July 20, 2007

EA-03-025; EA-03-0214; EA-05-066;EA-05-067;
EA-05-068; EA-05-069; EA-05-070; EA-05-071;
EA-05-072

Mr. Mark B. Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2007003

Dear Mr. Bezilla:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Davis-Besse Nuclear Power Station. The enclosed integrated inspection report documents the inspection findings which were discussed on July 3, 2007, with Mr. Kaminskas and other members of your staff. Additionally, this inspection report documents special inspection activities associated with your compliance with the March 8, 2004, Confirmatory Order (EA 03-214).

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

Based on the results of this inspection, there are two NRC-identified findings and one self-revealed finding of very low safety significance, all of which involved violations of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these findings as non-cited violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

This inspection report also documents the results of a follow-up review conducted to assess the corrective actions associated with nine previously documented violations related to the reactor pressure vessel head degradation issue. These violations were originally issued in the April 24, 2005, "Notice of Violation and Proposed Imposition of Civil Penalties -$5,450,000; (NRC Office of Investigations Report No. 3-2002-006; NRC Special Inspection Report No. 50-346/2002-08(DRS))." The results of this review are located in Section 4OA5 of this report.

Section 4OA5.3 of the attached inspection report documents our assessment of your activities in complying with the March 2004 Confirmatory Order in the safety culture and safety conscious work environment (SC/SCWE) area. Based on our inspection activities, we have concluded that the SC/SCWE at Davis-Besse continues to be adequate to support safe facility operations. Further, actions taken by FirstEnergy Nuclear Operating Company (FENOC) to improve SC/SCWE continued to have a positive impact as evidenced by a continued decline in the number of negative responses to FENOC's internal SCWE survey. In addition, we confirmed that FENOC had met its commitment in 2006 to conduct an annual external independent assessment of the SC/SCWE at Davis-Besse. However, we are concerned that the survey tool used by your contractor for the 2006 external assessment was not developed using standard scientific methods. Because the external survey was not developed using industry-recognized techniques, the staff is concerned with its use as a stand alone methodology for assessing safety culture and questioned its results. The staff concluded that the external assessment's results were reasonable after using your internally-developed Employee Concerns Program SCWE survey results as an independent check against the external survey. The Employee Concerns Program SCWE surveys have been in good agreement with previous external assessments and the 2006 survey was consistent with the 2005 survey. We are aware of ongoing activities to evaluate the contractor's survey instrument. The results of that evaluation have the potential to affect our concern and will be reviewed when available.

If you contest the subject or severity of any 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 copies to the Regional Administrator Region III, U.S. Nuclear Regulatory Commission, 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 NRC 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 available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

***/RA by S. West Acting for/***

Cynthia D. Pederson, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000346/2007003
w/Attachment: Supplemental Information

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Sincerely,

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Cynthia D. Pederson, Director
Division of Reactor Projects

Docket No. 50-346
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Letter to M. Bezilla from C. Pederson dated July 20, 2007

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2007003

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
Richard Anderson, Vice President,
Nuclear Support - 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
Ohio State Liaison Officer
R. Owen, Administrator, Ohio Department of Health
Public Utilities Commission of Ohio
President, Lucas County Board of Commissioners
President, Ottawa County Board of Commissioners

Letter to M. Bezilla from C. Pederson dated July 20, 2007

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2007003

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

| | |
|---|---|
| Docket No: | 50-346 |
| License No: | NPF-3 |
| Report No: | 05000346/2007003 |
| Licensee: | FirstEnergy Nuclear Operating Company (FENOC) |
| Facility: | Davis-Besse Nuclear Power Station |
| Location: | 5501 North State Route 2<br>Oak Harbor, OH  43449-9760 |
| Dates: | April 1, 2007, through June 30, 2007 |
| Inspectors: | J. Rutkowski, Senior Resident Inspector<br>R. Smith, Resident Inspector<br>G. Wright, Project Engineer<br>J. Jacobson, Senior Reactor Inspector<br>P. Lougheed, Senior Engineering Inspector<br>N. Feliz-Adorno, Reactor Engineer<br>T. Go, Health Physicist<br>R. Winter, Reactor Inspector<br>J. Persensky, Senior Technical Advisor<br>M. Keefe, Human Factors Specialist |
| Observers: | R. Clagg, Reactor Engineer<br>P. Zurawski, Reactor Engineer |
| Approved by: | B. Burgess, Chief<br>Branch 6<br>Division of Reactor Projects |

Enclosure

## SUMMARY OF FINDINGS

IR 05000346/2007003; 4/1/2007 - 6/30/2007; Davis-Besse Nuclear Power Station; Surveillance Testing, Event Followup, Other Activities

This report covers a three-month period of baseline inspection. The inspection was conducted by resident inspectors and regional specialists. Three Green findings 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: Mitigating Systems

Green. A non-cited violation (NCV) of Technical Specification 6.8.1 was identified by the NRC regarding adherence to the procedural requirements for independent verifications required by safety-related surveillance procedures for instrumentation and control mitigation systems. The licensee used procedure-step verification techniques in their instrumentation and control department that were not in compliance with their procedures. Upon identification, the licensee entered the issue into their corrective action program and instructed personnel to use the procedure-required independent verification methodology.

The finding was more than minor because the finding was associated with the configuration control and testing procedure quality attributes of the mitigating systems cornerstone. This finding affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The improper completion of procedure-required verifications provided less than adequate assurance that important components of mitigation systems were properly positioned. The inspectors determined that the finding was of very low safety significance because there was no actual loss of safety function of mitigation systems. The inspectors also determined that the finding affected the cross-cutting area of human performance. The licensee's work practices did not support effective communication of the proper application of human error prevention techniques specified in instrument testing procedures, and supervisory oversight of the instrument testing work did not support proper application of the specified technique (H.4(b)). (Section 1R22)

#### Cornerstone: Barrier Integrity

Green. A self-revealing NCV of the plant operating license was identified during normal plant operations when on June 8, 2007, control room personnel observed that the plant's computer was not scanning reactor coolant letdown flow after work was performed to upgrade computer programs. Letdown flow was a variable used in the computer's calculation of reactor core power. The period of time that the variable was not being scanned was approximately 15 hours. That caused calculated reactor core power to be displayed as 0.15 percent lower than actual, which resulted in the plant exceeding 100 percent power when averaged over an 8-hour period. Exceeding an 8 hour average of 100 percent power was a violation of the plant operating license.

The finding was more than minor because it was associated with the fuel cladding thermal limits design control attributes of the barrier integrity cornerstone and did affect the cornerstone objective of reasonable assurance that the fuel cladding physical design barrier provide protection from radio nuclide release caused by accidents or events. The finding is of very low safety significance because the issue did not have any measurable impact on the fuel cladding. This finding was also associated with the cross-cutting area of human performance because in the work control process the operational impact of computer-upgrade work activities, that affected calculated reactor core power, was not appropriately considered (H.3(b)). (Section 4OA3.3)

**Cornerstone: Emergency Preparedness**

Green. Inspectors identified an NCV of 10 CFR 50.54(q) and 50.47.b(4) for the failure to provide alternate event assessment methods while the seismic force monitor was out-of-service during the period of March 29 through April 10, 2007. The licensee failed to provide a means for the emergency director to promptly classify seismic events at the alert or site area emergency levels while the seismic force monitor utilized by the operators (emergency director) was out of service. The licensee restored the seismic force monitor to service on April 10, 2007, which restored assessment capability.

The issue was more than minor because it was associated with the response organization planning standards attribute of the emergency preparedness cornerstone. This issue affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it did not result in the failure or degradation of a risk significant planning. Also, the unavailability of the seismic monitor did not prevent the declaration of a Site Area Emergency or Alert classification. This finding was also associated with the cross-cutting area of human performance. Licensee's work control process failed to establish compensatory measures for the out-of-service duration of the seismic force monitor (H.3(a)). (Section 4OA5.2)

**Cornerstone: Occupational Radiation Safety**

None

**B. Licensee-Identified Violations**

None.

# REPORT DETAILS

## Summary of Plant Status

At the beginning of the inspection period, the plant was operating at 100 percent power. On May 18, 2007, the licensee lowered power to about 19 percent to add oil to the upper bearing of the reactor coolant pump 2-1. Power was then raised to approximately 50 percent for condenser tube plugging activities. Following the condenser tube plugging activities, the plant was returned to 100 percent power on May 21, 2007. On June 22, 2007, the licensee lowered power to about 97 percent to connect a new cooling water return line to the condensate pumps and remove the temporary return line. Following the connection of the cooling water return line, the plant was returned to 100 percent power on June 23, 2007. The plant operated at approximately 100 percent power for the remainder of the inspection period.

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's restoration of systems from cold weather configurations and the licensee's preparations for hot weather operations. Particular emphasis was placed on the readiness of the emergency diesel generator systems for hot weather and restoration of cold weather actions for the service water system. This included a review of the requirements and work orders for changing to higher heat resistant oil in the diesel generator air intake filters and removal of plywood over the intake structure south ventilation penthouse. Additionally, the inspectors reviewed the licensee's procedural requirements and conducted walkdowns to determine whether ventilation systems and other equipment were properly realigned for hot weather. The inspectors also performed a walkdown of actions associated with Mayfly infestation. The inspectors reviewed licensee procedures and interviewed on-shift personnel for actions associated with grid reliability and communication protocols between the licensee and off-site transmission system operator. The inspectors monitored key equipment temperature trends during hot weather operations. The inspectors also interviewed operations personnel on their completion of hot weather preparations.

This constitutes one sample of a review of hot weather preparations of two risk significant systems.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed a partial walkdown of the following systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. 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, walked down control systems components, and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that the licensee had properly identified and resolved any equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barrier integrity and entered them into the corrective action program (CAP). Documents reviewed are listed in the Attachment. Systems reviewed were:

- control room emergency ventilation system train 2 on May 10, 2007, during a train 1 outage; and
- high pressure injection train 1 during train 2 inoperability for a maintenance activity on May 29, 2007.

This review represented two quarterly inspection samples of partial system walkdowns.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted a complete walkdown of the high pressure injection (HPI) system to verify the functional capability of the systems. The inspectors used the licensee procedures and other documents listed in the Attachment to verify proper system alignment.

The inspectors also verified electrical power requirements, operator workarounds, component labeling, hanger and support installation, and associated support systems status. Pumps, if operating, were examined to ensure that any noticeable vibration was not excessive, pump leakoff was not excessive, bearings were not hot to the touch, and the pumps were properly ventilated. The walkdowns also included evaluation of system piping and supports against the following considerations:

- piping and pipe supports did not show evidence of water hammer;
- oil reservoir levels appeared normal;
- snubbers did not appear to be leaking hydraulic fluid;
- hangers were functional; and
- component foundations were not degraded.

The inspectors also reviewed outstanding maintenance work orders to verify that any deficiencies identified did not significantly affect system function. In addition, the inspectors reviewed the condition report (CR) database to verify that any equipment alignment problems were being identified and appropriately resolved. The inspectors also reviewed an engineering change package for the installation of HP41 valve to prevent a recurrence of nitrogen migration into the HPI system. Documents reviewed during this inspection are listed in the Attachment.

This review represented one inspection sample of complete system walkdown.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Fire Zone Walkdowns (71111.05Q)

a. Inspection Scope

The inspectors toured the areas listed below to assess the material condition and operational status of fire protection features. The inspectors determined whether combustibles and ignition sources were controlled in accordance with the licensee's procedures; fire detection and suppression equipment was available for use; passive fire barriers were maintained in good material condition; and compensatory measures for out-of-service, degraded, or inoperable fire-protection equipment were implemented in accordance with the licensee's fire plan. Areas toured were:

- electrical penetration room 1 (Fire Area DG, Rooms 402);
- station blackout diesel generator building and associated yard areas during isolation of the fire main to these areas due to a leak in the system of about 20 gallons per minute;
- intake structure (Fire Areas BD, BE and BF, Rooms 50, 51, 52 and 54);
- auxiliary building 545' elevation (Fire Area A, Rooms 106, 107, 108, 109, 110, and 111);
- emergency core cooling system room 2 (Fire Area A, Room 115); and
- high voltage switchgear room B (Fire Area Q, Room 323).

This review represented six quarterly inspection samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

1. Internal Flooding (71111.06)

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and licensee procedures intended to protect the plant and its safety-related service water system from internal flooding events. The inspectors reviewed risk assumptions for internal flooding and updated final safety analysis report descriptions of factors affecting flooding potential. Additionally, the inspectors reviewed and monitored licensee contingency actions associated with internal flooding potential during the scheduled work for service water pump replacement. That work required opening of portions of the service water system through which, if lake levels were elevated, internal flood paths could exist. The inspectors also reviewed the visible condition of accessible below-grade service water system components.

This review represented one inspection sample for internal flooding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

On April 10, 2007, the inspectors observed an operating crew during a crew simulator quarterly evaluation and attended the post-session licensee controller critique. The operational scenario included a loss-of-service water due to a pump trip, failure of the main generator voltage regulator, main generator hydrogen leak, a reactor trip and overcooling event. The inspectors reviewed crew performance in the areas of:

- clarity and formality of communications;
- ability to take timely action in a safe direction;
- ability to prioritize, interpret and respond to alarms;
- procedure use;
- oversight and direction from supervisors; and
- group dynamics.

Crew performance in these areas was compared to licensee management expectations and guidelines as presented in Davis-Besse operational and administrative procedures.

This review represented one quarterly inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

.1 Quarterly Reviews (71111.12Q)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and resolution of performance issues associated with the core flood tank fill line from the high pressure injection system. The review consisted of evaluating the following items:

- use of the CR process in identifying deficiencies and issues with system equipment;
- if equipment performance issues were correctly categorized for reliability in accordance with the system's scoping sheet performance criteria;
- if the licensee effectively tracked key parameters, identified system trends, and monitored for signs of component failures;
- if the physical condition of the system appeared consistent with the status as reflected in CRs and open work orders; and
- if the licensee's corrective actions included the extent of condition.

Additionally, the inspectors performed a walkdown of the portions of the systems that were found to contain nitrogen and discussed planned corrective actions with the licensee's problem solving team lead.

This review represented one quarterly inspection sample.

b. Findings

No findings of significance were identified.

.2 Triennial Periodic Evaluation Reviews (71111.12T)

a. Inspection Scope

The inspectors examined the maintenance rule periodic evaluation report completed for the period of March 2004 through April 2006. The inspectors reviewed a sample of (a)(1) action plans, performance criteria, functional failures, and CRs to evaluate the effectiveness of (a)(1) and (a)(2) activities. These same documents were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. Also, the inspectors reviewed the licensee's maintenance rule procedures and processes. The inspectors focused the inspection on the following systems (samples):

- Medium Voltage AC;
- Condensate/Condenser;
- Safety Features Actuation System;
- Integrated Control System; and
- Station and Instrument Air.

The inspectors verified that the periodic evaluation was completed within the time restraints defined in 10 CFR 50.65 (once per refueling cycle, not to exceed 24 months). The inspectors also ensured that the licensee reviewed its goals, monitored structures, systems, and components (SSCs) performance, reviewed industry operating experience, and made appropriate adjustments to the maintenance rule program as a result of the above activities.

The inspectors verified that:

- the licensee balanced reliability and unavailability during the previous cycle, including a review of high safety significant SSCs;
- (a)(1) goals were met, that corrective action was appropriate to correct the defective condition, including the use of industry operating experience, and that (a)(1) activities and related goals were adjusted as needed; and
- the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, and reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

In addition, the inspectors reviewed maintenance rule self-assessments and audit reports that addressed the maintenance rule program implementation.

This review represented five triennial inspection samples.

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 following activities to determine whether the appropriate risk assessments were performed prior to removing equipment for work. The inspectors determined whether the risk assessments were performed as required by 10 CFR 50.65(a)(4) and if they appeared accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee risk assessment tool and risk categories in accordance with procedures and observed licensee personnel’s response to changes in planned activities. Documents reviewed are listed in the Attachment. Activities reviewed were:

- initial risk summaries for the week of April 1, 2007, during the implementation of the leading edge flow meter (LEFM) used for feedwater system parameter inputs to the secondary heat balance;
- initial risk summaries for the week of April 8, 2007, and revised schedules due to emergent issues with high pressure injection train 1 inoperability due to voided discharge lines;
- initial risk summaries for the week of April 15, 2007, and revised schedules due to the removal of orange risk work activities from the schedule and emergent switchyard air operated breaker work;
- risk summaries for the week of May 13, 2007, with emphasis on the planning and risk associated with a power reduction for condenser tube leak inspections and a containment entry for reactor coolant pump oil addition; and
- initial risk summaries for the week of May 20, 2007, and revised schedules for exiting yellow risk work activities due to the component cooling water pump 2 returning from outage sooner than originally scheduled and moving the auxiliary feedwater functional test to later in the week to avoid entry into orange risk level.

This review represented five inspection samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

For the operability evaluations described in the CRs listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that Technical Specification (TS) operability was properly evaluated and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors reviewed the Updated Safety Analysis Report (USAR) to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as intended and the measures were adequately controlled. The inspectors also reviewed a sampling of CRs to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment. The CRs reviewed were:

- CR 07-18074, which addressed the operability of high pressure injection train 1;
- CR 07-20803, which addressed 4.16 KV essential bus D1 minimum bus voltage for degraded voltage relays; and
- CR 07-21914, which addressed an oil leak from the inboard motor bearing housing for decay heat pump 2.

This review represented three inspection samples.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing activities associated with scheduled and emergent work activities:

- testing following the eight year preventive maintenance task of replacing the solenoid valve for NN236 (containment isolation nitrogen valve) on April 27, 2007;
- test of control room emergency ventilation system train 1 following preventive maintenance activities on May 10, 2007;
- test of safety function actuation system (SFAS) channels 1 and 3 following module replacement for output relay 3K25A for CS1530 (containment spray discharge valve) on May 17, 2007;
- test of SFAS channel 4 output module relay replacement for containment air cooler 2 slow speed start on May 22, 2007; and
- quarterly test of component cooling water pump 2 and associated motor on May 23, 2007, after motor and pump preventive maintenance.

The reviews were conducted to allow the inspectors to determine if the testing was adequate for the scope of the maintenance work performed. The inspectors reviewed the acceptance criteria of the tests to ensure that the criteria was clear and that testing demonstrated operational readiness consistent with the design and licensing basis documents. Documents reviewed during this inspection are listed in the Attachment.

This review represented five inspection samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed the surveillance test or evaluated test data to determine if the equipment tested met TS, USAR, and licensee procedural requirements, and also demonstrated that the equipment was capable of performing its intended safety functions. The inspectors reviewed the documents listed in the Attachment. The inspectors also determined if the test met the TS frequency requirements; if the test was conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; if the test acceptance criteria were met; and if the results of the test were properly reviewed and documented. The following surveillances were evaluated:

- DB-SP-03338, containment spray train 2 quarterly pump and valve test on April 24, 2007;
- DB-ME-03046, D1 under voltage relay monthly functional test on May 5, 2007;
- DB-MI-03201, channel functional test and calibration of steam-feedwater rupture control system (SFRCS) actuation channel (ACH)1 pressure inputs on May 8, 2007; and
- DB-OP-01101, quarterly containment entry and inspection on May 19, 2007.

This review represented four inspection samples of which one was a quarterly inservice testing (IST) inspection sample.

b. Findings

Improper Implementation of Independent Verification Requirements in Performance of Instrumentation and Control (I&C) Surveillance Test Procedures for TS Required Mitigation Systems

Introduction: A Green Non-Cited Violation (NCV) of TS 6.8.1 was identified by the NRC regarding adherence to the requirements for independent verifications required by safety-related surveillance procedures for I&C mitigation systems.

Description: On May 8, 2007, the inspectors observed I&C technicians performing safety-related procedure DB-MI-03201, "Channel Functional Test and Calibration of SFRCS ACH 1 Pressure Inputs." The procedure required independent verification of two separate valve positions in the restoration section for each of the eight pressure switches tested. The inspectors observed the technicians conduct peer checks of all steps and then, when the procedure required an independent verification, conduct what they were instructed to do for an independent verification and initialed the step in the procedure indicating completion of the independent verification. The independent verification consisted of the technician, who had performed the peer check, turning his head away from the work, then immediately turning his head back to the work, and checking the position of the valve that required independent verification.

The inspectors determined that the method of independent verification did not appear consistent with industry practices or with practices being used by operations personnel. The inspectors were told by technician supervisory personnel that the independent verification method used by the technicians was the shop practice that had been established for the technicians, that the practice was used at least during the current operating cycle, and that there were many surveillance procedures that required independent verifications to be performed by the technicians.

The inspectors reviewed licensee procedure NOBP-LP-2603, "Event-Free Tools and Verification Practices," dated July 25, 2006. The procedure stated that independent verification is "a series of actions by two individuals working independently to confirm the condition of a component, system or product quality." The procedure, in a note, added that the performer and the verifier must be separated by time and distance during the evolution. Procedure NOP-WM-4006, "Conduct of Maintenance," refers to procedure NOBP-LP-2603 as describing verification practices. Procedure NOP-LP-2601, "Procedure Use and Adherence," states that all personnel shall adhere to the requirements of the procedures that are governing the activities being performed.

The inspectors reviewed the CRs and written comments from the licensee's work activity observation program to determine if independent verification methods were reviewed. The inspectors' review encompassed approximately one year of data. The inspectors found that several instrument surveillances, that required independent verifications, were observed. The inspectors found CRs that addressed issues related to the potential non-compliance with the requirements of NOP-LP-2601 and NOBP-LP-2603; however, the inspectors did not identify any CRs or comments documenting inadequate or improper verifications within the I&C maintenance organization.

The licensee provided the inspectors CR 03-10925 dated December 11, 2003, and CR 04-03043 dated April 29, 2004. Both CRs questioned the methods used by technicians for doing independent verifications. The investigation summary for those CRs stated that independent verification did require physical separation sufficient to preclude the direct observation of the initial activity that was to be verified, or sufficient physical separation to prevent preconceived ideas about equipment status. Both CRs, with the procedural requirements that existed at the time, stated that the department performing the work was responsible for determining the specific verification requirements. Condition report 03-10925 stated that exceptions to physical separation may be permitted by the Maintenance Supervisor on a case-by-case basis.

The licensee, after discussion with the inspectors, documented the issue in CR 07-21258. In this CR, the licensee stated that the independent verification method used by technicians was a long-standing practice based on management guidance that was given after peer checking became a common practice. The condition report also stated the long-standing practice was not adjusted to industry changes in philosophy of verification methods. The licensee did, after initiation of the condition report, instruct technicians to perform independent verifications in accordance with the current requirements of NOBP-LP-2603.

<u>Analysis</u>: The inspectors determined that failure to modify independent verification requirements to address current industry standards and current procedural requirements was a performance deficiency warranting a significance determination in accordance with IMC 0612, "Power Reactor Inspector Reports," Appendix B, "Issue Disposition Screening," issued on November 2, 2006. The inspectors determined that the finding was more than minor because the finding was associated with the configuration control and testing procedure quality attributes of the mitigating systems cornerstone and did affect the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The improper completion of procedure-required verifications provided less than adequate assurance that important components of mitigation systems were properly positioned.

The inspectors determined that the finding could be evaluated using IMC 0609, "Significance Determination Process," Appendix A, "Determining The Significance of Reactor Inspection Findings for At-Power Situations," dated March 27, 2007, because the finding had potential impact on systems associated with primary and secondary heat removal systems. The inspectors determined that the finding was of very low safety significance (Green) because there was no actual loss of safety function and all phase 1 screening questions for the mitigating systems cornerstone were answered "no." The inspectors also determined that the finding affected the cross-cutting area of human performance (H.4(b)). The licensee's work practices did not support effective communication of the proper application of human error prevention techniques specified in instrument testing procedures, and supervisory oversight of the instrument testing work did not support proper application of the specified technique.

Enforcement: Technical Specification 6.8.1 required procedures to be established, implemented, and maintained for activities covered in Appendix A of Regulatory Guide 1.33, dated February 1978, and for surveillance and test activities of safety-related equipment. Appendix A, among other items, required a procedure for procedure adherence. The licensee had procedures addressing those requirements but contrary to those requirements, licensee personnel, on May 8, 2007, while performing a surveillance procedure on a safety-related mitigation actuation system, and other dates when equivalent instrument testing activities were performed, did not comply with the procedural requirements for conduct of independent verifications. Because the violation of requirements under TS 6.8.1 was determined to be of very low significance, and it was entered into the licensee's CAP (CR 07-21258); this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000346/2007003-01).

**Cornerstone: Emergency Preparedness**

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors monitored the licensee's emergency preparedness drill conducted on April 17, 2007. The observations included licensee preparations, evaluation of drill conduct, review of the drill critique, and the identification of weaknesses and deficiencies. Specifically, the inspectors reviewed the licensee's scenario and preparations to determine if the drill evolution was of appropriate scope to be included in the performance indicator (PI) statistics. The inspectors observed drill activities and personnel performance primarily in the technical support center. The inspectors evaluated the effectiveness of the licensee's communications, the accuracy of situation evaluations, and the timeliness of required reporting (simulated) of event-related information to the appropriate agencies. Finally, the inspectors reviewed the licensee's technical support center drill critique to determine if weaknesses and deficiencies were acknowledged and if appropriate corrective actions were identified.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

### Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the Davis-Besse Nuclear Power Station Updated Safety Analysis Report (USAR) to identify applicable radiation monitors associated with measuring transient high and very high radiation areas including those used in remote emergency assessment. The inspectors identified the types of portable radiation detection instrumentation used for job coverage of high radiation area work including instruments used for fixed area radiation monitors used to provide radiological information in various plant areas, and continuous air monitors used to assess airborne radiological conditions and work areas with the potential for workers to receive a 50 millirem or greater committed effective dose equivalent (CEDE). In addition, the inspectors identified contamination monitors, whole body counters, and those radiation detection instruments utilized for the release of personnel and equipment from the radiologically controlled area (RCA).

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Walkdowns of Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors conducted walkdowns of selected area radiation monitors (ARMs) in the Auxiliary Building to verify that they were located as described in the USAR and were adequately positioned relative to the potential source(s) of radiation they were intended to monitor. Walkdowns were also conducted of those areas where portable survey instruments were calibrated/repaired and maintained for radiation protection (RP) staff use to determine if those instruments designated “ready for use” were sufficient in number to support the radiation protection program, had current calibration stickers, were operable, and were in adequate physical condition. Additionally, the inspectors observed the licensee’s instrument calibration units and the radiation sources used for instrument checks to assess their material condition and discussed their use with RP staff to determine if they were used appropriately. Licensee personnel demonstrated the methods for performing source checks of portable survey instruments and for source checking personnel contamination and portal monitors used at the egress from the RCA.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Calibration and Testing of Radiation Monitoring Instrumentation

a. Inspection Scope

Portable survey instrument calibrations were performed at the facility by RP personnel. The inspectors interviewed involved RP personnel to determined if the methods for calibration and source checks of portable survey instruments were consistent with procedures. To assess for adequacy, the inspectors observed personnel performing source checks of selected survey instruments, personnel contamination monitors, and the Fastscan whole body counting system. The inspectors reviewed records of calibration, operability, and alarm set points of selected process radiation monitors and personnel monitoring devices. This review included, but was not limited to the following:

- certificate of calibration of J. L. Shepherd portable instrument calibrator model 78-2M with single source rod;
- certificate of calibration for small article contamination monitors (SAMs);
- certificate of calibration for Eberline radiation detection device model RM-14s;
- main steam lines monitors RE600 and RE609;
- radwaste gas outlet monitors RE1822A and RE1822B;
- containment post accident monitors RE4597AA, RE4597AB, RE4597BA, and RE4597BB; and
- station vent normal range RE4598AA, station vent accident range RE4598AB, and station vent normal range RE4598BA.

The inspectors evaluated those actions that would be taken when, during calibration or source checks, an instrument was found to be out of calibration by more than 50 percent. Those actions included an investigation of the instrument's previous usages and the possible consequences of that usage since the last calibration or source check. The inspectors also reviewed the licensee's 10 CFR Part 61 source term analyses to determine if the calibration sources used were representative of the plant source term.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and condition reports that involved personnel contamination monitor alarms due to personnel internal exposures to determine if identified problems were entered into the corrective action program for resolution. There were no internal exposure occurrences greater than 50 millirem committed effective dose equivalent that were evaluated during the inspection. However, the licensee's process for investigating this type of occurrence was reviewed to determine if the affected personnel would be properly monitored utilizing the appropriate equipment and if the data would be analyzed and internal exposures properly assessed in accordance with licensee procedures.

The inspectors reviewed corrective action program reports related to exposure of workers or to 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 determine if follow-up activities were being conducted in an effective and timely manner commensurate with its 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.

The inspectors evaluated the licensee's self-assessment activities to determine if they would identify and address repetitive deficiencies or significant individual deficiencies observed in problem identification and resolution.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 Radiation Protection Technician Instrument Use

a. Inspection Scope

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

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.6 Self-Contained Breathing Apparatus (SCBA) Maintenance/Inspection and User Training

a. Inspection Scope

The inspectors reviewed the status, maintenance and surveillance records of selected self-contained breathing apparatuses staged and ready for use in the plant and assessed the licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room during emergency conditions. The inspectors determined whether control room operators and other emergency response and radiation protection personnel were trained and qualified in the use of self-contained breathing apparatus including personal bottle change-out. The inspectors also reviewed the training and qualification records for selected individuals on each control room shift crew and selected individuals from each designated department that were currently assigned emergency duties, including onsite search and rescue to determine if an adequate number of personnel were qualified for emergency response activities.

The inspectors reviewed the self-contained breathing apparatus manufacturer's maintenance training certifications for licensee personnel qualified to perform self-contained breathing apparatus maintenance on vital components (regulator and low pressure alarm). The inspectors reviewed maintenance records for several self-contained breathing apparatuses designated as "ready for service." The inspectors verified that maintenance was performed by qualified personnel over the past five years. The inspectors also determined if the required, periodic air cylinder hydrostatic testing was current and documented. The inspectors also evaluated if the licensee's maintenance procedures were consistent with the self-contained breathing apparatus manufacturer's maintenance manuals.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

**Cornerstone: Initiating Events**

The inspectors sampled licensee submittals for the PIs listed below for the period from June 2006 through May 2007 to verify the accuracy of the PI data reported during that period. Performance indicator definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

- Scrams With Loss of Normal Heat Removal; and
- Unplanned Transients per 7000 Critical hours.

The inspectors reviewed portions of operating logs, licensee event reports (LERs), and inspection reports for consistency with the PIs reported values.

This review represented two inspection samples of the PIs listed above.

b. Findings

No findings of significance were identified.

a. Inspection Scope

**Cornerstone: Barrier Integrity**

The inspectors sampled the licensee's PI submittals for the periods listed below. The inspectors used PI definitions and guidance contained in Revision 4 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The following PI was reviewed:

- Reactor Coolant System Specific Activity

  The inspectors reviewed Chemistry Department records and selected isotopic analyses from January 2006 through June 2007 to determine if the greatest Dose Equivalent Iodine (DEI) values obtained during those months corresponded with the values reported to the NRC. The inspectors also reviewed selected DEI calculations to verify that the appropriate conversion factors were used in the assessment. Additionally, the inspectors observed a chemistry technician obtain and analyze a reactor coolant sample for DEI to determine if there was adherence with licensee procedures for the collection and analysis of reactor coolant system samples.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Review

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's CAP. This screening was accomplished by reviewing documents entered into the CAP and review of document packages prepared for the licensee's daily Management Alignment and Ownership Meetings.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a significant safety issue not identified by the licensee. The review was focused on repetitive equipment issues, but also considered the results of daily CAP item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The review included the six-month period from December 2006 through May 2007; the Davis-Besse Fleet Oversight Quarterly Performance Report (first quarter 2007); Site Roll-Up Integrated Performance Assessment (May 2006 through December 2006); and issues documented in the licensee's system health reports, maintenance rule committee meeting minutes for 2007, and other documents prepared for the daily management meeting.

This review represented one semiannual trend review sample.

b. Assessment and Observations

No findings of significance were identified. The inspectors determined that the licensee's implementation of trending was adequate. The inspectors compared the licensee's process results with the results of the inspectors' daily screening and did not identify any discrepancies or potential trends that were not currently captured in the CAP or other licensee generated documents.

.3 Annual Sample: Review of Operator Workaround Program

a. Inspection Scope

The inspectors reviewed the licensee's program for identifying, assessing, and correcting conditions that required operators to perform more steps to accomplish an activity than would be required by the plant and system design or by the plant's procedures. The inspectors determined whether the licensee was identifying operator workaround problems at an appropriate threshold and was entering identified issues into the CAP. The inspectors reviewed the items that were identified as workarounds. Included in the review was the consideration of the timeliness of correction of the workarounds, the operability of the system impacted by the workaround, and potential extent of condition. Additionally, the inspectors reviewed documents that provided direction for identifying and correcting operator workarounds.

b. Assessment and Observations

Licensee Work Process Guideline 2 (WPG-2), "Operations Equipment Issues," Revision 06, dated December 3, 2003, provided "a method to identify, evaluate, report and track plant equipment and support equipment deficiencies that significantly impact routine plant operations or could affect the plant during abnormal or emergency situations." Additionally, the guideline provided a definition of a control room deficiency, an operator workaround level 1, and an operator workaround level 2. This guideline was not classified as a quality or a non-quality procedure on the cover sheet and was not listed in the licensee's data base that identifies classification of procedures. Discussion with licensee personnel indicated that this guideline was the last remaining document from a set of guidelines and would eventually be cancelled after development of a replacement document.

Procedure NOP-WM-1003, "Nuclear Maintenance Notification, Screening, and Minor Deficiency Monitoring Process," supplemented guideline WPG-2 by providing guidance for assigning priorities for work orders that addressed operator workarounds and control room deficiencies. Classification of workarounds and priority assigned to work orders appeared consistent with the requirements contained within existing guidance and procedures.

The inspectors noted that the daily-reviewed operations shift turnover documents contained listings of control room deficiencies and operator workarounds. As a minimum, the deficiencies and operator workarounds, level 1 and level 2, were mentioned during the first shift turnover meeting of an operations shift that returned from days off of shift work. In addition to those reviews, operations management reviewed the status of non-outage control room deficiencies and operator workarounds with management representatives from engineering, maintenance, and work control. Those reviews were normally scheduled weekly. Inspectors' review of operator workarounds and control room deficiencies did not identify any items as improperly classified or that appeared to be scheduled inappropriately.

c. Conclusions

No findings of significance were identified. The licensee's program provided a means for identifying and prioritizing operator workarounds, highlighting the items to plant management, and tracking the items until they were corrected.

This review represented one inspection sample.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 05000346/2006-004-00 and LER 05000346/2006-004-01: Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures Davis-Besse Nuclear Power Station, Unit No.1

On December 15 and 16, 2006, for an extent-of-condition evaluation associated with questions on the emergency diesel generator ventilation systems, the licensee performed additional evaluations of ventilation systems required for safe shutdown in the event of a tornado. The licensee determined that dampers for the Low Voltage Switchgear rooms (LVSGR) and Component Cooling Water (CCW) room could be overstressed during the depressurization associated with a design basis tornado. The Licensee documented the problem in CR 06-11269.

Compensatory actions were put in place to ensure that, upon issuance of a Tornado Watch, dampers and doors were positioned such as to minimize the differential pressure across the affected dampers. Those actions were placed in the tornado off-normal procedure on January 26, 2007. Those actions were documented in the original submittal of the LER.

Revision 1 to LER 05000346/2006-004, submitted on May 9, 2007, provided information about study calculations created to support the removal of actions from the tornado off-normal procedure. The study calculations performed evaluations of the 24-hour time period following a tornado that damages the CCW room and LVSGRs dampers. The evaluations determined that the temperature in the CCW room and LVSGRs would stabilize within acceptable limits. The equipment, necessary for safe shutdown, in these rooms would operate for 24 hours following a tornado event without any compensatory measures. The licensee determined that 24 hours following a design basis tornado event should be sufficient time for diagnoses and corrective actions using existing and/or portable ventilation equipment. On June 1, 2007, the tornado off-normal procedure was revised to remove these actions associated with the CCW Pump room and LVSGR ventilation dampers.

LER 05000346/2006-004-00 and revision 01 to that LER are closed.

This review represented two inspection samples.

.2 <u>Loss of Cooling to the Condensate Pump Motors Upper Radial and Thrust Bearings</u>

a. <u>Inspection Scope</u>

The inspectors responded to a loss of cooling to the condensate pump motors 1 and 2 upper radial and thrust bearings on May 19, 2007. The loss of cooling was due to an apparent blockage in the cooling water common return line for the condensate pump motors. The loss of cooling caused temperatures of these upper motor bearings to increase rapidly. The inspectors observed licensee personnel take actions in directing activities to mitigate the event. These actions included reducing reactor power to approximately 41 percent and removing condensate pump #1 from service prior to exceeding bearing temperature limits. Additionally, the inspectors performed a walkdown of the temporary modification installed by the licensee to restore cooling to the condensate pump motors upper radial and thrust bearings.

This review represented one inspection sample.

b. <u>Findings</u>

No findings of significance were identified.

## .3 Licensed Reactor Thermal Power Exceeded During Normal Plant Operations

### a. Inspection Scope

The inspectors reviewed the conditions leading to and operator response to a condition where reactor thermal power exceeded the licensed thermal power limit for an eight-hour average.

This review represented one inspection sample.

### b. Findings

Introduction: A self-revealing NCV of the plant operating license was identified during normal plant operations when on June 8, 2007, control room personnel observed that the plant's computer was not scanning reactor coolant letdown flow after work was performed to upgrade computer programs. Letdown flow was a variable used in the computer's calculation of reactor core power. As a result, the licensee operated the plant in excess of 100 percent power when averaged over 8 hours, during a period of approximately 15 hours.

Description: On June 7, 2007, at 1440 hours, the licensee commenced a plant computer software and calculations update. The licensee declared their computer report group 38 inoperable and, in accordance with existing plant procedures, took action to maintain power level below 100 percent power. The licensee's computer group 38 provided a display of several variables associated with core power and core power distribution and also displayed calculated core power. The core power calculation involved several plant parameters including feedwater flows, feedwater temperatures, and reactor coolant cleanup letdown flow. The control room operators, when group 38 was operable, maintained core power level at or below 100 percent power using the calculated core power number displayed on group 38.

On June 7, 2007, at 1610 hours, licensee personnel declared computer group 38 operable after completion of the computer software upgrades and completion of verifications that group 38 was responding and updating in response to measured plant parameters. On June 8, 2007, at 0720 hours, licensee personnel, using another regularly displayed computer group (computer report group 22) determined that the plant computer was not recording or displaying the actual reactor coolant cleanup letdown flow. Licensee personnel determined that the computer point associated with the letdown flow had not been returned to scan or active status after the computer software upgrade. Letdown flow during this time period was approximately 25.5 kilo-pounds per hour (approximately 50 gallons per minute); the computer used a value of zero letdown flow in the calculation for core power.

The licensee determined that not having letdown flow included in the core power calculation caused the calculated core power to display 0.15 percent low. They also determined that during the approximate 15 hours that calculated core power did not consider letdown flow, actual core power, when averaged over 8 hours, exceeded 100 percent. The highest value calculated by the licensee was 100.06 percent. After discovery, licensee returned the computer point associated with letdown flow to an operating status and verified that the computer groups were functioning properly

Analysis: The inspectors determined that, in accordance with Appendix B of IMC 0612, failure to maintain the reactor thermal power 8-hour average below 2772 megawatts-thermal (MWt), as required by the plant Operating License, was a performance deficiency that was considered greater than minor because it could affect the fuel cladding barrier. Thus, it degraded the barrier integrity cornerstone objective and was associated with the cornerstone attributes of thermal limits and reactivity control. The inspectors evaluated the significance of this finding using IMC 0609, Appendix A, Phase 1, where findings affecting only the fuel cladding screen out as green or of very low safety significance. This finding was also associated with the cross-cutting area of human performance (H.3(b)) because in the work control process, the operational impact of computer work activities that affected calculated core power, was not appropriately considered.

Enforcement: Condition 2.C.(1) of the Davis-Besse Operating License as Amended states “FENOC is authorized to operate the facility at steady-state reactor core power levels not in excess of 2772 megawatts (thermal).” Contrary to this, an 8-hour average thermal power exceeded 2772 MWt by about 0.06 percent during the period when reactor core power calculations did not account for heat loss due to reactor coolant cleanup letdown flow. Exceeding the power limitations specified in the plant Operating License is a violation. Since this violation is of low safety significance, it is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000346/2007003-02). Once it was determined that calculated core power was not considering coolant letdown, licensee personnel took action to rectify the issue and entered this issue into the CAP (CR 07-21802).

### .4 Nitrogen Intrusion into the Safety Injection System

#### a. Inspection Scope

On April 10, 2007, the licensee performed a six-month preventive maintenance task to ultrasonic test the high pressure injection (HPI) piping to verify that the piping was “full with water.” The licensee discovered voiding in approximately 90 feet of the train 1 HPI discharge piping. The licensee declared the system inoperable, vented and filled the piping and formed a problem solving team. The inspectors reviewed the problem solving team’s charter, cause identification efforts, and performed a walkdown of the system. The inspectors also reviewed past condition reports and core flood tank fill rates and performed independent calculations to confirm the extent of condition (length of voided piping and estimated volume of air). Finally the inspectors assessed the licensee’s planned corrective actions and independently confirmed that the pressure necessary to introduce gas into the HPI piping was higher than the pressure at which the licensee’s compensatory actions would require venting.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.5 Licensee Entry into TS Action Statements for Tornado Warning

a. Inspection Scope

Because of questions that existed on the structural viability of several ventilation dampers when subjected to design basis tornado differential pressure loadings, the licensee developed compensatory actions to be taken in the event of a tornado warning. The actions essentially consisted of placing the suspect dampers in a position that would not subject them to full tornado-generated differential pressures. On May 15, 2007, the licensee received an unanticipated notification that a tornado warning was issued for Ottawa County, Ohio, which is the location of the plant. This caused the licensee to declare their emergency diesel generators and component cooling water pumps inoperable until the compensatory actions were taken. The associated TS action statements required the licensee to commence a plant shutdown within one hour if the tornado warning was not cancelled or the compensatory actions remained incomplete by that time. Changing weather conditions and completion of compensatory actions negated the need for a shutdown. The inspectors reviewed the licensee's response to the event and compliance to their procedures.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Licensee Activities and Meetings

The inspectors observed select portions of licensee activities and meetings. The activities that were sampled included:

- Station ALARA [as low as reasonably achievable] Committee meeting on May 4, 2007;
- planning meeting on plant power reduction for condenser inspection and repair on May 4, 2007; and
- Davis-Besse supervisory briefing on June 25, 2007.

Additionally the licensee met with the inspectors on June 28, 2007, to discuss the status of efforts associated with analyzing and modifying masonry block walls. The licensee advised the inspectors that analyses had been completed and had identified the desirability of modifying some walls and wall attachments. The licensee estimated that all modifications, except one that may require a plant outage, should be completed in 2007.

No items of significance were identified.

.2 Out of Service Seismic Force Monitoring Equipment Affecting Emergency Plan Response

a. Inspection Scope

The inspectors performed a review of activities associated with seismic peak recorders that had failed calibration checks. The inspectors questioned the status of the seismic monitoring equipment and were informed by the licensee that they were in compliance with the technical requirements manual for seismic equipment. On April 9, 2007, the inspectors observed the seismic monitoring cabinet to be turned off and determined that the equipment was in that status since March 29, 2007. The inspectors then reviewed the off-normal instruction for an earthquake and emergency classifications.

b. Findings

Introduction: An NCV of very low safety significance (Green) was identified associated with the emergency classification assessment capability for Alert or Site Area Emergency during, and following, a major earthquake event. The inspectors determined that the licensee had failed to establish compensatory measures to ensure the prompt implementation of the Davis-Besse Emergency Plan, as required by 10 CFR 50.54(q) and the risk significant planning standard (RSPS) found in 10 CFR 50.47(b)(4).

Description: The inspectors noted through condition report review that Davis-Besse had removed the seismic monitoring cabinet from service on March 29, 2007, for calibration checks. The licensee determined that the peak recording accelerometers were out of calibration and replacement parts were not on hand. The seismic monitoring cabinet remained out of service until April 10, 2007, when it was returned to service after questioning by the inspectors.

Procedure RA-EP-01500 (Davis-Besse Emergency Plan Implementing Procedure entitled “Emergency Classification”), Section 8.A.2 and 8.A.3 required operators to determine the magnitude of an earthquake (to classify the event in accordance with the emergency plan) using the seismic monitoring cabinet in the main control room. When the seismic monitoring cabinet was removed from service for calibration and replacement of parts on March 29, 2007, the operators were not provided with direction or training to implement the emergency plan with respect to assessing the magnitude of a seismic event without the seismic monitoring cabinet available. The inspectors determined that without the seismic monitoring panel that assessment of earthquake conditions could not be promptly performed for an Alert or Site Area Emergency. This resulted in the decrease in the ability of the emergency director to appropriately assess a seismic event per the emergency action levels (EALs) and a reduction in the effectiveness of the emergency plan.

The shift manager, in the event of an emergency, initially functions as the Emergency Director for emergency plan purposes. The inspectors questioned the shift manager on April 9, 2007, about the loss of assessment capability for determining an Alert or Site

Area Emergency for earthquake conditions per the emergency plan classification procedure. The shift manager discussed the issue with plant management and at 1517 on April 9, 2007, the shift manager made an 8-hour non-emergency notification to the NRC headquarters for a loss of the seismic monitoring system capability to assist in determining the magnitude of an earthquake (event notification #43292).

Davis-Besse basis for EALs was NUREG-0654, Revision 1, Appendix 1. The EALs affected by lack of seismic instrumentation were Table 8.A of Davis-Besse Emergency Plan Implementing Procedure, Revision 06:

- Natural Events (Within Ottawa County) (All Modes) Earthquake 1., Unusual Event, "Any earthquake felt in-plant OR detected by station seismic instrumentation";
- Natural Events (Within Ottawa County) (All Modes) Earthquake 2., Alert, "Ground motion felt **AND** OBE alarm on seismic alarm panel C5764A"; and
- Natural Events (Within Ottawa County) (All Modes) Earthquake 3., Site Area Emergency, "Ground motion felt **AND** SSE alarm on seismic alarm panel C5764A."

After questioning by the inspectors, Davis-Besse placed compensatory actions in their unit operating logs for operators to determine if an earthquake met the Alert or Site Area Emergency action levels. They restored assessment capability by placing the seismic monitoring panel in service in the main control room at 0639 on April 10, 2007. The licensee placed this issue into their CAP (CR 07-18003).

Analysis: Failure to provide compensatory actions for the timely implementation of the Davis-Besse Emergency Plan for "Natural Events (Within Ottawa County) (All Modes) Earthquake" is a performance deficiency warranting a significance determination. The finding was more than minor because it was associated with response organization planning standards attribute of the emergency preparedness cornerstone and did affect the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Additionally, the finding, using example "4i" and "4j" of IMC 0612, Appendix E, "Examples of Minor Issues," dated June 6, 2006, was not minor because the issue would affect the licensee's timely response to a serious earthquake event.

The finding is of very low safety significance because it did not result in a failure or degradation of the RSPS, and the unavailability of the seismic monitor did not prevent the declaration of an Alert or Site Area Emergency classification. Other seismic instrumentation was available for the period of March 29 thru April 10, 2007, that would permit the licensee classification process to make an appropriate classification, although the classification could have been delayed beyond a 15-minute period. This finding was also associated with the cross-cutting area of human performance. Licensee's work control process failed to establish compensatory measures for the out-of-service duration of the seismic force monitor (H.3(a)).

Enforcement: In accordance with 10 CFR 50.54(q), the licensee shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b). In accordance with 10 CFR 50.47(b)(4) a standard emergency classification and action level scheme shall be in use by facility licensees. State and local response plans call for reliance on information provided by facility licensees for the determination of minimum initial offsite response measures. Contrary to the above requirements, the licensee failed to establish compensatory measures to ensure the prompt implementation of the Davis-Besse Emergency Plan. However, since the finding was of very low safety significance (Green) and the licensee entered this issue into their CAP (CR 07-18003), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2007003-03)

.3 Confirmatory Order Related Activities (95003)

On March 8, 2004, the NRC issued "Confirmatory Order Modifying License No. NPF-3 (EA-03-0214)," which required, in part, that the licensee perform annual independent assessments, for a period of five years, in the areas of operations performance; organizational safety culture, including safety conscious work environment; CAP implementation; and engineering program effectiveness. The following activities associated with EA-03-0214 were reviewed during this inspection period.

a. Calendar Year (CY) 2007 Operations Performance Independent Assessment Plan

(1) 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 (EA 03-214), the inspectors verified that the licensee submitted, in letters dated March 13, 2007, and April 12, 2007, the required inspection plan for the Operations Independent Assessment prior to the performance of the CY2007 annual Operations Assessment. The assessment was scheduled for June 2007. As part of the inspection activities, the inspectors reviewed the scope of the Independent Assessment Plan and the qualifications of the team members designated to perform the assessment.

(2) Findings and Observation

After evaluating the Operations Performance Independent Assessment Plan for CY2007, the inspectors determined that the scope and depth of activities outlined in the plan would be sufficient to obtain an appropriate assessment of operations department performance.

The inspectors evaluated the qualifications of the assessment team members and concluded that the individuals designated to perform the assessment were independent from FENOC and possessed the necessary expertise to accomplish the assessment, as outlined in the assessment plan.

b. Observation of CY2007 Operations Performance Independent Assessment

(1) Inspection Scope

The independent assessment of operations performance for CY2007, required by Confirmatory Order EA 03-0214, was conducted on-site from June 11 to June 22, 2007. The inspectors evaluated the on-site activities. In particular, the inspectors attended licensee debriefs, monitored in-process evaluations, and discussed preliminary findings with assessment team members. Additionally the inspectors observed independent assessment activities to determine the effectiveness of the assessment and the potential impact, if any, of the unexpected unavailability of one of the team members for part of the assessment.

(2) Findings and Observations

No findings of significance were identified.

The inspectors concluded that the assessment team conducted all activities prescribed by the Operation Performance Assessment Plan. A final discussion of the results of the independent assessment of operations was planned for July 6, 2007, after the end of the inspection period. The March 8, 2004, Confirmatory Order required that the licensee provide the NRC Region III Administrator with all assessment results and actions planned to address the assessment results, within 45 days of the completion of the independent assessment and the final debrief.

c. CY2007 Engineering Program Effectiveness Independent Assessment Plan

(1) 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 verified that the licensee submitted the required inspection plan for the Engineering Program assessment for CY2007. The licensee submitted its plan 90 days prior to the performance of the assessment (start date of September 10, 2007) in a letter to the NRC dated June 12, 2007. The inspectors reviewed the licensee's letter describing the assessment plans and evaluated the scope and depth of the plans, including the credentials, experience, objectivity, and independence of the designated assessors.

(2) Findings and Observations

The inspectors verified that the individuals designated to perform the assessment were independent from FENOC and that they brought the appropriate credentials and experience necessary to accomplish the assessment. The plan included six team members for a period of two weeks. Three of the team members have participated in the 2005 and 2006 assessments (Marathon Consulting Group), the other three are on loan from Florida Power and Light, Entergy Northeast, and Constellation Energy. The purpose of the plan was to provide an independent and comprehensive assessment of the Engineering Program effectiveness. The plan included details to assess Engineering effectiveness in the following areas:

- Plant Modification Process;
- Calculation Process;
- System Engineering Programs and Practices;
- CAP Implementation;
- Corrective actions taken in response to the seven Areas in Need of Attention (ANAs) identified during the 2006 Independent Assessment of the Davis-Besse Engineering Program Effectiveness; and
- Self Assessment Effectiveness.

The scope and depth of the proposed plan appeared adequate to accomplish the objective of assessing Engineering Program effectiveness. The NRC inspectors will observe portions of the on site assessment activities and attend the exit meeting at the conclusion of on site activities. The NRC will review the team report when it becomes available.

d. Safety Culture/Safety Conscious Work Environment (SC/SCWE) Independent Assessment, CY2006

(1) Inspection Scope

The inspection team observed work activities and reviewed documents to assess the licensee's implementation of the March 2004 Confirmatory Order as it applied to the annual external, independent evaluations of safety culture (SC) and safety conscious work environment (SCWE). In assessing the licensee's 2006 activities, the team observed the licensee's implementation of its Business Practice, NOBP-LP-2501, Rev. 3, for assessing SC and SCWE. In addition, the team observed one of the external independent contractor's meetings when they were discussing input from interviews of selected staff members. The team also reviewed the results of the licensee's SCWE survey and the contractor's survey methodology and final report submitted to the NRC by FENOC letter dated February 2, 2007, (ML070520652). The February 2 letter also provided an Action Plan to address Areas for Improvement (AFIs) identified during the assessment.

Further, the team reviewed selected condition reports associated with the individual AFIs to evaluate the licensee's corrective actions.

(2) Findings and Observations

a) Actions for AFIs.

The team concluded that the licensee's Action Plan had adequately addressed all AFIs identified in the SC/SCWE independent assessment report. The AFIs were:

(1) "The Nuclear Plant Systems Engineering organization and the Nuclear Warehouse organization provided ratings of Not Effective for

the NSC (Overall Nuclear Safety Culture), NS VB&P (Nuclear Safety Values, Behaviors & Practices), SCWE (Safety Conscious Work Environment) and ECP(Employee Concerns Program) key cultural metrics;"

(2) "The Engineering Programs organization provided ratings of Marginally Effective for the Overall NSC, NS VB&P and ECP key cultural metrics. The approximate trends for the ratings of the Overall NSC and the SCWE were Very Significantly Declined;"

(3) "The DBNPS Site Composite Organization rating of the NS VB&P attribute 'Functional Organization staffing levels are consistent with the demands of maintaining Nuclear Safety and safe plant operations' was Not Effective. Thirteen individual DBNPS Functional Organizations also provided low ratings of the 'Adverse Effects of Workload on Nuclear Safety' metric: eight were Not Effective and five were Marginally Effective. These organizations are identified in Section IV.B.12. These low ratings represent indicators of localized staffing, workload and/or workload management related issues that are perceived to be having an adverse impact on Nuclear Safety performance in those organizations;"

(4) "The DBNPS Site Composite Organization rating of the NS VB&P attribute 'Appropriate levels of oversight and control of contractor work activities are provided to ensure that Nuclear Safety is maintained' was Not Effective. Other sources of information available to the Assessment Team confirmed that oversight and control of contractor work activities during plant outages is perceived by many to be a significant area of concern;"

(5) "The DBNPS Site Composite Organization rating of the NS VB&P attribute 'Site funding levels are consistent with the demands of maintaining Nuclear Safety and safe plant operations' was Not Effective. Other sources of information available to the Assessment Team indicate that this low rating represents, at a minimum, a significant communications issue" and

(6) "The DBNPS Site Composite Organization rating of the SCWE attribute 'Performance reviews, financial rewards, promotions, personnel recognition and personnel sanctions foster and reinforce attitudes and behaviors that are consistent with a strong Nuclear Safety Culture' was Not Effective. Other sources of information available to the Assessment Team indicate that the breakdown (real or perceived) of the DBNPS performance appraisal process after RFO 14 is likely to have significantly contributed to this low rating."

b) Review of Condition Reports Associated with Areas for Improvement

(1) All CRs were categorized as limited apparent cause assessments. Based on interviews with evaluators, this categorization may have limited the evaluation they performed. For example, one AFI dealt with the staff's perception of budget over safety. The evaluation stated that the budget had been benchmarked; however, the appropriateness of the benchmarked site was not reviewed. In addition, the issue of staffing versus safety wasn't reviewed from a parent organization perspective as to whether the parent organization was providing adequate staffing for the sub-group. This condition may have limited the effectiveness of the licensee's corrective actions.

(2) For AFIs involving "trust" and "change management," no specific corrective actions were identified because, as stated in the CR, nothing would make an immediate improvement in the condition. The team noted that the evaluation was performed by individuals with no experience in human performance or human response and there is no indication that individuals with those skills were contacted to assist in the evaluation. With no corrective actions being implemented, the team questioned whether significant improvements in this area would be realized. Also, the team questions whether "immediate improvement" should be the criteria for the CAP.

(3) In a couple of instances, the independent contractor had not provided specific detailed information germane to the issue(s) thereby limiting the ability of the licensee to assess the condition.

c) Contractor Activities:

(1) Observation on Contractor's daily Debrief

Based on team members' observation of one meeting held by the contractor to assess the information the contractor had gained during interviews with individual licensee staff members, the team concluded that the assessment and integration of the interview information was thorough, productive and appropriate.

(2) Survey Methodology

The staff reviewed the contractor's survey instrument and compared the results of the survey with the results of both the FENOC Employee Concerns Program (ECP) SCWE survey and the FENOC business practice for internally evaluating SC/SCWE. The comparison between the ECP SCWE survey and the contractor's results indicated agreement in a number of areas; however, there were areas where the ECP survey results were not mirrored by the contractor's results. Regarding the FENOC business practice, the staff found that the business practice still has a number of the same weaknesses that existed in the past, (refer to section d below), so a comparison with the contractor’s survey would not result in any meaningful information.

The staff confirmed that all employees at Davis-Besse, including those on loan to other FENOC sites were given the opportunity to participate in the survey. In addition, the staff confirmed that the survey covered areas that were associated with SC/SCWE. However, the staff questioned the reliability of the contractor’s survey tool because it was not developed using standard scientific methods for survey design and testing. The inspectors concluded that the failure to use an industry accepted methodology resulted in several weaknesses, for example:

- The staff could not easily link questions to specific safety culture components;
- The rating scale used is positively biased rather than being neutral;
- Many of the questions were complicated and complex, which leads to confusion;
- Questions that were added to the survey tool for this assessment were not tested prior to inclusion; and
- No statistical analysis was used to assure the reliability or validity of the survey tool.

In addition, the contractor’s typical process did not use multiple independent tools to measure the same process to see if there was convergence of the results. For the 2006 Davis-Besse assessment, the contractor included some interviews and document reviews; however, the organizations from which individuals were selected were based on the survey results, and the sampling methods were unclear and not statistically based. Also, in reviewing the contractor’s report, it was not possible to draw specific relationships between the AFIs and the survey results. Without this linkage, the basis for corrective actions is questionable.

Because the external survey was not developed using industry-recognized techniques, the staff was concerned with its use as a stand alone methodology for assessing safety culture and questioned its results. The staff used the licensee’s internally-developed Employee Concerns Program SCWE survey results as an independent check against the external survey results. The Employee Concerns Program (ECP) SCWE surveys have been in good agreement with previous external assessments. Further the 2006 and 2005 ECP SCWE surveys were consistent. Because the findings of the contractor’s survey and the FENOC ECP SCWE survey exhibited a number of similarities, the staff concluded that the results of the contractor’s assessment were reasonable and FENOC had met its obligation under the Order.

As a point of clarification, it should be noted that the team’s concerns with the contractor’s survey tool is one of confidence in the results, not that the results are necessarily unacceptable. An instrumentation analogy would be confidence in test data. If instrumentation is properly calibrated, there is a high level of confidence in the data obtained from the instrument; however, if the instrument is not calibrated the level of confidence is much reduced, even though the data may be correct.

d) Review of Business Practice NOBP-LP-2501, Rev. 3, September 29, 2006

Based on the review and observation of the implementation of the business practice, the team’s conclusion is the same as documented in previous SC/SCWE reviews. The process of conducting a meeting with site managers and directors discussing issues common to all organizations is beneficial; however, the Business Practice defined process and individual evaluation criteria have many of the same deficiencies previously identified with its parent process (IR 05000346/2006012, ML040580673). For example:

- The mathematical averaging of individual ratings without using weighting criteria did not account for the difference in significance in items or organizations.
- A number of evaluation criteria appeared not fit the item to be measured, e.g., using the number of meetings held by a supervisor to assess the effectiveness of the supervisors communications skills.
- A number of evaluation criteria appeared to be inappropriate, e.g., in assessing critical safety functions of the decay heat removal system - the level at which action is required to be taken is only after the system has been lost with “significant reactivity or core impact.”

In addition to the process issues, the team observed, on more than one occasion, that individuals made comments that the criteria were forcing the evaluation towards RED. The inference taken away by the team member was that some licensee managers did not see the process as a way to identify weakness and correct them, rather, the objective was to have criteria which kept the evaluations in the WHITE or GREEN areas where corrective actions were not required.

e) Employee Concerns Program SCWE survey.

The team noted a continued decrease in the number of negative response in the results from the licensee’s internal SCWE survey. While a number of organizations continue to have greater than ten percent negative responses in a number of areas, the number of negative responses has declined.

(3) Conclusion

Overall, the inspectors concluded that the SC/SCWE at Davis-Besse continued to be adequate to support continued safe facility operation and that corrective actions were being effective. In addition the team concluded that the licensee had met requirements contained in the NRC's March 8, 2004, letter, "Approval to Restart the Davis-Besse Nuclear Power Station, Closure of Confirmatory Action Letter, and Issuance of Confirmatory Order" in the SC/SCWE area. However, because of the deficiencies identified in the development and assessment of the contractor's survey tool, the team is concerned with the tool's use as a stand alone assessment of SC/SCWE.

.4 Follow-up Inspection Activities Associated with the Reactor Pressure Vessel Head Degradation Issue

Davis-Besse was shutdown on February 16, 2002 for a refueling outage. During scheduled inspections of the control rod drive mechanism nozzles, significant degradation of the reactor pressure vessel (RPV) head was discovered. As a direct result of the need to resolve many issues surrounding the degradation, NRC management implemented Inspection Manual Chapter (IMC) 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems." The fuel was subsequently removed from the reactor, and the plant remained shut down until NRC issued "Approval to Restart" on March 8, 2004.

Based upon the discovery of the RPV head degradation, the NRC issued Confirmatory Action Letter (CAL) 3-02-001 to Davis-Besse documenting six commitments required to be accomplished prior to restart of the reactor. The NRC also chartered an Augmented Inspection Team (AIT) to inspect the circumstances surrounding the vessel head degradation issue; the results of which were documented in NRC Inspection Report (IR) 50-346/2002-03. On October 2, 2002, the NRC issued the AIT follow-up IR 50-346/2002-08, which documented ten apparent violations of regulatory requirements.

In a February 25, 2003, letter to FirstEnergy Nuclear Operating Company (FENOC), the NRC documented a performance deficiency associated with the control rod drive penetration cracking and RPV head degradation. The performance deficiency involved FENOC's failure to properly implement its boric acid corrosion control and CAP, which allowed reactor coolant system (RCS) pressure boundary leakage to occur undetected for a prolonged period of time resulting in RPV head degradation. The NRC assessed the significance of the performance deficiency using the Significance Determination Process (SDP) and preliminarily concluded that the significance was RED. A RED finding is one with high importance to safety that will result in increased NRC inspection and other NRC action. On April 24, 2003, FENOC submitted a written response that acknowledged the performance deficiency and did not contest the RED finding.

In a letter to FENOC, dated May 29, 2003, the NRC documented its conclusion that the significance of the performance deficiency involving the control rod drive penetration cracking and the RPV head degradation was appropriately characterized as RED. The NRC also noted that the results of a then ongoing Office of Investigations (OI) investigation into the cause of the apparent violations would be a factor in the final enforcement deliberations. As a result, no Notice of Violation (Notice) was issued concurrent with the May 2003 letter.

Based upon its investigation into the causes of the apparent violations, documented in OI Investigation Report No. 3-2002-006, OI determined that the apparent violations involved the licensee's willful failure to: (1) properly implement the boric acid control program; (2) properly implement the CAP; (3) adequately remove, on several occasions, boric acid and rust deposits from the reactor head; (4) maintain the plant shutdown, i.e., not startup and return the plant to power from the twelfth refueling outage (12RFO) until boric acid deposits were removed and the reactor head was inspected, and; (5) maintain and submit to the NRC, complete and accurate information. As a result, the NRC referred the OI report to the U.S. Department of Justice (DOJ) for its review and consideration of criminal prosecution. A Deferred Prosecution Agreement Between the United States of America and FENOC was later executed on January 19, 2006.

Since the initial discovery of the head degradation and the NRC's issuance of the CAL that outlined the actions necessary for the Davis-Besse plant to restart, the NRC provided extensive oversight of the licensee's evaluation of, and corrective actions for, the conditions that contributed to the performance deficiency and head degradation. In a letter dated March 8, 2004, the NRC documented its determination that the matters contained in the CAL and Restart Checklist had been adequately resolved and that the NRC had reasonable assurance that Davis-Besse could be restarted and operated safety. Davis-Besse entered Mode 1 on March 14, 2004.

Based on information developed during the AIT follow-up inspection and OI investigation, the NRC determined that nine violations of NRC requirements occurred. The NRC issued a Notice and Proposed Imposition of Civil Penalties – $5,450,000 on April 21, 2005. The licensee responded to the Notice by letters dated July 8, 2005, September 14, 2005, and January 23, 2006.

As discussed above, during the shut down period, the NRC performed extensive inspections and reviews of the licensee's corrective actions associated with the reactor head degradation event. As such, the current review focused on the verification of these corrective actions. The following violations and associated corrective actions were reviewed for closure. As much has been docketed, this inspection effort did not attempt to capture everything; rather, the violations and significant licensee actions are discussed and applicable references cited.

a. <u>Issues Associated with Incomplete and Inaccurate Information Provided to the NRC by FENOC</u>

(1) <u>Inspection Scope</u>

The inspection of corrective actions for issues associated with FENOC's willful provision of incomplete and inaccurate information to the NRC, as described below, was the subject of NRC IR 50-346/03-19. With respect to the licensee's extent-of-condition (EOC) report, the NRC determined that these efforts provided a reasonable approach to address NRC Restart Checklist Item 3.I (Process for Ensuring Completeness and Accuracy of Required Records and Submittals to the NRC), and that the licensee had taken appropriate corrective actions to ensure that future regulatory submittals will be complete and accurate in all material respects.

The NRC inspectors reviewed CR 03-04302, Corrective Action 2, and found that the revisions to procedure NOP-LP-2001, Revision 6, “Condition Report Process” provided an enhanced discussion of the requirements for completeness and accuracy of information and independent reviews.

The inspectors also reviewed the licensee’s Root Cause Analysis Report, “Apparent Violation of 10 CFR 50.9, Completeness and Accuracy of Information,” dated April 4, 2003. This report concluded that the root causes were attributed to “less than adequate nuclear safety focus” and “less than adequate analysis of safety implications.” These issues were included in the licensee’s Management and Human Performance Improvement Plan (NRC IR Nos. 50-346/02-15, 02-18, 03-12, and 04-03). Effectiveness of corrective actions in the organizational effectiveness and human performance area was considered acceptable as documented in the NRC “Approval to Restart the Davis-Besse Nuclear Power Station, Closure of Confirmatory Action Letter, and Issuance of Confirmatory Order” dated March 8, 2004.

To prevent and detect any other similar occurrence FENOC implemented a number of corrective actions. Corrective actions included:

1) Issuing administrative procedures governing outgoing NRC correspondence and reports;

2) Performing Root Cause Analysis Report, "Apparent Violation of 10 CFR 50.9, Completeness and Accuracy of Information, CR 2002-04914" (Completeness and Accuracy Root Cause Report), April 4, 2003;

3) Performing an EOC review of the completeness and accuracy of documents based on a sample population of previously submitted NRC correspondence. FENOC's EOC review consisted of verification of the statements of fact contained in the submittals, and resolution of any discrepancies identified during the review. The review initially covered approximately 20 percent of the FENOC submittals to the NRC between January 1996 and March 2002 for the DBNPS. The sample size was subsequently expanded according to pre-established criteria after several of the initial submittals were determined to contain information that was not complete and accurate in all material respects. FENOC documented the results of its EOC Review in "Final Report: Results of the Extent of Condition Review, NRC IMC 0350 Restart Checklist Item 3.1, 'Process for Ensuring Completeness and Accuracy of Required Records and Submittals to the NRC'" (EOC Final Report), October 24, 2003.

4) Addressing discrepancies via the CAP.

5) Developing a new corporate policy and conducting site-wide supervisory awareness training of the requirements of 10 CFR 50.9.

6) Revising new employee orientation manuals and developing an initial training program requirement for all new supervisors.

7) Conducting site wide employee awareness training of the requirements of 10 CFR 50.9 including reinforcing the requirement for maintained records as well as formal regulatory submittals.

8) Revising the CR Process procedure to include requirements for completeness and accuracy.

(2) Findings and Observations

a) (Closed) VIO 05000346/2005013-02: Information included in Condition Report (CR) 2000-1037 and Work Order 00-001846-000 was not Complete and Accurate in all Material Respects.

Description

This violation concerned FENOC willfully maintaining incomplete and inaccurate information in documents required to be submitted to the NRC. The documents indicated that accumulated boric acid deposits were removed from the RPV head and that the entire head was inspected. However, the licensee did not clean or inspect the entire RPV head. The licensee's willful failure to accurately document the condition and cleanliness of the RPV head, including the willful failure to fully describe the accumulated boric acid deposits that remained on the head, is a significant violation that permitted uncorrected RCS pressure boundary leakage and boric acid corrosion of the RPV head to continue for an extended period of time. Had the NRC known of the RCS pressure boundary leakage, the NRC would have taken different regulatory actions. This willful violation of 10 CFR 50.9, 10 CFR 50, Appendix B, Criterion XVI, and 10 CFR 50, Appendix B, Criterion XVII was categorized in accordance with the Enforcement Policy at Severity Level I (Supplement VII). Civil Penalty - $110,000 (EA-05-068)

Corrective Actions Reviewed

Based on the NRC's review of the corrective actions discussed in the above Inspection Scope section, the licensee's actions were found to be consistent with the relevant docketed correspondence, and this violation is considered closed.

b) (Closed) VIO 05000346/2005013-05: Failure to Provide Complete and Accurate Information to NRC in Response to Bulletin 2001-01.

Description

This violation concerned the completeness and accuracy of information reported to the NRC in the licensee's response to the Bulletin, dated September 4, 2001, and in the licensee's supplemental response to the Bulletin, dated October 17, 2001. This violation of 10 CFR 50.9 was categorized at Severity Level I

(Supplement VII) in accordance with the Enforcement Policy. Civil Penalty - $120,000 (EA-05-072). Specific examples included:

(1) In a September 4, 2001, response to the Bulletin entitled, “Response to Bulletin 2001-01,” the licensee made the following four materially inaccurate and incomplete statements:

(a) The licensee’s response to Bulletin Item 1.c, on page 2 of 19, stated: "the minimum gap being at the dome center of the RPV head where it is approximately 2 inches, and does not impede a qualified visual inspection."

The licensee’s response was materially inaccurate, in that, the statement contradicted statements in the licensee’s documents identified as Potential Condition Adverse to Quality Report (PCAQR) Nos. 94-0295 and 96-0551, which clearly stated that inspection capability at the top of the reactor vessel head was limited. The limitation was stated to be caused by the restricted access to the area through the service structure "weep holes," the curvature of the RPV head, and by the limited space to manipulate a camera due to the insulation that creates the two inch gap.

(b) The licensee’s response to Bulletin Item 1.d, which requested inclusion of a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations, did not include a description of any limitations.

The licensee’s response was materially incomplete in that the response did not mention that accessibility to the bare metal of the RPV head was impeded, during the 11RFO (1998) and 12RFO (2000), by the presence of significant accumulations of boric acid deposits.

(c) The licensee’s response to Bulletin Item 1.d, which also requested a discussion of the findings of RPV head inspections, stated that for 12RFO, the inspection of the RPV head/nozzles indicated some accumulation of boric acid deposits.

The licensee’s response was materially incomplete and inaccurate in that it mischaracterized the accumulation of boric acid on the RPV head and did not mention the evidence of corrosion that was evidenced by the pictures and the video examination of RPV head conditions documented at the beginning and ending of the 12 RFO.

(d) The licensee’s response to the Bulletin, on Page 3, stated: "The boric acid deposits were located beneath the leaking flanges with clear evidence of downward flow. No visible evidence of nozzle leakage was detected."

The licensee’s response was materially inaccurate in that the boric acid deposits were not solely located under leaking flanges as implied by that statement and the licensee lacked clear evidence of the absence of downward flow for all nozzles. The build-up of boric acid deposits was so significant that the licensee could not inspect all of the nozzles. As a result, the licensee also did not have a basis for stating that no visible evidence of nozzle leakage was detected.

(2) In an October 17, 2001 response to the Bulletin entitled, “Supplemental Response to Bulletin 2001-01,” the licensee stated: “In May 1996, during a refueling outage, the RPV head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM [control rod drive mechanism] nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed."

The licensee’s response was materially inaccurate in that: (1) each RPV head control rod drive penetration was not inspected in May 1996, as documented in PCAQR 96-0551, and; (2) the RPV head, including the area around each control rod drive penetration, was not completely cleaned, as noted in PCAQR 98-0649, which was prepared at the start of 11RFO, which stated that there were old boric acid deposits on the head.

In FENOC’s Reply to a Notice of Violation, dated September 14, 2005, the licensee denied the violation. Specific examples supporting the licensee’s denial were not defined in the reply; however, the reply did mention that responses to the Bulletin that were misleading were a product of communication by committee.

On January 19, 2006, FENOC and the United States of America executed a Deferred Prosecution Agreement. In this Agreement, FENOC admitted that the Department of Justice could prove that from September 3, 2001, through November 28, 2001, FENOC employees, acting on FENOC’s behalf, knowingly made false representations to the NRC in the course of attempting to persuade the NRC that Davis-Besse was safe to operate beyond December 31, 2001. Subsequently, on January 23, 2006, FENOC issued a Supplemental Reply to a Notice of Violation. In this reply, the licensee reassessed its September 14, 2005, denial of this violation and amended the reply to accept the violation as stated.

Corrective Actions Reviewed

With respect to the individuals found by FENOC to be directly involved in the inaccurate information, disciplinary action up to and including termination was taken. The inspectors determined that the appropriate updates had been made to the Personnel Access Data System.

Based on the NRC's review of the corrective actions discussed here and in the above Inspection Scope section, the licensee's actions were found to be consistent with the relevant docketed correspondence, and this violation is considered closed.

c) (Closed) VIO 05000346/2005013-08: Failure to Maintain Complete and Accurate Records.

Description

This Severity Level III violation (Supplement VII) (EA-05-069) was associated with inaccurate information regarding the cleaning of boric acid from the head contained in PCAQR 98-0649, dated April 18, 1998 and PCAQR 98-0767, dated April 25, 1998.

PCAQR 98-0649, contained the following closure statement: "Accumulation of boric acid on the reactor vessel caused by leaking CRDMs [control rod drive mechanisms] has not resulted in any boric acid corrosion. This was identified through inspections following reactor vessel head cleaning in past outages....Additionally, B&W (Babcock & Wilcox) documentation discussing CRDM nozzle cracking further stated that boric acid deposits on the head caused by leaking CRDM flanges would not result in head corrosion."

However, the quoted statements were not accurate in all material respects in that the licensee had previously not cleaned all areas of the reactor head of boric acid deposits, had not inspected the base metal under all the deposits to determine whether corrosion was present, and no B&W documentation was available to support the claim that boric acid would not result in head corrosion.

PCAQR 98-0767 included the following closure justification, "The boric acid deposits were removed from the head." However, the quoted statement was not accurate in all material respects in that the licensee had not removed all of the boric acid deposits from the head as of the end of 11RFO.

Corrective Actions Reviewed

In the licensee's reply to a Notice of Violation dated September 14, 2005, it was stated that B&W documentation was, in fact, available to support the claim that boric acid from leaking CRDM flanges would not cause significant corrosion and therefore, that portion of the violation was denied. The inspectors reviewed the referenced B&W report and, though not explicitly stated as in PCAQR 98-0649, the report does conclude that at substrate temperatures above 550 degrees F, corrosion rates from boric acid were found to be very low (not measurable). The operating temperature of the Davis-Besse head is in excess of 550 degrees F.

Based on the NRC's review of the corrective actions discussed in the above Inspection Scope section, the licensee's actions were found to be consistent with the relevant docketed correspondence, and this violation is considered closed.

d) (Closed) VIO 05000346/2005013-09: Failure to Maintain Complete and Accurate Records.

Description

This Severity Level IV violation (Supplement VII) (EA-05-070) was associated with inaccurate information regarding the cleaning of boric acid from the head contained in a "Document Void Request" to cancel Modification 90-012, dated September 23, 1993, and in Quality Assurance Audit Report AR-00-OUTAG-01, dated July 7, 2000.

On September 23, 1993, the licensee processed a "Document Void Request" to cancel Modification 90-012 which stated, "Current inspection techniques using high-powered cameras preclude the need for inspection ports, additionally, cleaning of the reactor vessel head during the last three outages was completed successfully without requiring access ports." However, the quoted statement was not accurate in all material respects, in that, the licensee left boric acid deposits on the reactor vessel head at the end of both the seventh and eighth refueling outages, the two outages preceding this statement.

Quality Assurance Audit Report AR-00-OUTAG-01, dated July 7, 2000, stated, in part, "Boric Acid Corrosion Control Checklists and CRs were initiated by inspectors when prudent to document and evaluate boric acid accumulation and leaks. Boric acid leakage was adequately classified and corrected when appropriate. Engineering displayed noteworthy persistence in ensuring boric acid accumulation from the reactor head was thoroughly cleaned." However, the audit report was not accurate in all material respects in that the licensee did not: 1) thoroughly clean the reactor head during the outage; 2) did not prepare a boric acid corrosion control checklist for the boric acid left on the head after the cleaning attempt; and 3) (did not) identify, properly classify, or correct the boric acid accumulation and leaks.

Corrective Actions Reviewed

While the focus of this violation was on creating and maintaining complete and accurate records, the violation also involved the implementation and effectiveness of the Boric Acid Corrosion Control (BACC) Program. Review of corrective actions for this problem are discussed in the closure of Violations

05000346/2005013-04 (Failure to Adequately Implement the BACC Program) and 05000346/2005013-07 (Inadequate BACC Procedure), discussed below.

Based on the NRC's review of the corrective actions discussed in the above Inspection Scope section, the licensee's actions were found to be consistent with the relevant docketed correspondence, and this violation is considered closed.

b. (Closed) VIO 05000346/2005013-01: Operating with Reactor Coolant Pressure Boundary Leakage.

Description

This concerned a violation of Davis-Besse TS 3.4.6.2.a, which prohibits plant operation in Modes 1 through 4 with any RCS leakage associated with the RCS pressure boundary. From at least May 18, 2000, to February 16, 2002, FENOC started up and operated the Davis-Besse Station in Modes 1 through 4 while being aware of the presence of significant boric acid deposits on the RPV head, which were indicative of RCS leakage and which could not be justified as being caused by RCS non-pressure boundary leakage alone. The licensee conducted limited cleaning and inspection of the RPV head during the 12RFO in April-May 2000. However, the limited cleaning and inspection of the reactor head were not sufficient to ensure the integrity of the RCS pressure boundary. The startup and operation of the Davis-Besse Nuclear Power Station (DBNPS), with RCS pressure boundary leakage, was a violation of Davis-Besse TS 3.4.6.2.a. This violation was associated with a RED SDP finding. Civil Penalty - $5,000,000 (EA-05-071)

Corrective Actions Reviewed

As discussed at length in FENOC's Integrated Report to Support Restart (IRR) of the DBNPS, issued on November 23, 2003, as supplemented on February 6, 2004, FENOC developed a Return to Service Plan, with seven Building Block Plans, that incorporated comprehensive corrective actions to address the issues identified in both the Technical and Management Root Cause Reports.

The Reactor Head Resolution Plan included replacement of the reactor head and modification of the RPV service structure to facilitate inspections of the RPV head (see IRR, Section IV.B). Pursuant to this Plan, the degraded DBNPS RPV head was replaced with an unused head from the canceled Midland Plant (NRC IR 50-346/02-07). After installation of the new RPV head, the RCS was brought to normal operating pressure and visual inspections were performed for evidence of leakage. The RPV head-to-flange seals and the CRDMs were confirmed to be leak tight (NRC IR 50-346/03-23). The installed replacement head was determined to be in compliance with applicable NRC and industry requirements.

The NRC recognized and accepted the adequacy and effectiveness of FENOC's corrective actions in a letter from James L. Caldwell, Regional Administrator, NRC, to Lew W. Myers, Chief Operating Officer, FENOC, "Approval to Restart the Davis-Besse Nuclear Power Station, Closure of Confirmatory Action Letter, and Issuance of Confirmatory Order" (NRC Restart Approval), March 8, 2004.

In addition to the actions discussed above, FENOC implemented other corrective actions to address the identification of unidentified leakage and associated performance deficiencies, including:

(1) FENOC revised the Boric Acid Corrosion Control (BACC) Program Manual to include the CRDM nozzles as a probable location of leakage, hired a new person to become the plant BACC Program Owner, and implemented a new Job Familiarization Guideline which established specific training requirements and qualifications for boric acid inspectors and the BACC Program owner (NRC IR Nos. 50-346/02-11, 03-09, and 03-17).

(2) FENOC developed and implemented an RCS Integrated Leakage Program to improve the capability for detecting and correcting small leaks that are within the TS limits (NRC IR Nos. 50-346/02-11and 03-09).

(3) FENOC revised the In-Service Inspection (ISI) program to provide for the performance of augmented examinations for selected components, including the CRDM nozzles. Additionally, a formal interface between the ISI Pressure Test and the BACC Program has been established, and training of personnel has been revised to emphasize identification of the leakage source (NRC IR 50-346/03-09).

(4) In an effort to enhance the leak detection capabilities at the DBNPS, a leak detection system, "FLUS," was installed below the RPV to monitor potential leakage of the incore instrumentation nozzles. This system, first of its kind in the United States, operates on the principle of humidity detection.

Based on the above, the licensee's actions were consistent with the relevant docketed correspondence, and this violation is considered closed.

c. <u>(Closed) VIO 05000346/2005013-03</u>: Failure to Determine the Cause of a Significant Condition Adverse to Quality Involving Three Examples of Identified Boric Acid Leakage.

<u>Description</u>

This violation concerned FENOC willfully failing to ensure that a significant condition adverse to quality, associated with the presence of boric acid on the RPV head, at the end of 12RFO, on May 18, 2000, was evaluated and corrected prior to restart of the plant. Specifically, the licensee closed at least three CRs documenting the presence of significant boric acid deposits on the RPV head and associated components without determining the cause of each condition, i.e., the source of the RCS leakage, without taking corrective action to address the immediate condition adverse to quality, i.e., the presence of significant deposits of boric acid on the reactor vessel head, and without taking corrective action to prevent recurrence. This willful violation of 10 CFR 50, Appendix B, Criterion XVI was categorized at Severity Level II (Supplement I) in accordance with the Enforcement Policy. Civil Penalty - $110,000 (EA-05-066).

Corrective Actions Reviewed

In response to Restart Checklist item 3.a (Corrective Action Program) and to ensure that conditions adverse to quality were properly identified, evaluated, and corrected, FENOC improved its CAP. A detailed list of the corrective actions taken to address this violation and prevent recurrence was provided in the Integrated Report to Support Restart of the DBNPS and Request for Restart Approval, Section IV.D, dated November 23, 2003. The NRC recognized and accepted the adequacy and effectiveness of FENOC's corrective actions, as documented in NRC's Restart Approval. The NRC reviewed the CAP and evaluated FENOC's effectiveness in correcting the deficiencies in the program. Specifically, the NRC evaluated the effectiveness of the implementation of various aspects of FENOC's CAP, including: (1) identifying and documenting plant design-related deficiencies; (2) categorizing and prioritizing safety issues for resolution; (3) conducting apparent and root cause analyses; (4) determining EOC; and (5) implementing appropriate and timely corrective actions to ensure adequate resolution of problems.

The NRC's conclusions are documented in IR Nos. 50-346/02-11, 50-346/03-09 and 50-346/03-10. The NRC specifically concluded that the CAP was "sufficiently acceptable" to support plant restart and closed Restart Checklist item 3.a in NRC Special Team IR 50-346/03-10, dated March 5, 2004. Furthermore, the NRC evaluated the implementation of the CAP in IR 50-346/04-17, dated January 30, 2005, and concluded that implementation was adequate for continued operation of the plant.

Based on the above, the licensee’s actions were consistent with the relevant docketed correspondence, and this violation is considered closed.

d. (Closed) VIO 05000346/2005013-04: Failure to Adequately Implement the BACC Program.

Description

This violation documented that at the end of 12RFO, FENOC willfully failed to fully implement the BACC procedure. Specifically, FENOC did not conduct a complete cleaning and inspection of the RPV head as required by the boric acid corrosion control procedure. In addition, FENOC willfully deferred the implementation of a modification which was a corrective action for previous boric acid corrosion control program implementation non-conformances. As a result, FENOC willfully restarted the plant on May 18, 2000, and operated until February 16, 2002, with visible boric acid deposits on the RPV head and uncharacterized RCS pressure boundary leakage. This willful violation of 10 CFR 50, Appendix B, Criterion V was categorized at Severity Level II (Supplement I) in accordance with the Enforcement Policy. Civil Penalty - $110,000 (EA-05-067).

Corrective Actions Reviewed

To prevent and detect any recurrence and in response to Restart Checklist item 3.d (Boric Acid Corrosion Management Program), FENOC performed a detailed, systematic evaluation of the BACC Program and made comprehensive programmatic improvements.

In addition to the installation of the unused reactor head from the canceled Midland Plant, the existing service structure was refurbished, modified with new inspection access openings, and transferred to the RPV head service structure support skirt. A detailed list of the corrective actions taken to address this violation and prevent recurrence is provided in the IRR, at Section IV.D.

The NRC conducted two special inspections associated with identifying and evaluating the effects of boric acid corrosion of components and systems within containment and also performed inspections of the Boric Acid Corrosion Management Program. The inspections are documented in NRC IR Nos. 50/346/02-09, 50-346/02-12, 50-346/02-11, 50-346/03-09 and 50-346/03-17.

Based on the above, the licensee's actions were consistent with the relevant docketed correspondence, and this violation is considered closed.

e. (Closed) VIO 05000346/2005013-06: Failure to Determine the Cause of and Take Corrective Actions to Preclude Repetition for Significant Conditions Adverse to Quality.

Description

This violation concerned the licensee's failure to ensure that the cause of identified conditions adverse to quality were promptly corrected and, for significant conditions adverse to quality, that corrective actions were taken to preclude repetition. The following significant conditions adverse to quality were documented by the licensee in several CRs however, the cause of the conditions was not determined and subsequently actions to preclude repetition were not taken. This violation was associated with a RED SDP finding (EA-03-025).

(1) Fouling of containment air cooling fins by boric acid, between June 2000 and February 16, 2002.

(2) Fouling of the containment radiation elements by boric acid and iron oxide, between April 2001 and February 16, 2002.

(3) An increasing trend in unidentified RCS leakage, between March 2001 and December 2001.

Corrective Actions Reviewed

NRC performed extensive inspections of the licensee's CAP. Specifically, the NRC evaluated the effectiveness of the implementation of various aspects of FENOC's CAP, including: (1) identifying and documenting plant design-related deficiencies; (2) categorizing and prioritizing safety issues for resolution; (3) conducting apparent and the root cause analyses; (4) determining extent of condition; and (5) implementing appropriate and timely corrective actions to ensure adequate resolution of problems.

The NRC's conclusions are documented in NRC IR Nos. 50-346/02-11, 50-346/03-09 and 50-346/03-10. The NRC specifically concluded that the CAP was "sufficiently acceptable" to support plant restart, and closed Restart Checklist item 3.a in NRC Special Team IR 50-346/03-10, dated March 5, 2004. Furthermore, the NRC evaluated the implementation of the CAP in IR Number 50-346/04-17, dated January 30, 2005, and concluded that implementation was safe for continued operation of the plant.

Licensee Event Report 05000346/2002-008-01 & -02 “Containment Air Coolers Collective Significance of Degraded Conditions” was reviewed and closed in NRC IR No. 50-346-05-05.

In addition to the actions described above, FENOC implemented several other corrective steps, including:

> The Containment Air Coolers (CACs) were modified to correct damage from boric acid corrosion. Most of the system was replaced including new CAC motors, plenum, cooling coils, and local service water piping that supplies cooling water to the CACs.
>
> FENOC developed and implemented an RCS Integrated Leakage Program to improve the capability for detecting and correcting small leaks that are within the TS limits. This program was reviewed and found acceptable in NRC IR Nos. 50-346/02-11 and 03-09.
>
> In an effort to enhance the leak detection capabilities at the DBNPS, a leak detection system, “FLUS,” was installed below the RPV to monitor potential leakage of the incore instrumentation nozzles. This system, first of its kind in the United States, operates on the principle of humidity detection.

f. <u>(Closed) VIO 05000346/2005013-07</u>: Inadequate Boric Acid Corrosion Control Procedure.

<u>Description</u>

This violation was associated with a RED SDP finding (EA-03-025) and documented multiple inadequacies in the BACC Program Procedure:

(1) The procedure inappropriately focused on bolted and flanged connections in the procedural definitions for leakage and RCS pressure boundary components. Also, the procedure inappropriately focused on identifying investigation locations rather than identifying the potential for through-wall leakage.

(2) The procedure did not include adequate guidance, specifications, or threshold levels for initiating a “detailed inspection” in order to ensure consistent implementation of the procedure.

(3) The procedure did not require the identification of and corrective actions to preclude the repetition of boric acid leaks, a significant condition adverse to quality, but instead only required the preparation of a repair tag or work order to facilitate repair of the leak.

(4) The procedure did not define the qualifications and training necessary to permit engineering staff to conduct inspections and evaluations in a consistent manner, including the use of proper inspection techniques, observations, recording of results, and evaluations.

(5) The procedure inappropriately exempted stainless steel or Inconel components from further examination related to boric acid corrosion, unless the examination was during an ASME Section XI test which might require a bolting examination.

(6) The procedure inappropriately did not require the licensee staff to maintain records necessary to demonstrate the proper completion of activities affecting quality.

Corrective Actions Reviewed

To prevent and detect any recurrence and in response to Restart Checklist item 3.d (Boric Acid Corrosion Management Program), FENOC performed a detailed, systematic evaluation of the BACC Program, and made comprehensive programmatic improvements.

The NRC conducted two special inspections associated with identifying and evaluating the effects of boric acid corrosion of components and systems within containment. The inspections are documented in NRC IRs 50/346/02-09 and -12. The results of the inspections initially found weaknesses. For example, the inspectors identified that the licensee failed to adequately train personnel for VT-2 certification to perform containment area EOC walk-downs. Also, the licensee lacked visual inspection acceptance requirements, and some components with corrosion and boric acid were identified by the NRC staff were not identified by the licensee.

The licensee implemented corrective actions and subsequent NRC inspection found that the licensee satisfactorily resolved the lack of inspection quality and thoroughness associated with implementation of their extent-of-condition plan for assessing and resolving boric acid issues. The NRC reviewed the licensee's programmatic improvements as documented in NRC IRs 50-346/02-11 and 50-346/03-09, and found that the licensee's programmatic boric acid issues were properly resolved. Restart Checklist item 3.d (Boric Acid Corrosion Management Program) was closed in NRC IR No. 50-346/03-17.

## 4OA6 Meeting, Including Exit

### .1 Exit Meeting Summary

On July 3, 2007, the resident inspectors presented the inspection results to Mr. V. Kaminskas and other members of the licensee’s staff, who acknowledged the findings.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Reviewing the 2006 SC/SCWE Final Report with Mr. V. Kaminskas, Director Site Operations and others of your staff on May 3, 2007 and Mr. C. Price, Director, Performance Improvement on July 17, 2007.

- Radiation Monitoring Instrumentation and Protective Equipment and Barrier Integrity Performance Indicator with Mr. V. Kaminskas, Director Site Operations on June 22, 2007

- Maintenance Effectiveness Periodic Evaluation with Mr. B. Boles, Director Maintenance on June 29, 2007.

- Follow-up Inspection Activities Associated with the Reactor Pressure Vessel Head Degradation Issue with Mr. C. Price, Director, Performance Improvement on July 16, 2007

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

M. Bezilla, Site Vice President
B. Boles, Director, Maintenance
T. Brooks, Radiation Protection
K. Byrd, Manager, Design Engineering
V. Capizziello, Chemistry Supervisor
G. Chung, Nuclear Engineer
A. Dawson, Supervisor. Nuclear Chemistry
J. Grabnar, Director, Engineering
J. Hasselbach, Radiation Protection
C. Hawley, Manager Site Projects
R. Hovland, Manager, Site Training
R. Hruby, Manager, Regulatory Compliance
R. Jarosi, Employee Concerns Program
V. Kaminskas, Director, Plant Operation
G. Kendrick, Work Management Manager
J. Marley, Problem Solving Team Lead
G. Melssen, Maintenance Rule Coordinator
D. Moul, Manager, Site Operations
W. Mugge, Outage Management Manager
A. Parcival, Adv. Nuclear Specialist (Chemistry)
M Parker, Plant Engineering Supervisor
S. Plymale, Manager, Plant Engineering
C. Price, Director, Performance Improvement
J. Rinckel, Vice-President, Fleet Oversight
J. Scott, Lead Radiation Protection
J. Vetter, Manager, Emergency Response
G. Wilson, Plant Engineering/Reactor Engineering Supervisor
D. Wuokko, Regulatory Compliance Supervisor

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open and Closed

| | | |
|---|---|---|
| 05000346/2007003-01 | NCV | Improper Implementation of Independent Verification Requirements in Performance of Instrument and Control Surveillance Test Procedures for TS Required Mitigation Systems (Section 1R22) |
| 05000346/2007003-02 | NCV | Licensed Reactor Thermal Power Exceeded During Normal Plant Operations (Section 4OA3) |

| | | |
|---|---|---|
| 05000346/2007003-03 | NCV | Out of Service Seismic Force Monitoring Equipment Affecting Emergency Plan Response (Section 4OA5) |

Closed

| | | |
|---|---|---|
| 05000346/2006-004-00 | LER | Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures Davis-Besse Nuclear Power Station, Unit No.1 (Section 4OA3) |
| 05000346/2006-004-01 | LER | Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures Davis-Besse Nuclear Power Station, Unit No (Section 4OA) |
| 05000346/2005013-01 | VIO | Operating with Reactor Coolant Pressure Boundary Leakage (Section 4OA6) |
| 05000346/2005013-02 | VIO | Information included in Condition Report (CR) 2000-1037 and Work Order 00-001846-000 was not Complete and Accurate in all Material Respects (Section 4OA6) |
| 05000346/2005013-03 | VIO | Failure to Determine the Cause of a Significant Condition Adverse to Quality Involving Three Examples of Identified Boric Acid Leakage (Section 4OA6) |
| 05000346/2005013-04 | VIO | Failure to Adequately Implement the BACC Program (Section 4OA6) |
| 05000346/2005013-05 | VIO | Failure to Provide Complete and Accurate Information to NRC in Response to Bulletin 2001-01 (Section 4OA6) |
| 05000346/2005013-06 | VIO | Failure to Determine the Cause of and Take Corrective Actions to Preclude Repetition for Significant Conditions Adverse to Quality (Section 4OA6) |
| 05000346/2005013-07 | VIO | Inadequate Boric Acid Corrosion Control Procedure (Section 4OA6) |
| 05000346/2005013-08 | VIO | Failure to Maintain Complete and Accurate Records (Section 4OA6) |
| 05000346/2005013-09 | VIO | Failure to Maintain Complete and Accurate Records (Section 4OA6) |

## LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected 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 stated in the body of the inspection report.

<u>1R01</u> <u>Adverse Weather Protection</u>
DB-OP-01300; Switchyard Management; Revision 03
DB-OP-06913; Seasonal Plant Preparation Checklist; Revision 16
NOP-OP-1003; Grid Reliability Protocol; Revision 00
WO200232619; PM 7290 Intake Structure/Penthouse Summer Prep
WO200065564; PM 6382 F108-1 “RPLC” Winter Oil & Fltr
WO200065565; PM 6383 F108-2 “RPLC” Winter Oil & Fltr

<u>1R04</u> <u>Equipment Alignment</u>
CR 05-05614; #2 HPI Pump DC Lube oil Pump is Not Operating
CR 06-00471; Question Raised Regarding High press Injection Min Recirc During Mid Size LOCA
CR 06-01304; HP2B stroke Time Increase
CR 06-01485; HPI Pump 1 D/P Above CC 14.92B Max. 103 percent of Baseline Curve
CR 06-01616; PCR DB-SP-03802 HPI 1 Baseline Test
CR 06-01618; Requested Design Engineering Eval Of 4-5-06 DB-PF-03802 HPI Baseline Test Data
CR 06-01619; HISHP2D Indicates Closed with 130 GPM Flow Indicated On FYIHP3D
CR 06-01620; HPI 1-1 Baseline Test, Motor Data DB-PF-05064 Greater Than 100 percent Load Amps
CR 06-01635; Piping Downstream Of HP60 Subjected to #1 HPI Pump Discharge Pressure
CR 06-02826; ISTB3 SFAS Stroke Time For HP2C and HP2D
CR 06-6828; Potential Motor Degradation Of HPI #2 (MP58-2)
CR 06-9157; FIS HP4B out of Tolerance - Evaluation Required
CR 06-10541; Small Oil #1 HPI Pump Oil Reservoir
CR 07-13063; HP12 Does Not Operate Smoothly
CR 07-15079; Lube Oil reservoir Is Leaking Oil from Cover
CR 07-16092; P58-2 HPI Pump 2 Vibration For C-H Velocity Apparently Mis-recorded
CR 07-16890; Oil Leak On #2 HPI Pump Inboard Bearing
CR 07-18665; Minor oil Leak On #1 HPI Cooler
CR 07-18705; HPI train 1 Pump test DB-SP-03218 Completed As A Partial Test
CR 07-18752; Valve Bracket Missing Bolt
CR 07-19240; Documentation of the NRC Resident Inspector Observation/Question (NRC Identified)
CR 07-20128; BACC - Boric Acid Packing Leak On HP35 (NRC Identified)
DB-OP-06011;High Pressure Injection System; Revision 18 and Revision 19
DB-OP-0605; Control Room Emergency Ventilation System; Revision 9
Drawing M-033A; P&ID High Pressure Injection; Revision 36
Drawing OS-003; High Pressure Injection System; Revision 28 and Revision 29
Drawing )S-032B; Control Room Emergency Ventilation System; Revision 16

Drawing ISIM2-233D, Sheet 3; H.P. Injection System; Revision 7
ECP 07-0062-00; Installation of a Manual Ball Valve (DB-HP41); Revision 00

1R05 Fire Protection
CR 07-20980; NRC Resident Inspector Expresses Concern Over Extinguisher Hydro Periodicity (NRC Identified)
CR 07-22345; Incorrect Number Of Fire. Detectors Shown On A-223F (NRC Identified)
Davis-Besse Nuclear Power Station Fire Hazard Analysis Report
DB-FP-00007; Control Of Transient Combustibles; Revision 07
DB-FP-00009; Fire Protection Impairment and Fire Watch; Revision 09
Drawing A-223F; Fire Protection General Floor Plan EL 585'; Revision 18
Drawing A-224F; Fire Protection General Floor Plan EL 603'; Revision 21
Drawing A-221F; Fire Protection General Floor Plan 545'; Revision 08
Drawing A-230F; Fire Protection Intake Structure; Revision 09
Drawing OS-047A; Operational Schematic Fire Protection System; Revision 09

1R06 Flood Protection
DB-MM-09061; Service Water Pump Maintenance; Revision 5
RA-EP-02830; Flooding; Revision 1
RA-EP-02880; Internal Flooding; Revision 3
Davis-Besse Probabilistic Safety Assessment Summary Report; October 1999

1R11 Licensed Operator Requalification Program
DBBP-TRAN-0017; Conduct of Simulator Training; Revision 2
DBBP-TRAN-0502; Development and Conduct of Continuing Training Simulator Evaluations; Revision 3
ORQ-EPE-S231; SWP Trip, Mn Gen volt Reg malfunction, Mn Gen H2 Leak, Reactor Trip and Overcooling; Revision 1

1R12 Maintenance Effectiveness
CR-06-6268; Excessive Nitrogen Additions to No.1 Core Flood Tank; dated September 13, 2006
CR-06-6320; CF1544 Leaking by Closed Seat; dated September 13, 2006
CR-07-15718; Issues with HP31 Packing Leakage; dated March 7, 2007
CR-07-18074; HPI Train 1 Discharge Piping – Potential for Air Intrusion; dated April 10, 2007
CR-07-18182; Request for Prompt Operability Determination for HPI Train 1; dated April 12, 2007
CR-07-18732; HP83A Vent Pressures – Operations Evolution Order; dated April 18, 2007
CR-07-18751; HPI Gas Intrusion Issue; dated April 19, 2007
CR-07-18777; Document Results of HPI Venting on April 19, 2007; dated April 19, 2007
CR-07-18769; Narrative Log Expectations and Standards Not Met; dated April 19, 2007
CR 06-02709; ICS does not Maintain Desired Power Level; dated June 30, 2006
CR 06-06003; Root Cause Analysis Report Manual Trip due to Loss of Condenser Vacuum from Broken Turbine Waste Water and Oil Drain; dated September 6, 2006
CR 06-01878; Failure of Breaker AACD1 to Close to Maintain Bus D1 Energized during Testing; dated April 16, 2006
CR 06-02816; Unexpected Trip of SAC-1 - Entered Loss of Station Air Procedure; dated July 16, 2006

Containment Isolation Valves CF-1541 and CF-1544 Local Leak Rate Trends; dated September 12, 1991 - March 20, 2006
Core Flood Tank Computer Trends (Level and Pressure); dated September 1, 2006 - April 18, 2007
Unit Logs; dated April 17 - 20, 2007
Maintenance Rule Periodic Assessment; March 2004 - April 2006; dated June, 2006
Medium Voltage AC; (a)(1) Action Plan; dated August 2006
Condensate/Condenser; (a)(1) Action Plan; dated November 2006
Safety Features Actuation System; (a)(1) Action Plan; dated October 2003
Station and Instrument Air; (a)(1) Action Plan; dated September 2003
List of Systems Within the Scope of the Maintenance Rule; dated June 2007
List of Functional Failures for Assessment Period from March 2004 to April 2006
Medium Voltage AC System Health Report; First Quarter 2007
Station and Instrument Air System Health Report; First Quarter 2007
Safety Features Actuation System Health Report; First Quarter 2007
Integrated Control System Health Report; First Quarter 2007
Expert Panel Meeting Minutes; dated November 11, 2004
Expert Panel Meeting Minutes; dated November 9, 2006
Expert Panel Meeting Minutes; dated January 11, 2007
DB-PF-00003; Maintenance Rule; Revision 7
NOP-ER-3004; FENOC Maintenance Rule Program; dated April 2, 2007

1R13 Maintenance Risk Assessment and Emergent Work Evaluation
Maintenance Risk Summaries for the Week of April 1, 2007; Revision 0
Maintenance Risk Summaries for the Week of April 8, 2007; Revisions 0 through 2
Maintenance Risk Summaries for the Week of April 15, 2007; Revisions 0 through 3
Maintenance Risk Summaries for the Week of May 13, 2007; Revision 0
Maintenance Risk Summaries for the Week of May 20, 2007; Revisions 0 through 1
Level 1 Schedule for May 18, 2007 Downpower; May 4, 2007
NG-DB-00001; On-Line Risk Management; Revision 6
Station ALARA Committee Meeting Package; May 4, 2007
Work Implementation Schedule, Subsystem Sort; April 1, 2007
Work Implementation Schedule, Subsystem Sort; April 8, 2007
Work Implementation Schedule, Subsystem Sort; April 15, 2007
Work Implementation Schedule, Subsystem Sort; May 20, 2007

1R15 Operability Evaluations
Calculation C-EE-004.01-049; 4.16 kV C1/D1 Degraded Voltage Relay (DVR) setpoint; Revision 14
Calculation C-EE-015.03-008; AC Power System Analysis: Revision 4
Calculation C-NSA-049.02-024; DHP O/B Bearing Reservoir Allowable Oil leakage; Revision 1
CR 07-18074; HPI Train 1 Discharge Piping Potential Air Intrusion
CR 07-18280; Request For A Prompt Operability Determination For HPI Train 1
CR 07-18732; HP83A Vent Rig Pressures - Operations Evolution Order
CR 07-18751; HPI Gas Intrusion Issue
CR 07-18769; Narrative Log Expectations And Standards Not Met
CR 07-18777; This CR Was written To Document results Of HPI Venting On 4/19/07
CR 07-20803; CDBI self Assessment Bus D1: Bus D1 Voltage Band in DB-SC-03041
CR 07-21914; Decay Heat Pump 2 Oil Leak

DB-SC-03041; On-Site AC Bus Sources Lined Up, Available and Isolated (Modes 1, 2, 3, and 4); Revision 9
Drawing CCB-19-1; Core Flooding Supply From HP Injection Pumps to Core flood Tanks; Revision 3
Drawing CCB-19-2; From HP Injection Pumps to Core Flooding Tank T9-2; Revision 3
Drawing CCB-19-3; From HP Injection Pumps to Core Flooding Tank T9-1; Revision 4
Drawing M-233D; Piping Isometric HP Injection System Auxiliary Building; Revision 26
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1R19 Post-Maintenance Testing
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DB-PF-03811; Miscellaneous Valves Test; Revision 13
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DB-SC-03122; SFAS Component Tests; Revision 02
DB-SS-03041; Control Room Emergency Ventilation System Train 1 Monthly Test; Revision 5
WO200190022; SFAS Control Relay
WO200186977; PM 3906 SV236 ‘RPLC’ CTMT N2 ISOL
WO200209017; PM 2169 Inspect Control Room Emergency Ventilation System Train 1
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WO200264845; CTMT Air Cooler Logic SFAS Channel 4 Output Module

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CR 06-7870; Step performance Prior to Independent Verification of a Previous Step May Violate NOP-LP-2601
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CR 07-20843; Cond Pump 1 Exceed 205 Degrees On bearing Temperature
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CR 07-21904; Condition Of TPCW Return Line
CR 07-22137; Ongoing Commitment from LER Not Captured In Commitment Database (NRC Identified)
LER 2006-004-00; Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures Davis-Besse Nuclear Power Station, Unit No.1
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M-233D; Piping Isometric – High Pressure Injection System – Auxiliary Building; Revision 26
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## LIST OF ACRONYMS USED

| | |
|---|---|
| AC | Alternating Current |
| ADAMS | Agency-wide Document Access and Management System |
| AFI | Area For Improvement |
| ARM | Area Radiation Monitors |
| CAP | Corrective Action Program |
| CCW | Component Cooling Water |
| CEDE | Committed Effective Dose Equivalent |
| CFR | Code of Federal Regulations |
| CR | Condition Report |
| CRDM | control rod drive mechanism |
| CST | Condensate Storage Tank |
| DEI | Dose Equivalent Iodine |
| DH | Decay heat |
| DRP | Division of Reactor Projects |
| DRS | Division of Reactor Safety |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| EOC | Extent of Condition |
| FENOC | FirstEnergy Nuclear Operating Company |
| HPI | High Pressure Injection |
| IE | Initiating Events |
| IMC | Inspection Manual Chapter |
| IP | Inspection Procedure |
| IR | Inspection Report |
| IST | Inservice Testing |
| LER | Licensee Event Report |
| MS | Mitigating Systems |
| MSPI | Mitigating Systems Performance Index |
| MWe | Megawatts Electric |
| NCV | Non-Cited Violation |
| NRC | Nuclear Regulatory Commission |
| NRR | Nuclear Reactor Regulation |
| ODCM | Offsite Dose Calculation Manual |
| OE | Operating Experience |
| PI | Performance Indicator |
| ppm | parts per million |
| RCA | Radiologically Controlled Area |
| RETS | Radiological Environmental Technical Specifications |
| RIS | Regulatory Information Summary |
| SC/SCWE | Safety Culture/Safety Conscious Work Environment |
| SCBA | Self-Contained Breathing Apparatus |
| SDP | Significance Determination Process |
| SFAS | safety function actuation system |
| SSC | structures, systems and components |
| TS | Technical Specification |
| USAR | Updated Safety Analysis Report |

| | |
|---|---|
| VAC | Volts Alternating Current |
| VHRA | very high radiation area |
| WPG | Work Process Guideline |