6-5-SWA; dated January 10, 2008.
-Data sheet 15-VT-095, Visual VT-3 examination of DG-JKT WTR HTXCHR-1-1 Supports; dated January 8, 2008.
-Data sheet 15-VT-093, Visual VT-1 examination of DG-JKT WTR HTXCHR-1-1 AW; dated January 8, 2008.
-Relief Request A30; For Application of Structural Weld Overlay on Dissimilar Metal Welds of Pressurizer Nozzles and Hot Leg Branch Connections (TAC NO. MD4452); Revision 2
-Relief Request A31; For Alternative for Pressurizer J-Groove Nozzle Weld Repairs (TAC NO. MD5956); dated December 14, 2007
-Code Case N-638-1; Similar and Dissimilar Metal Welding Using Ambient Temperature machine GTAW Temper Bead Technique Section XI, Division 1; dated February 13, 2003

Repair and Welding Documents:
-Drawing 403467; 3” Safety, Davis-Besse Construction Drawing; Revision 2
- Drawing DB-08Q-01; Pressurizer 3” SAFETY Relief Valve Nozzle Weld Overlay Design; Revision 1
- Drawing DB-08Q-03; Pressurizer 2.5” Pressure Relief Valve Nozzle Weld Overlay Design; Revision 1
- DB-SRV-2-WOL; Phased Array UT Examination Record, 3” SRV Nozzle WOL, SRV-2; dated January 16, 2008
- DB-SRV-2-WOL; Phased Array UT Examination Record, 3” SRV Nozzle WOL, SRV-3; dated January 16, 2008
- Report No. 103571-PT-052; Liquid Penetrant Inspection of Pressurizer SRV-2; dated January 16, 2008
- Report No. 103571-PT-051; Liquid Penetrant Inspection of Pressurizer SRV-3; dated January 16, 2008
- WPS 01-08-T-804-Top; Machine (GTAW), P-1 Groups 1 and 2 to P-8; Revision 1
- PQR-01-08-T-032; Manual (GTAW), P-1 to P-8; Revision 0
- PQR-01-01-T-802; P-1 Groups 1&2 to P-1 Groups 1 and 2; Revision 2
- PQR A843258-52; P-43 to P-8 Group 1; Revision 1
- WPS 08-08-T-001-Butter SS; Manual, Machine (GTAW), P-8 to P-8; dated Revision 0
- PQR-08-08-T-009; Manual (GTAW), P-8 to P-8, Revision 0
- PQR-08-08-TS-001; Machine (GTAW), Manual (SMAW), P-8 to P-8; dated Revision 0
- PQR-08-08-TS-002; Manual (GTAW, SMAW), P-8 to P-8; dated Revision 0
- PQR 8.8.6-OKG; Machine (GTAW), P-8 to P-8; dated June 4, 1998
- WSI Traveler No. 103571-TR-001; Work Traveler Pressurizer Nozzle Repairs; dated Revision 0

RV Upper Head Visual Documents:
- 03-6027636, Reactor Vessel Nozzle Penetration Remote Visual Inspection Plan for Davis-Besse Unit 1; dated November 7, 2007

Boric Acid Corrosion Inspection Documents:
- NOP-ER-2001; Boric Acid Corrosion Control Program; dated February 28, 2007
- CR 06-00669; Boric Acid on RC14CB – Pressurizer Level Transmitter Maintenance Isolation Valves; dated March 8, 2006
- CR 06-00707; Boric Acid on HP81 – High Pressure Injection Line Vent Valve; dated March 9, 2006
- CR 06-00774; Boric Acid on RC33 – Loop 2-1 Cold Leg Drain Valve; dated March 9, 2006
- CR 06-02402; Boric Acid Residue on MU32 – Normal Make-up Control Valve; dated May 30, 2006
- CR 07-16861; Boric Acid on CF3A1 - Core Flood Tank Level Transmitter; dated March 24, 2007

SG Inspection Documents:
- 51-9019601-000; Evaluation of Steam Generator Tubing at Davis-Besse, 14RFO; dated May 15, 2006
- 51-9064198-000; Davis Besse Degradation Assessment for 15th Refueling Outage (January 2008); dated December 19, 2007
- DBPM-SGMP-PE-001; Davis-Besse Steam Generator Management Program Manual; Revision 8

Condition Reports:
- CR 08-33006, Circumferential indication found in upper tubesheet in OTSG A
- CR 06-00773; BACC: Steam generator 1-2 Upper Manway
- CR 06-00775; Boric Acid on W/X Axis Pressurizer Relief Nozzle and Safe End (A-600 Walkdown)
- CR 06-00925; NRC report required Due to Circumferential Crack Found in Tube End in OTSG 1-B
- CR 06-00972; NRC ISI: Additional Examinations not Performed per ASME Section XI
- CR 06-01182; NRC ISI: Further Documentation of CR 06-00972 Deficiency
- CR 06-01183; NRC ISI: Reportability of CR 02-08782
- CR 06-00973; White Streaks on Main Feedwater Piping
- CR 06-01216; AREVA CR2006-1371: SG Tube Plugged Before ECT Exam Performed
- CR 06-01295; OTSG Eddy Current Examination Identified Defects in the OTSG Tubing
- CR 07-26238; 2005 INPO SG Program Review Visit

Corrective Action Documents Generated as a Result of ISI Inspection:
- CR 08-33133, Weld head travel speed calculation log not maintained
- CR 08-33129, Illumination checks on the RV Head Visual Exams performed in ambient lighting
- NCR No: 08-147; Second Layer Welding Performed Without Weld Head Travel Speed Calculation Sheet; dated January 12, 2008

<u>1R11 Licensed Operator Requalification Program</u>

Procedures:
- DBBP-TRAN-0017; Conduct of Simulator Training; Revision 4
- DBBP-TRAN-0502; Development and Conduct of Continuing Training Simulator Evaluations; Revision 5

Other:
- SG-ORQ-S230; Loss of NNI Y-AC, MU2B Fails to Reopen, Increasing Condenser Vacuum & Turbine Fails to Trip; Revision 1

1R12 Maintenance Effectiveness

Condition Reports:
- CR 06-6003; Manual Reactor Trip Due to Lowering Condenser Vacuum
- CR 08-36573; Inadvertant Addition of Station Air Into the Condenser Causing Degraded Vacuum
- CR 08-36574; Evaluate Rising Condenser Pressure and rise in Rx Power For Reactivity Management Event
- CR 08-36609; Evaluate Condenser Air In-Leakage For Reactivity Management Event
- CR 08-36734; Davis-Besse Snapshot Self Assessment DB-SA-08-037

Procedures:
- DB-OP-02518; Abnormal Procedure: High Condenser Pressure; Revision 4
- DB-PF-00003; Maintenance Rule Program Manual; Revision 7
- NOP-OP-1004; Reactivity Management; Revision 5

Drawings:
- OS-010 SH 1; Condensate System; Revision 14
- OS-010 SH 2; Condensate System; Revision 9
- OS-010 SH 3; Condensate System; Revision 13
- OS-013 SH 3; Extraction Steam System; Revision 17
- 012A SH 1; Main Feedwater System; Revision 23

Calculations:
- C-NSA-099.16-035 Current, Clean Condenser, 75 Injection Temp, Design CW Flow; Revision 0

Other:
- Unit Log Entries Report for Condensate Condenser; 3/7/08- 3/11/08
- Maintenance Rule a(1) Action Plan for Condensate Condenser; 9/6/2006
- INPO EPIX Failure Summary Report; 3/20/2008
- DB System Health Report, Condensate Condenser; First Quarter, 2006
- DB System Health Report, Condensate Condenser; Third Quarter, 2006
- DB System Health Report, Condensate Condenser; Second Quarter, 2007
- DB System Health Report, Condensate Condenser; Third Quarter, 2007
- DB System Health Report, Condensate Condenser; Fourth Quarter, 2007
- Maintenance Rule Expert Panel Meeting Minutes; October 12, 2006
- Maintenance Rule Expert Panel Meeting Minutes; November 9, 2006
- Maintenance Rule Expert Panel Meeting Minutes; February 22, 2008

1R13 Maintenance Risk Assessments and Emergent Work Control

Condition Reports:
- CR 08-32472; Plant in Orange Risk Longer than Planned During Initial Drain
- CR 08-32521: Pressure Boundary Leak Found During Decay Heat Drop Line Weld Overlay
- CR 08-32593; DB-PA-08-01; Decay Heat Nozzle Overlay Work in Orange Risk Needs to be CR 08-32640; Head Lift Stopped Due to Discrepancy With Package Weight
- CR 08-32656; Polar Crane Load Cell Questionable Readings

Procedures:
- DB-OP-06002; RCS Draining and Nitrogen Blanketing; Revision14

- DB-OP-06402; CRD Operating Procedure; Revision 14
- DB-OP-06901; Plant Startup; Revision 30
- DB-OP-06904; Shutdown Operations; Revision 24
- DB-OP-06912; Approach to Criticality; Revision 9
- NG-DB-00117; Shutdown Defense in Depth Assessment; Revision 18
- NOP-OP-1006; Shutdown Defense in Depth; Revision 10
- Areva 03-5016126; Reactor Vessel head Removal; Revision 4

Calculations:
- C-NSA-049.02-020; Core Decay Heat for RFA 02-0028; Revision 1, Addendum A03
- T-006A; Piping Stress Analysis for the Decay Heat Removal System for the Reactor to Containment Penetration P-29; Revision 3, Addendum A3

Other:
- 15RFO-2; Contingency Plan for RCS Drain/Fill Without Adequate RCS Vent Path; Revision 0 and 1 and 2
- 15RFO-3; Contingency Plan for RCS Drain Below Flange level and Operation Below 80 Inches Above the RCS Hot Leg Centerline; Revision 0 and 2
- 15RFO-4; Contingency Plan for NO CACs Available When at or greater than 23' in Refueling Canal; Revision 0
- 15RFO-OPS-01; RCS Drain to 16 to 20 Inches to Break RCP Siphon; January 3, 2008
- Unit Log Entries from January 4, 2008, Addressing Decay Heat Leak Issues
- Reactivity Plan Review Package, Cycle 16 Zero Power Physics Testing and Beginning of Cycle 16 Power Escalation; January 30, 2008
- Reactor Plant Event Number 43880; Declaration of Inoperability for Both Trains of Decay Heat Removal System; January 4, 2008
- Shutdown Defense in Depth Review for 15RFO; November 16, 2007

1R15 Operability Evaluations

Condition Reports:
- CR 07-30672; Boron-10 Depletion Evaluation
- CR 08-32521; Pressure Boundary Leak Found During Decay Heat Drop Line Weld Overlay
- CR 08-32708; ODMI: Operation with Leaking Decay Heat Suction Piping
- CR 08-33923; APSR D10 Has Missing Tang
- CR 08-35702; Concerns Regarding the Absence of Validation Air Was Not Introduced to DH System
- CR 08-35173; Main Steam Line #2 Remains Pressurized with MS100 and MS100-1 Closed
- CR 08-36648; Boron Concentrations Do Not Include Sample Accuracy
- CR 08-33964; Broken Tang on Axial Power Shaping Rod D10

Other:
- USAR Section 15.4; Class 3 Design Basis Accidents; Revision 25
- USAR Section 6.2; Containment Systems; Revision 25
- Event Notification 43880; Pressure Boundary Leakage Found During Refueling; January 4, 2008

1R18 Permanent Plant Modifications

Work Orders:
- 200249953; Alloy 600 12 Inch Decay Heat Line to Hot Leg Nozzle

Other:
- OPS-JIT-S07120; Plant Shutdown Cooldown Just-In-Time-Training 15RFO; Revision 0
- ECP 06-0143-00; Alloy 600 Mitigation; Revision 0 and 1
- ECP 06-0143-07; Alloy 600 Mitigation; Revision 0 and 1

1R19 Post Maintenance Testing

Condition Reports:
- CR 08-35267; Test Tags on SW 1357 and Shop Did Not Want Test Tags
- CR 08-35349; SW 1357 Stroked Only Partially
- CR 08-35465; SW 1358 Actuator Vent Valve (IA1358H) Was Not in Correct Position

Procedures:
- DB-MM-9118; EDG Governor Removal, Installation, and Adjustment; Revision 6
- DB-OP-3013; Containment Daily Inspection and Containment Closeout Inspection; Revision 5
- DB-OP-06402; CRD Operating Procedure; Revision 14
- DB-PF-3010; RCS Leakage Test; Revision 8
- DB-PF-3027; Service Water Train 2 Valve Test; Revision 23
- DB-PF-3065; System Leakage Tests; Revision 10
- DB-SC-3077, Emergency Diesel Generator 2 184 Day Test, Revision 14
- DB-SC-03270; Control Rod Assembly Insertion Time Test; Revision 6

Work Orders:
- WO200222807; PM 3379 – Calibration of Containment Air Cooler 2 Service Water Outlet Valve
- WO200229152; SC3270-001 08.200 Rod Drop CRA Insertion Time Test FA Norm DB-SC-03270
- WO 200242571; Replace Governor Controls, ECR 02-0738-00

1R20 Outage Activities

Condition Reports:
- CR 08-32418; Evidence of Leakage Found on the Reactor Vessel Flange
- CR 08-32482; Fuel Transfer Tube Blank Flanges Improperly Sealed
- CR 08-32509; Leadscrew for CRDM H-14 Was Dropped Several Inches (Areva CR 2008-49-CR)
- CR 08-33509; Incorrect Revision of AREVA Procedure Used to Perform RV Head Inspection
- CR 08-33710; Groundwater Inseepage Identified in the Annulus Sandpocket
- CR 08-33850; DB-PA-08-01 Potential Impact of RTV on Penetration P
- CR 08-34043; CFT1 Discharge
- CR 08-34050; Gas Void Detected at Core Flood Tank 1-1 Discharge Pipe
- CR 08-34346; CF30 Knocking Noise (NRC Identified)
- CR 08-36134; Areas Where AREVA Field Services Did Not Meet Expectations

Procedures:
- Areva 03-9060724; Davis Besse Reactor Vessel Head Reinstallation; Revision 0
- DB-MM-6002; Polar Crane Operation; Revision 12
- DP-OP-6900; Plant Heatup; Revision 39
- DB-OP-6901; Plant Startup; Revision 30
- DB-OP-6902; Power Operations; Revision 22

- DB-OP-6903; Plant Shutdown and Cooldown; Revision 26
- DB-OP-6904; Shutdown Operations; Revision 24

Calculations:
- C-NSA-049.02-049; Surface Areas Within Containment for Latent Debris Analysis; Revision 0

Other:
- 15RFO-4; Contingency Plan for No CACs Available When at $\geq$23’ in Refueling Canal; Revision 0
- 15RFO-5; Contingency Plan for Startup Transformer 01 Outage with Reduced Inventory in the RCS; Revision 0
- 15RFO-9; Contingency Plan for Work on MS603 and MS611 while in Orange Risk; Revision 0
- Problem Solving Plan; Transformer x02 Lighting Arrester Failure (CR 08-34065); Revision 0

1R22 Surveillance Testing

Condition Reports:
- CR 06-8387; Corrections for EDG 2 Monthly Test (DB-SC-03071)
- CR 07-32026; EDG 1 Exhaust Thermocouple 5 Reading Low
- CR 07-32110; DB-PF-03008; Containment LLRT Inconsistency
- CR 08-34339; NRC Tour of Mode 4 Readiness in Containment (NRC Identified)
- CR 08-34554; AFPT #2 LSS Speed is Higher Than 1100 RPM
- CR 08-35402; AFPT #2 Cover Gasket Degraded
- CR 08-36366; MS5889B Stroke Time Was Not Accurate
- CR 08-37256; Typographical Error in DB-SC-03071

Procedures:
- DB-OP-1101; Containment Entry; Revision 6
- DB-OP-3013; Containment Daily Inspection and Containment Closeout Inspection; Revision 5
- DB-OP-6402; CRD Operating Procedure; Revision 14
- DB-PF-3008; Containment Local Leakage Rate Tests; Revision 9
- DB-SC-4119; RPS Channel 3 Flow Scaling Factor Determination; Revision 8
- DB-SC-03071; EDG 2 Monthly Test; Revision 16
- DB-SP-3160; AFW 2 Quarterly; Revision 18
- DB-SP-3219; HPI Train 2 Pump and Valve Test; Revision 15
- DB-SP-04159; AFP 2 Monthly Test; Revision 10

Work Orders:
- WO200157512; Containment Vessel LLRT – Penetration 41
- WO200294711; TI2 0175 – T/C #5,16,29 Reading Low
- WO200313754; SP 3160-001 05.000 P14-2 QTRLY
- WO200413755; SP 3166-004 05.003 P14-2 QTRLY-S/D

Drawings:
- Operational Schematic OS-017A SH1; Auxiliary Feedwater System, Revision 20
- Operational Schematic OS-0041C; EDG Diesel Oil System, Revision 16

Other:
- ISTB3; Pump and Valve Basis Document Volume III, Stroke Time Basis; Revision 34
- ISTB2; Pump and Valve Basis Document Volume II, Pump Basis; Revision 9
- Operator Workaround Log Entries for 3/6/08 EDG Surveillance Activity

- USAR Section 8.3; Onsite Power Systems; Revision 20
- Notification 600432756; EDG 1-1 Exhaust T/C #5 Reading Low
- Notification 600451264; EDG 2 Cylinder Temp #7 and #20
- Notification 600454812; DB-SC-03071 Correction

1EP2 Alert and Notification System (ANS) Evaluation

Condition Reports:
- CR 07-27935; Initial Indications That Four EPZ Sirens Failed to Respond to Test
- CR 07-17181; Five EPZ Sirens in Lucas County Were Not Activated during Tornado Drill
- CR 07-27004; Post Maintenance Test of Ottawa County Primary ARM Command for Sirens Failed
- CR 07-26804; Siren Test Command from Ottawa County Sheriff Dispatch Did Not Transmit

Procedures:
- RA-EP-0400; Prompt Notification System Maintenance; Revision 5
- RA-EP-0420; Response to Prompt Notification System Malfunction; Revision 4
- RA-EP-0440; Prompt Notification System Test; Revision 9

Other:
- Davis-Besse Nuclear Power Station Emergency Plan, Section 7.7; Prompt Notification System; Revision 25
- Davis-Besse Prompt Notification System Design Report; dated November 1986
- Final Updated Design Report for the Davis-Besse Nuclear Power Station Prompt Notification System; dated April 27, 2007
- Records of 2006 and 2007 Annual Siren Preventative Maintenance

1EP3 Emergency Response Organization (ERO) Augmentation Testing

Condition Reports:
- CR 07-27069; Paging Upgrade Issue; dated September 25, 2007
- CR 07-27267; Emergency Response Facility Walkthrough Attendance; dated September 27, 2007
- CR 07-25576; Paging Delays Experienced on August 20, 2007
- CR 07-25843; Staff Augmentation Drill Results
- CR 0725944; ERO Qualifications Not Correct in FITS Qualification Matrix

Procedures:
- RA-EP-00100; Emergency Plan Training Program; Revision 11
- RA-EP-00550; Computerized Automated Notification System; Revision 5
- RA-EP-02110; Activation Notification; Revision 8
- RA-EP-2310; TSC Activation and Response; Revision 5
- RA-EP-2410; OSC Activation and Response; Revision 12

Other:
- Davis-Besse Nuclear Power Station Emergency Plant, Table 5-1; Manpower, Location, and Response Considerations for Emergencies; Revision 25
- Emergency Plan Telephone Directory; Revision 96-

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

Condition Reports:
- CR 07-27287; EP Drill: Dose Assessment Critique Items from the September 20, 2007 EP Integrated Drill
- CR 07-25163; NRC NCV: Seismic Monitor Out of Service Affecting Emergency Plan Response
- CR 07-21429; Implementation of Site Radiological Controls during Emergencies Lack Consistency
- CR 07-20678; E-Plan Exercise Items Noted in the Emergency Operations Facility
- CR 07-21043; EP Evaluated Exercise: NRC Comments Regarding Player Critique
- CR 07-20669; EP Exercise: Concerns Regarding Similarities of Emergency Response Drill Scenarios

Procedures:
- DB-SA-08-24; Davis-Besse 2008 EP Baseline Pre-Inspection Assessment; dated February 18, 2008

Other:
- Davis-Besse Oversight EP Program Annual Reviews for 2006 and 2007

2OS1 Access Control to Radiologically Significant Areas

Condition Reports:
- CR 08-32725; MG Dose Rate Alarm on NPS Insulator
- CR 08-32371; DB-PA-08-01, Blowout Panel from MPR-4 to Turbine Building was not Posted as RCA Opening
- CR 08-32628; Adverse Trend in Dose Rates Alarms
- CR 08-32631; Use of Wrong RWP Task for Radiological Work
- CR 08-35812; Personnel airlock hand-wheels to containment, a Lock High Radiation Area was Found Unsecured

Procedures:
- NOBP-SS-2008; High and Locked High Radiation Area control Action Plan; Revision 0
- DB-HP-1109; Radiation Protection Procedure; High Radiation Area Access Control; Revision 24
- DB-MS-1637; Scaffolding Erection and Removal, Revision 10
- DB-HP-1802; Control of Shielding; Revision 08
- DB-HP-1152; Radiation Protection Procedure; Performance of High Exposure Work; Revision 10
- NOP-WM-1001; Nuclear Operating Procedure; Order Planning Process; Revision 9

2OS2 ALARA Planning and Controls

Condition Reports:
- CR 08-32682; Two Air Carts Failed To Operate While Supporting S/G Work Using Bubble Hood
- CR 08-10937; Work Order Could Not Be Worked as Scheduled Because ALARA Reviewed Was Not Performed
- CR 08-32532; MU6 Valve Work Was Delayed By RP Due To The General Area Dose Rates Being Higher Than The Estimated Dose Rates

CR 08-32401; RWP/ALARA Plan and Work Orders Do Not Agree
- CR 07-23867; Reverse Benchmark of Davis Besse RP Readiness for the Coming RFO-15
- CR 08-32408; WSI Welder Received a Dose Rate Alarm during a Fire Watch of Surge Nozzle Work
- CR 08-32482; Fuel Transfer Tube Blank Flanges Improperly Sealed, Caused Hundred Millirems Dose Spent during Every Outage
- CR 08-32725; Insulator Received MG Dose Rate Alarm While Working in the Containment; dated PDA-07-024
- CR 08-32711, AMS-4S Alarming in Containment
- CR 08-32732; WSI Supervisor Received an MG Dose Rate Alarm While Providing Oversights of Alloy 600 Project
- CR 08-33007; Worker Received Dose Rate Alarm during Spent Fuel Pool Filter Transfer

Procedures:
- RWP 2008-5600; Alloy 600 PZR Weld Overlay on Ventline, Safety Relief Valves, Spray Line, Surge Line to Pressurizer, and Lower and Upper Sample Nozzles; dated December 26, 2007
- RWP 2008-5300; Stream Generator Platform Setup To Include Scaffolding, Interference Removal, Install/Remove Tent, HEPA Setup, Staging Equipment and Temporary Shielding; dated December 22, 2007
- RWP 2008-6003; Replacement and Changeout of Various Filters, Including But Not Limited to, the Spent Fuel Pool Make Up, Purification and Let-Down and Seal Injection; dated December 20, 2007
- RWP 2008-5114; Incore Tank Work Activities that Included Cutting, Transfer, Decon Remove and Radiation Support for these Activities; dated December 22, 2007
- RWP 2008-5114; ALARA Plan for Incore Work Activities; dated October 28, 2007
- RWP 2008-5601; Decay Heat Suction Line Overlay; dated December 26, 2007
- RWP 2008-5602; Work To Include Grinding Off Weld Overlay Beads, Non Destructive Examinations on Repair Area, Radiation Protection Support, Decontamination and Shielding Support, Weld Repair and Firewatch; dated January 8, 2008
- RWP 2008-5602; Decay Heat Suction Line Weld Preparations to Support Overlay, Activity Include Grinding and NDE Repair; dated January 9, 2008
- RWP 2008-5600 and ALARA Plan; Pressurizer Weld Overlay on Surge Line to Pressurizer (Pzr) And Pzr Surge Line to Hot Leg Work Activities; dated September 29, 2007
- NOP-WM-7001; ALARA Program; Revision 01
- NOP-WM-7002; Operational ALARA Program; Revision 01
- DB-HP-1801; ALARA Design Review; Revision 03

Other:
- Cycle 15 Outage Log; Radiation Protection OCC Manager; dated January 9, 2008
- 15 RFO Human Performance Message and Human Performance Stand-down; dated January 3, 2008
- 15 RFO Dose Estimate; High Dose Activity Projects
- Shift RP Turnover; Outage Shift Turnover; dated January 9, 2008
- Davis Besse RFO15; 100-Hour Safety and Human Performance Standdown; dated January 3, 2008

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

Condition Reports:
- CR 07-29410; Root Cause Analysis Report; Technical Specification Violation Due to Both Trains of Station Vent Radiation Monitors Out of Service
- CR 07-28428; RE4598BA Station Vent Normal Range Radiation Monitor Was Declared Inoperable And TS 3.7.6.1 Action B Was Entered Into 7 Days LCO
- CR 07-29011; No Adequate Flow Of Sample To The Station Vent Normal Range Monitor During DB-CN-03008, "Station Vent Releases" Surveillance And Requesting Maintenance Rule Functional Failure Evaluation To Be Performed
- CR 07-30509; Main Steam Line No. 1 Radiation Monitor Warning Light Set At 50 Cpm Intermittently Alarming Due N-16 Back Ground
- CR-08-33443; Containment Wide Range Radiation Monitor Alarming Due Internal Check Source Problems
- CR-07-32123; Dose Rate Alarm Was Received While Attempting To Access Service Water Piping To Perform A UT Evaluation
- CR-07-27578; As Found Readings Were Greater Than 1 Percent Difference On The Gilibrator Flow Cell When Vendor Sensidyne, Inc. Performed As Found Calibration
- CR 07-26512; A Quantitative Respiratory Fit Test Was Not Performed At Davis Besse For A Perry Radiation Protection Technician Who Was Not Qualified To Perform Tasks At Davis Besse
- CR 08-32682; Two Air Supply Carts Failed To Operate Properly While Supporting Steam Generator Work Using Bubble Hood, It Was An Electronic Problem
- CR 07-31908; A Vendor Representative Informed The Licensee That Sample Collected From Bauer And Quincy Air Compressors Had Exceeded The Limit For Carbon Dioxide
- CR-08-33054; A Forklift Exhaust In High Bay Tripped The Carbon Dioxide Alarm On Air Compressor Breathing Air Carts
- CR 08-32490; There Were Issues Associated With Obtaining Grade D Air Quality Certification For The Station Air, Specifically, On The Bauer And Quincy Air Compressor
- CR 07-24825; Failed Station Effluent Radiation Element RE8433 Due To Thunderstorm
- CR 07-24773; Station Effluent Radiation Monitor RIM8433 Failed Requiring Chemistry To Sample Per DB-OP-6412
- CR 07-23373; During A Preventive Maintenance, I&C Crew Discovered That The Single Channel Analyzer Circuit Board On RE1998 Digital Ratemeter Appeared To Be
- CR 07-22520; RE45977AA Containment Normal Range Radiation Monitor Flow Meter Contained Water. This Was Re-Occurring Issues
- CR 07-21729; Waste Gas System Discharge To Station Vent RE1822A Was Found To Be Out Of Tolerance And It Was Recalibrated

Procedures:
- DB-HP-06122; Calibration and use of the PCM-2, Revision 3
- DB-HP-01320; Operation of whole body counters, Revision 9

Other:
- Radiation Monitor Setpoint Manual For Small Article Monitor (SAM-11); Revision 1
- Radiation Monitor Setpoint Manual For Portal Monitor (SPM-906); Revision 4
- DB-HP-01435; Calibration and use of the portal monitor SPM 904C/SPM 906; Revision 3
- DB-HP-01442; Calibration data sheet for MG Telepole;, February 22, 2008
- DB-0125-2; Calibration data sheet for small article monitor; December 7, 2007
- DB-0190-2; Portal monitor calibration record; May 4, 2007
- DB-0190-2; Portal monitor calibration record; July 24, 2007

- DB-0178-3; AMS-4 calibration record; August 6 and 14, 2007
- MS-C-07-08-03; Quality Assurance Audit Report; Radiation Protection And Radwaste Processing Program; September 28, 2007
- DB-SA-07-039; Assessment To Determine The Accuracy And Operability Of Radiation Monitoring Instruments Used For The Protection Of Occupational Workers, And To Review The Adequacy Of The Respiratory Protection Program To Provide SCBA To Individuals Entering RCA; May 24, 2007
- Davis Besse System Health Report 2007-1, DB-SUB079-01-Radiation Monitoring and Process and Area; Health Improvement Plans for Kaman Radiation Monitors; Replacement Project for 2010 in DB 5 Year Capital Plan; May 24, 2007

4OA1 Performance Indicator (PI) Verification (71151)

Alert and Notification System Reliability (ANS)
- DBBP-EMER-0003; NRC Performance Indicator for ANS Reliability; Revision 7
- CR 07-28073; Alert Notification System Reliability PI Correction Due to Siren 093 Failure; dated October 8, 2007
- CR 07-21749; Siren Number 201 Out of Service for Maintenance during Weekly Test; dated June 7, 2007

Drill and Exercise Performance (DEP)
- DBBBP-EMER-004; NRC Performance Indicator for Drill and Exercise Performance;
- Revision 4
- CR 08-35503; Drill and Exercise Performance Inconsistencies
- CR 07-19093; Changes Made to Two NRC/NEI Emergency Response Performance Indicators

Emergency Response Organization Participation
- DBBP-EMER-0002; NRC Performance Indicator for ERO Drill Participation; Revision 6
- CR 07-25073; Recent Decline in NRC Performance Indicator for ERO Drill Participation
- CR 07-19880; NRC ERO Drill Participation Performance Indicator Data Correction

Other:
- Licensee Logs Documenting Results of Daily RCS Leakage Measurements
- Davis-Besse Licensee Event Reports for Events Occurring in 2007

4OA2 Problem Identification and Resolution

Condition Reports:
- CR 07-12189; Unexpected Trip of CREVS Train 2 Compressor During Monthly Test
- CR 07-19255; CREVS 2 Compressor Trip During Monthly Surveillance Testing DB-SS-03042
- CR 07-29963; Control Room Emergency Cooler Units
- CR 07-32112; {ressurizer Level Decrease While Placing DH Train 1 in Standby
- CR 08-32662; Work Associated with Order 200097432
- CR 08-33026; Broken Tubing On Train 2 CREVS Causes Loss of Refrigerant
- CR 08-33531; CREVS Has Exceeded Its Maintenance Rule Performance Criteria
- CR 08-34173; CREVS Train 1 Low Refrigerant Charge
- CR-08-34348; Small R-12 Leak on S33-1 and S33-2
- CR 08-34425; Question About CREVS Train 1 Operability Testing
- CR 08-36684; CREVS 1 Compressor Trip During Monthly Test DB-SS-03041

- CR 08-36706; CREVS Train 1 Refrigerant Leak
- CR 08-37010; Work Performed By Unqualified Personnel- Contractor

Procedures:
- DB-PF-00003; Maintenance Rule Program Manual; Revision 7
- DB-SS-3041; Control Room Emergency Ventilation System Train 1 Monthly Test
- DB-OP-6505; Control Room Emergency Ventilation System Procedure; Revision 10
- NOBP-LP-2008; FENOC Corrective Action Review Board; Revision 6
- NOBP-LP-2008-01; Evaluation Review Checklist; Revision 7
- NOP-LP-2001; Corrective Action Program; Revision 13

Work Orders:
- WO 200097432; Replace CREVS Train 1 Piping

Drawings:
- OS-032B; Control Room Emergency Ventilation System; Revision 16
- OS-004, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 45

Other:
- Unit Log Entries Report for CREVS; 8/3/07- 3/25/08
- Maintenance Rule (a)(1) Evaluation Form: CREVS; January 9, 2008
- DB Corrective Action Review Board Meeting Minutes; March 17,2008
- CARB Package, March 17, 2008; CR 07-29963 Control Room Emergency Cooler Units
- DB System Health Report, HVAC Control Room; Third Quarter, 2006
- DB System Health Report, HVAC Control Room; Second Quarter, 2007
- DB System Health Report, HVAC Control Room; Third Quarter, 2007
- DB System Health Report, HVAC Control Room; Fourth Quarter, 2007
- Maintenance Rule Expert Panel Meeting Minutes; February 22, 2008
- DB Plant Health Red/Yellow Actions as of 3/26/08
- INPO EPIX Failure Summary Report for Davis-Besse CREVS; 3/20/2008
- Notification 600149677; Replace CREVS Train 1 Piping
- Trane Service Bulletin HCOM-SB-49; Reciprocating Compressors, All Models – Operating Oil Level; July 1, 1981

4OA3 Followup of Events and Notices of Enforcement Discretion

Condition Reports:
- CR 08-35573; Power Decrease to Repair 2-6 HP FW Normal Level Controller
- CR 08-36528; Probable FW Heater Tube Leak in Train 1 High Pressure FW Heaters
- CR 08-36573; Inadvertent Addition of Station Air into the Condenser Causing Degraded Vacuum
- CR 08-36574; Evaluate Rising Cond. Press and Rise in RX PWR for Reactivity Management Event
- CR 08-36575; Temporary Plug Ejected During Leak Test of HPFW Heater 1-5
- CR 08-35889; Cycle 16 Fuel Defect: Radiochemistry Data Indicates Fuel Defect
- CR 08-36341; ODMI: Cycle 16 Fuel Defect Operation

Procedures:
- DB-OP-00000; Conduct of Operations; Revision 13
- DB-OP-6202; Turbine Operating Procedure
- DP-OP-6229; High Pressure Feedwater Heater System Operation; Revision 9

- DB-SP-3212; Venting of ECCS Piping; Revision 10
- NOP-NF-1102; Fuel Integrity Monitoring and Assessment; Revision 2

LIST OF ACRONYMS USED

AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
BA	Boric Acid
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CAR	Corrective Action Report
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CR	Condition Report
DH	Decay Heat System
DOT	Department of Transportation
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDY	Effective Degradation Years
EPRI	Electric Power Research Institute
ET	Eddy Current Examination
GTAW	Gas Tungsten Arc Welding
HPI	High Pressure Injection System
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IR	Inspection Report
IR	Issue Report
ISI	Inservice Inspection
KJ/in	Kilojoules per Inch
mrem	Millirem
NCV	Non-Cited Violation
NDE	Non Destructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PARS	Publicly Available Records
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Planned or Preventative Maintenance
PMT	Post-Maintenance Testing
PT	Dye Penetrant Examination
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RI	Resident Inspector
RI-ISI	Risk-Informed Inservice Inspection Program
RP	Radiation Protection
RPS	Reactor Protection System
RWP	Radiation Work Permit
SAM	Small Article Monitor
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
SG	Steam Generator

SRV	Safety Relief Valve
SSC	Structures, Systems and Components
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
UT	Ultrasonic Examination
VHRA	Very High Radiation Area
WANO	World Association of Nuclear Operators
WOL	Weld Overlay
WPS	Weld Procedure Specification