May 3, 2007

EA-03-0214

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 05000346/2007002

Dear Mr. Bezilla:

On March 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Davis-Besse Nuclear Power Station. The enclosed inspection report documents the inspection findings which were discussed on March 29, 2007, with Mr. V. 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 03-0214), and activities associated with your Independent Spent Fuel Storage Installation program under Docket Number 07200014.

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, the NRC identified two findings of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report.

If you contest the subject or severity of any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at Davis-Besse.

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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eric R. Duncan, Chief
Branch 6
Division of Reactor Projects

Docket Nos. 50-346 and 72-014
License No. NPF-3

Enclosure: Inspection Report 05000346/2007002
w/Attachment: Supplemental Information

cc w/encl: The Honorable Dennis Kucinich
J. Hagan, President and Chief
Nuclear Officer - FENOC
J. Lash, Senior Vice President of
Operations and Chief Operating Officer
Richard Anderson, Vice President, Nuclear Support
Manager - Site Regulatory Compliance
D. Pace, Senior Vice President of
of Fleet Engineering
J. Rinckel, Vice President, Fleet Oversight
D. Jenkins, Attorney, FirstEnergy
Director, Fleet Regulatory Affairs
Manager - Fleet Licensing
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

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Letter to Mark B. Bezilla from Eric R. Dundan Last dated May 3, 2007

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2007002

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346 and 72-014
License No:	NPF-3
Report No:	05000346/2007002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	January 1, 2007, through March 31, 2007
Inspectors:	J. Rutkowski, Senior Resident Inspector R. Smith, Resident Inspector C. Acosta Acevedo, Reactor Inspector S. Bakhsh, Materials Inspector T. Bilik, Reactor Inspector J. Cassidy, Health Physicist A. Dahbur, Reactor Inspector M. Gryglak, Materials Inspector T. Go, Health Physicist N. Valos, Senior Operations Engineer M. Wilk, Resident Inspector (Perry) G. Wright, Project Engineer
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000346/2007002; 1/1/2007 - 3/31/2007; Davis-Besse Nuclear Power Station; Other Activities

This report covers a 3-month period of baseline inspection. The inspection was conducted by Region III inspectors, resident inspectors and regional health physics inspectors. Two 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 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 55.46(d)(1), "Continued assurance of simulator fidelity," when the facility licensee failed to conduct a simulator "Generator Trip" malfunction performance test in a manner sufficient to ensure that simulator fidelity had been demonstrated and met. The "Generator Trip" malfunction performance test is one of 25 tests (Item Number 16) required by Section 3.1.4 in ANSI/ANS-3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training." The facility licensee is committed to adhering to the requirements of this standard. Specifically, the licensee failed to adequately conduct the required "Generator Trip" malfunction performance testing to ensure that simulator fidelity was demonstrated and met to allow conduct of the generator trip evolution. The licensee's corrective actions included revising the simulator "Generator Trip" test procedure, and then performing the revised procedure to adequately test the generator trip malfunction.

  This finding was considered more than minor because of the realistic potential of providing negative training based on significant simulator deficiencies compared to the actual plant. This resulted from inadequate testing of the simulator to assure that the simulator appropriately replicated the actual plant and would not negatively affect operator actions on the actual plant. The finding was of very low safety significance because the discrepancy was on the simulator and the real plant functioned properly. Furthermore, no actual plant emergency occurred and there was no actual impact on equipment or personnel safety. (Section 4OA5)

Cornerstone: Other

Green**.** The inspectors identified a Severity Level IV Non-Cited Violation of the Certificate of Compliance, No. 1004, Condition 1.1.3, Quality Assurance and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for failure to correct a condition adverse to quality. Specifically, the licensee failed to remove transient combustible material within 50 feet from the Horizontal Storage Modules (HSMs) to restore compliance with the NRC issued 10 CFR Part 72 license and its fire protection procedure. After the issue was identified, the licensee took immediate corrective actions to remove all transient combustible material inside the 75-foot zone around the HSMs and generated a condition report to enter this issue into the corrective action program.

This finding was more than minor because the lack of adequate corrective actions resulted in a more significant safety concern since the prolonged presence of combustible materials within 50 feet of HSMs for approximately 10 months increased the vulnerability of the HSMs to a fire. In addition, the lack of adequate corrective actions had the potential to become a programmatic issue and could have adversely affected NRC regulatory oversight and enforcement processes, as the agency relied on the licensee's adequacy of corrective actions to correct an NRC identified violation. The inspectors determined that the finding was not suitable for SDP evaluation because the noncompliance involved 10 CFR Part 72 dry fuel storage activities. Therefore, this finding was reviewed by Regional Management and dispositioned using traditional enforcement. The finding was determined to be of very low safety significance. The combustible material was contained within metal containers which could have mitigated the spread of a potential fire. Also, the plant fire brigade could have been dispatched to extinguish a fire involving the transient combustible material before the HSMs incurred significant damage. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution because licensee personnel failed to thoroughly evaluate the problem (P.1(c)). (Section 4OA5)

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

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

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, the plant was operating at 100 percent power. On March 3, 2007, the licensee lowered power to about 92 percent to perform main turbine valve testing and returned to 100 percent power on March 4, 2007. On March 23, 2007, the licensee lowered power to about 50 percent to perform condenser tube plugging activities. Following the condenser tube plugging activities, the plant was returned to 100 percent power on March 26, 2007. The plant operated at approximately 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed applicable licensee procedures, attended meetings, and performed a walkdown of staged equipment that addressed preparations for a severe winter storm. The inspectors evaluated the licensee's preparations for adverse weather with emphasis on conditions that could result from high winds and heavy snow fall. The inspectors focused on plant management's actions for implementing the station isolation procedure, specifically ensuring adequate personnel for safe plant operation and emergency response. The inspectors also determined if any adverse effects had occurred during the storm and if the licensee took the appropriate actions to address the conditions.

This review represented one inspection sample associated with the licensee's preparation for adverse weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

.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 equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program. Documents reviewed are listed in the Attachment to this report. The inspectors reviewed:

- auxiliary feedwater train 1 during a train 2 outage;
- emergency diesel generator (EDG) number 2 system alignment during EDG number 1 scheduled maintenance outage;
- makeup train 2 system alignment after maintenance outage on makeup pump 2;
- decay heat (DH) train 2 during DH train 1 scheduled maintenance outage; and
- EDG number 1 system alignment during EDG number 2 scheduled maintenance outage.

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (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 if 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. The following areas were toured:

- auxiliary feedwater pump 1 room (Fire Area E, Rooms 237);
- high voltage switchgear room 'A' (Fire Area S, Room 325);
- EDG number 2 rooms (Fire Area J and Rooms 319, 319A and 320A);
- electrical penetration room number 2 (Fire Area DF and Room 427);
- EDG number 1 room (Fire Area K and Room 318);
- radwaste and fuel handling and air supply equipment area (Fire Area EE and Room 500);
- radwaste exhaust equipment and main station exhaust fan room (Fire Area EE and Room 501);
- emergency core cooling system (ECCS) pump room 1 (Fire Area AB and Room 105); and
- low voltage switchgear room for F bus (Fire Area X and Room 428).

This review represented nine quarterly inspection samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors evaluated the auxiliary feedwater pump rooms for internal flooding hazards. The inspectors toured the rooms and visually reviewed the large drain openings and drain path check valves. As part of this inspection, the inspectors reviewed the assumptions the licensee made in flooding impact mitigation and sampled the regulatory documents developed by the licensee that addressed potential flooding in the rooms. Additionally, the inspectors reviewed the procedures used by the licensee to address flooding in the auxiliary feedwater pump rooms and the procedures used to ensure that plant conditions were consistent with room flooding assumptions. This included reviewing the procedure for isolation of potential high energy lines associated with the motor driven startup feed pump and the work order for periodically testing the functionality of the drain path check valves.

This review represented one sample of internal flooding.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the licensee program for maintaining ultimate heat sink availability in the event of conditions that caused loss of connectivity between Lake Erie and the plant's service water intake canal and forebay. Specifically, the inspectors reviewed the licensee's response to frazil ice conditions at the plant's in-lake submerged intake crib that prevented normal makeup to the intake canal and forebay for the service water pumps during the period of February 2 to February 4, 2007. Additionally, the inspectors reviewed assumptions associated with ultimate heat sink capability with reduced intake canal water volumes and with assumptions on the ability to maintain intake canal water volume with the licensee's contingency actions for frazil ice.

This review represented one annual sample of heat sink performance.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

On January 19, 2007, the inspectors observed an operating crew during an individual evaluation for crew familiarity in the simulator and attended the post-session licensee critique. 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. The operational scenario included a failed nuclear instrument, a loss of secondary service water, and a steam generator tube rupture with a failed open turbine bypass valve.

This review represented one quarterly inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and resolution of performance issues associated with the following system and program:

- Motor Operator Valve (MOV) Program, and
- Auxiliary Feedwater System.

The reviews consisted of evaluating the following items:

- use of the condition report (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;
- appropriateness of goals and corrective actions associated with long-term reliability;

- if the physical condition of the system appeared consistent with the status as reflected in CRs and open work orders;
- if the licensee's corrective actions included extent of condition; and
- appropriateness of maintenance rule system status classification with emphasis on if current reclassification appeared appropriate for the equipment's recent history.

Additionally, the inspectors performed a walkdown of the systems and selectively discussed planned corrective actions with the system or program engineer. For the MOV program, the inspectors reviewed the timeliness of target thrust revised calculations as identified in CR 03-08893, corrective action 2.

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

- initial risk summaries for the week of January 21, 2007, during the emergency EDG number 1 outage and revised schedules due to the emergent issue of MU32 (pressurizer level control valve) failure to operate properly;
- initial risk summaries for the week of February 11, 2007, during the DH train 1 outage;
- initial risk summaries for the week of February 25, 2007, during the EDG number 2 outage and revised schedules during the week; and
- initial risk summaries for the week of March 18, 2007, during the down power for repairs of condenser tube leaks and the impact of an emergent issue with the EDG number 1 fuel transfer pump.

This review represented four 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 reviewed the technical adequacy of the evaluations to ensure that Technical Specification (TS) operability was properly evaluated and that the subject component or system remained available such that no unrecognized increase in risk occurred. In addition, the inspectors reviewed compensatory measures implemented to verify the effectiveness of those measures and that they 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 following issues were evaluated:

- CR 06-11269, CR 06-11421, CR 06-11483, and CR 06-11740, which addressed room damper adequacy for the EDGs, low voltage switchgear rooms, and component cooling water (CCW) pump room for the design basis tornado;
- CR 07-12189, which addressed the operability of control room emergency ventilation system (CREVS) train 2;
- CR 07-13103, which addressed the operability of DH pump number 2;
- CR 07-14758, which addressed the operability of service water train 1 with a test failure of valve SW276 which is a vacuum breaker in one of two vent paths on the train 1 service water return lines from the containment air coolers; and
- CR 07-15233, which addressed the Post-LOCA [Loss-of-Coolant-Accident] containment water level and long-term cooling while using the containment emergency sump strainer.

This review represented five inspection samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed Engineering Change Package 99-00047-00, “Feedwater Flow Rate Caldon (Leading Edge Flow Meter) Check Plus System,” as a sample of a permanent plant modification. The inspectors reviewed the modification after installation and reviewed testing documents to determine if the design basis, licensing basis, and performance capability of the plant heat balance would be accurate for the change in the plant’s licensed power to 2817 Megawatts Thermal (MWt) from 2772 MWt. This power up-rate license amendment request will be submitted to the NRC in the near future. This power up-rate, if approved by the NRC, would allow the licensee to increase licensed thermal power by 1.63 percent due to measurement uncertainty recaptured by this modification. Additionally, the inspectors reviewed procedures specified in the design package to determine if the procedures had been revised to address new testing requirements necessitated by the design change.

This review represented one inspection sample.

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:

- auxiliary feed train and pump 2 surveillance test after auxiliary feedwater train maintenance on January 10 and 11, 2007;
- steam-feed rupture control system response time testing after replacement of micro-switches in pressure differential switches PDS2685C/D on January 16, 2007;
- safety features actuation system (SFAS) channel 1 functional test for CC1328 proper indication on channel 3 after the replacement of a internal signal cable on January 18, 2007;
- post maintenance testing of make up system pressurizer level control valve MU32 after the repair of the valve actuator on January 26 and 27, 2007;
- EDG-1 184-day test after the performance of the 6-year preventive maintenance outage on January 26, 2007;
- control room emergency ventilation system train 2 monthly test after replacement of oil pressure switch PSL28016 on January 30, 2007;
- EDG-2 post maintenance testing on March 3, 2007, after relay replacement and diesel preventive maintenance; and
- reactor trip breaker B testing on March 21, 2007, after removal, testing, and reinstallation of the undervoltage sensing relay associated with the breaker's shunt trip.

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 eight 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, Updated Safety Analysis Report (USAR), and licensee procedural requirements, and also demonstrated that the equipment was capable of performing its intended safety functions. The inspectors used the documents listed in the Attachment to determine if the test met the TS frequency requirements; that the test was conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria was met; and that the results of the test were properly reviewed and documented. The following surveillances were evaluated:

- DB-SP-03137, DH train 2 pump and valve test on January 5, 2007;
- DB-MI-03059, reactor protection system (RPS) channel 3 functional test/calibration on January 25, 2007;
- DB-SC-03111, SFAS channel 2 functional test on January 29, 2007;
- DB-SP-03161, auxiliary feedwater train 2 testing of level controls, interlocks and flow measurement on January 30, 2007;
- DB-MI-03354, anticipatory reactor trip system (ARTS) channel 4 functional test on February 28, 2007;
- DB-SP-03357, reactor coolant system (RCS) water inventory balance on February 24 and 25, 2007;
- DB-ME-03001, station batteries quarterly test train 2 on March 6, 2007;
- DB-PF-03271, containment integrity verification including verification of the status of the containment purge isolation valves on March 20, 2007; and
- DB-OP-01101, quarterly containment entry and inspection on March 24, 2007.

This review represented nine inspection samples of which one was a quarterly in-service testing (IST) inspection sample, one was a containment isolation valve sample, and one was an RCS leakage measurement sample.

b Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed long-term scaffold installations and hose connections for determination if they should be classified as temporary modifications. The review included a walkdown and physical inspection of a large sample of the long-term scaffold installations and long-term hose connections for potential undesirable interactions with plant equipment and systems. The inspectors also reviewed the scaffold installations, hose connections, and licensee's review activities for compliance with the licensee's temporary modification control procedures. Additionally, the inspectors questioned licensee personnel on the rationale for long-term scaffold installations and hose connections not being classified as temporary modifications. The inspector also questioned the licensee on actions for addressing missed long-term scaffold annual reviews specified in the licensee's procedures. The inspectors reviewed the condition reports written to document the missed reviews.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors monitored the licensee's emergency preparedness drill conducted on March 22, 2007. The observations included licensee preparations, 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 simulator control room and technical support center. The inspectors reviewed the effectiveness of the licensee's communications, the accuracy of situation evaluations, and the timeliness of simulated required reporting 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

Cornerstones: **Public Radiation Safety**

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

.1 Onsite Inspection

a. Inspection Scope

The inspectors walked down the major components of the gaseous and liquid release systems (e.g., radiation and flow monitors, demineralizers and filters, tanks, and vessels) to observe current system configuration with respect to the description in the Final Safety Analysis Report (FSAR), ongoing activities, and equipment material condition.

The inspectors observed the routine processing (including sample collection and analysis) and release of radioactive gaseous effluent to assess whether appropriate treatment equipment was used and whether the radioactive liquid/gaseous effluent was processed and released in accordance with Radioactive Effluent TS/Offsite Dose Calculation Manual (RETS/ODCM) requirements. Specifically, the inspectors observed station vent releases, weekly radiological monitoring, and sampling analysis.

The inspectors reviewed records of the most recent instrument calibrations (channel calibrations) for each point-of-discharge liquid/gaseous effluent radiation monitor and for the plant service water effluent monitor to determine if these monitors had been calibrated consistent with industry standards and in accordance with station procedures, TSs and the ODCM.

The inspectors reviewed effluent radiation monitor setpoint bases and alarm setpoint values for the point of discharge liquid/gaseous effluent radiation monitors to assess their technical adequacy and for compliance with ODCM criteria. Additionally, the inspectors selectively reviewed gaseous effluent monitor operational trend data and discussed with system engineering staff the historical performance of the process/effluent radiation monitoring system to assess the overall condition of the system.

The inspectors reviewed chemistry department quality control data for those instrumentation systems used to quantify effluent releases for indications of potential degraded instrument performance. Specifically, the inspectors reviewed the most recent efficiency calibration records and lower limit of detection (LLD) determinations and selected other quality control data for Chemistry Department gamma spectroscopy systems and for the liquid scintillation counter.

The inspectors reviewed the licensee’s technical justification for changes made by the licensee to the ODCM as well as to the liquid or gaseous radioactive waste system design, procedures, or operation since the last inspection. The inspectors reviewed this information to determine whether the changes affect the licensee’s ability to maintain effluents As-Low-As-Reasonably-Achievable (ALARA) and whether changes made to monitoring instrumentation resulted in a non-representative monitoring of effluents.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.2 Effluent Release Data, Dose Calculations, and Laboratory Analytical Instrumentation Quality Control

a. Inspection Scope

The inspectors reviewed the licensee's radioactive liquid waste release permits, including the projected doses to members of the public and the controls (physical and administrative) used by the licensee to ensure that radioactive liquid effluents would not be inadvertently released from the station.

The inspectors reviewed records of abnormal releases or releases made with inoperable effluent radiation monitors. The review was to ensure that adequate defense-in-depth was maintained against an unmonitored, unanticipated release of radioactive material to the environment. Specifically, the inspectors reviewed a 400-gallon unmonitored release of condensate liquid from the turbine auxiliary boiler to an onsite Training Center Pond on September 11, 2005, through the storm sewer discharge line instead of through the condenser pit system in the turbine building. The inspectors reviewed the licensee's evaluation of the release, which included licensee sampling results, to ensure that the licensee had met regulatory requirements contained in 10 CFR Part 20 and the licensee's procedures. The licensee discharged the condensate containing a total of 6.4E-05 Curies of tritium. The release had a calculated whole body dose of 1.7E-07 millirem, which was significantly lower than the annual ODCM limit of 3 millirem. The inspectors also reviewed the licensee's corrective actions, including human performance investigation. The abnormal release was reported in the licensee's 2005 Annual Effluent Report. The inspectors also reviewed the licensee's implemented engineering change request to the turbine building sump drains to prevent future potential for unmonitored releases.

The inspectors selectively reviewed radioactive liquid/gaseous effluent release permits and associated gaseous effluent sampling data for selected periods in 2005 through 2006, including results of chemistry sample analyses, the vendor laboratory analysis results for difficult to detect nuclides, and the licensee's effluent release procedures and practices. Also, the inspectors reviewed the methods for calculating the projected doses to members of the public from gaseous effluent releases. These reviews were performed to determine whether the licensee adequately applied the analysis results in its dose calculations consistent with ODCM methodology and to determine if appropriate treatment equipment was used and if gaseous effluents were released in accordance with the RETS/ODCM to meet procedural requirements.

The inspectors accompanied a chemistry technician to observe the routine weekly collection of particulate, iodine, noble gas and tritium samples from the station vent effluent monitor sampling system. The inspectors accompanied the technician to determine if sampling practices, sampler restoration and analytical techniques were sound and consistent with procedure and to determine if the sampling system's configuration provided representative sampling. The inspectors reviewed the licensee's practices for compensatory sampling during periods of effluent monitor inoperability to assess compliance with ODCM requirements.

The inspectors selectively reviewed both monthly and quarterly dose calculations and projections to ensure that the licensee properly calculated the offsite dose from gaseous and liquid effluent releases and to determine if any RETS/ODCM (i.e., Appendix I to 10 CFR Part 50) design objectives (limits) were exceeded. The inspectors reviewed the plant's source term data to determine if all applicable radionuclides that were discharged were included in the dose calculations.

The inspectors reviewed the licensee's 10 CFR 50.75(g) file, which documented the licensee's evaluation of a residual soil contamination report dated September 18, 1992. The release occurred from a leaking check valve in the hydrogen addition system onto the ground, contaminating the ground in the vicinity of the borated water storage tank area. The inspectors reviewed the licensee's evaluation of the spill/leak incident and the licensee's remedial actions, including the associated projected dose to the public as applicable. Additionally, the inspectors reviewed the licensee's "Groundwater Flow Characteristics Report Davis-Besse Nuclear Power Station," dated January 16, 2007, describing a preliminary evaluation of site characteristics to support the design of groundwater well network for monitoring potential releases of radioactivity to groundwater at the site. Additional reviews were limited to the preliminary evaluation because the licensee had not reached a full implementation of the groundwater monitoring program for detecting potential leaks and spills. The licensee scheduled the implementation of the groundwater protection program at the end of the fiscal year of 2007.

The inspectors reviewed the results of the radio chemistry inter-laboratory cross-check comparisons for the past calendar quarters in order to validate the licensee's analyses capabilities. The inspectors reviewed the licensee's evaluation of the inter-laboratory comparisons and the associated corrective actions for any deficiencies identified. In addition, the inspectors reviewed the inter-laboratory comparison data for the licensee's laboratory vendors for 2005 and 2006, to assess the analytical capabilities of the vendors to analyze the difficult-to-detect nuclides specified in the ODCM. The inspectors assessed whether the licensee's audits met the requirements in RETS/ODCM.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.3 Ventilation Filter Testing

a. Inspection Scope

The inspectors reviewed the most recent results for ventilation system filter testing to determine whether the test methods, frequency, and test results met TS requirements. Specifically, the inspectors reviewed the results of in-place high efficiency particulate air (HEPA) and charcoal absorber penetration/leak tests, laboratory tests of charcoal absorber methyl iodide penetration and in-place tests of pressure differential across the combined HEPA filters/charcoal absorbers.

This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports, and Special Reports related to the radioactive effluent treatment and monitoring program since the last inspection to determine if identified problems were entered into the corrective action program for resolution. The inspectors also assessed whether the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also reviewed corrective action reports from the radioactive effluent treatment and monitoring program since the previous inspection, interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

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

This review represented one inspection sample.

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 PI listed below for the period from the second quarter 2006 through February 2007. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 04, were used to verify the basis in reporting for each data element.

- Unplanned Scrams per 7000 Critical Hours

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

This review represented one sample of the PI listed above.

Cornerstone: Barrier Integrity

The inspectors sampled licensee submittals for the one PI listed below for the period from the second quarter 2006 through February 2007. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 04, were used to verify the basis in reporting for each data element.

- Reactor Coolant System (RCS) Leakage

The inspectors reviewed portions of licensee daily reports to find historical data related to RCS leakage and reviewed if the values reported for the PI was consistent with the daily reported data.

This review represented one sample of the PI listed above.

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 corrective action program. This screening was accomplished by reviewing documents entered into the corrective action program and by reviewing document packages prepared for the licensee’s daily Management Alignment and Ownership Meetings.

b. Findings

No findings of significance were identified.

.2 Annual Sample: Review of Issues

a. Inspection Scope

The inspectors reviewed CR 07-13923, “Lost Power To MU59C,” and the associated evaluations by the licensee. The inspectors reviewed the appropriateness of the licensee’s actions to address the issues associated with the loss of valve indication in the control room to a containment isolation valve. The licensee also reviewed CR 07-14083, "Bad Fuse Clip In Breaker BE1177 Cubicle For MU59C," for extent of condition and past operability. Additionally, because the licensee initially classified the issue as a significant condition adverse to quality, but then downgraded the issue to a condition adverse to quality, the inspectors reviewed the appropriateness of the downgrade and licensee compliance with corrective action program requirements.

This review represented one annual inspection sample.

b. Findings and Observations

On February 3, 2007, during a routine control room panel walkdown, the licensee discovered that containment isolation valve MU59C, which provides containment isolation for RCP 1-1 seal return, had lost indication. The licensee investigation revealed that the power had been lost to the valve in the open position. The licensee entered TS 3.6.3.1 for containment isolation. The licensee did not have a verification of the status of the breaker supplying power to the valve, therefore licensee personnel cycled the breaker and indication to MU59C was restored. The licensee exited the TS action statement and declared the valve operable.

On February 6, 2007, during further investigation of the issues identified with MU59C, the licensee discovered that the fuse clip for breaker control power inside breaker cubicle BE1177 for MU59C was loose. Movement of the clip caused the indication light for the valve in the control room to lose power and then regain power. The licensee declared the valve inoperable and replaced the fuse clip. After completion and testing of the work, the breaker was closed to re-energize the valve, and the valve was declared operable.

After activities to repair the breaker fuse clip, the licensee wrote a CR to determine the extent of condition and past operability of the valve. Examination and testing of the defective fuse clip at the licensee's offsite test shop determined the failure mechanism to be caused by a loose rivet holding the wire lug piece to the fuse clip piece of the fuse holder. This degraded condition was believed to be present from the manufacturing process.

The licensee's offsite test shop also determined that the loss of indication in the control room was attributed to an oxidation layer on the fuse clip. During testing when the inrush current to the contactor was passed through the fuse clip, it was determined that this was sufficient to break down the oxide layer in the field condition. The licensee concluded that the design function was still maintained, and there was no hardware issues with past operability for this particular condition, including the time from when breaker BE1177 was cycled on February 3, 2007 until the fuse clip was replaced on February 6, 2007. The licensee also evaluated the as-found condition for a seismic event and determined that there would be no impact to the seismic qualification of the fuse holder due to the low mass of the fuse. The licensee performed an extent of condition review and determined this type of fuse holder was used in 735 applications at the plant. The licensee review of their supply records for the last 10 years revealed that only six of these fuse clips had been drawn from their supply system during that time period. They also determined that within the entire nuclear industry review of the thousands of applications of these fuse clips, only two reported failures existed, and only one of the conditions was due to the loose rivet similar to the condition found in breaker BE1177.

The licensee initially classified the fuse clip degraded condition as a significant condition adverse to quality since initial indications were that the installed fuse clip may not have allowed the valve to perform its intended function. As such, the licensee's corrective action program required performance of a root cause evaluation. When it was determined that the degraded fuse clip condition would not have prevented the valve from performing its intended function, the licensee downgraded the issue to a condition adverse to quality. Under the licensee's program, a root cause was not required to review an issue classified as a condition adverse to quality. The licensee performed a limited apparent cause evaluation.

c. Conclusions

No findings of significance were identified. The licensee's program provided a means for identifying and prioritizing the MU59C indication problem identified in the control room, highlighting the item to plant management, and tracking the item until it was corrected and extent of condition and past operability was properly determined.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 05000346/2006-001-02: EDG Engine Damaged Due to Improper Torquing of Lock Nut

On January 20, 2006, the licensee identified damage to the Number 4 cylinder valve bridge of EDG-2. The damage was identified while investigating a tapping noise, following engine overhaul, that was not investigated when first heard. The event was documented by the licensee in LER 05000346/2006-001-00 and by the NRC in Inspection Report (IR) 05000346/2006003. Inspection Report 05000346/2006003 also provided the results of the NRC's SDP Phase 2 and Phase 3 analyses, documented NCV 05000346/2006003-03, and closed the LER.

Licensee Event Report 05000346/2006-001-01, submitted in May 2006, presented the results of an evaluation into the decision that the EDG-2 tapping noise, when first heard, did not warrant immediate investigation. The licensee determined that a structured problem solving process, which was not used, would have increased the probability of the tapping noise issue being investigated more promptly. A procedure, outlining a structured problem solving process, became effective in June 2006. The inspectors' review of the licensee's investigation and the procedure with the more structured process did not identify any items of significance that had not been addressed in previous inspection reports and was documented in Inspection Report 05000346/2006004.

Licensee Event Report 05000346/2006-001-02, submitted in October 2006, provided information that there were no component manufacturing defects that might have contributed to the event. Additionally, the revision provided an update of the licensee's progress in implementing procedure and process changes designed to minimize the probability of recurrence. The inspectors' review of the licensee's procedure and process changes did not identify any items of significance that had not been addressed in previous inspection reports. This LER is closed.

This review represented one inspection sample.

4OA5 Other Activities

.1 Licensee Activities and Meetings

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

- Corporate Nuclear Review meetings and activities on February 7 and 9, 2007;
- Davis-Besse Integrated Performance Assessment Collegial Review on February 5, 2007;
- Corrective Action Review Board meeting on February 26, 2007;
- Station ALARA Committee meeting on March 9, 2007; and
- Station Health Committee on March 14, 2007.

No items of significance were identified.

.2 (Closed) Unresolved Item (URI) 0500346/2006005-01: Simulator Malfunction Test Potentially Not Being Performed

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 55.46(d)(1), "Continued assurance of simulator fidelity," when the facility licensee failed to conduct a simulator "Generator Trip" malfunction performance test in a manner sufficient to ensure that simulator fidelity had been demonstrated and met. The "Generator Trip" malfunction performance test is one of 25 tests (item number 16) required by Section 3.1.4 in ANSI/ANS-3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training." The facility licensee is committed to adhering to the requirements of this standard. Specifically, the licensee failed to adequately conduct the required "Generator Trip" malfunction performance testing to ensure that simulator fidelity was demonstrated and met to allow conduct of the generator trip evolution.

Description: On November 21, 2006, the inspectors identified an issue concerning the failure to comply with 10 CFR 55.46. Specifically, the issue concerned the adequacy of the licensee's simulator performance testing conducted in accordance with 10 CFR 55.46(d)(1). The licensee had committed to operate and maintain the plant-referenced simulator and conduct periodic performance tests, including malfunction tests, in accordance with ANSI/ANS-3.5-1998, "American Nuclear Society Nuclear Power Plant Simulators for Use In Operator Training and Examination." Section 4.1.4, "Malfunctions," of the standard required that it be demonstrated that simulator response during the conduct of the malfunctions required by Section 3.1.4, "Malfunctions," meet certain acceptance performance criteria.

The inspectors identified that the licensee's simulator testing procedure associated with Section 3.1.4, item number (16), "Generator Trip," did not meet the requirements of 10 CFR 50.46(d)(1) to provide continued assurance of simulator fidelity. The inspectors identified that the licensee's simulator "Generator Trip" test procedure "Davis-Besse Nuclear Power Station Simulator Training Certification Test T16, Generator/Turbine Trip," Revision 4, set up an initial plant condition that resulted in a turbine trip as an initiating event, rather than a generator trip, with the simulation being stopped immediately after the turbine trip. The performance test did not provide a demonstration of expected plant response to a generator trip as an initiating event as required by the standard nor to allow proper conduct of the generator trip evolution. The standard clearly required a "Turbine trip" malfunction (item number 15) and a "Generator Trip" malfunction (item number 16). These malfunctions are considered separate and independent from one another, and as such, must be performance tested on their own merit. Thus, simulator malfunction performance testing, with regard to "Generator Trip" malfunction testing, was not being conducted.

Following identification of this issue, the licensee entered the issue into their corrective action program as CR 06-10403. The licensee revised the "Generator Trip" test procedure, "Davis-Besse Nuclear Power Station Simulator Training Certification Test T16, Generator/Turbine Trip," on November 29, 2006, and adequately completed simulator performance testing for the generator trip malfunction on November 30, 2006.

The inspectors determined that the failure to conduct a simulator "Generator Trip" malfunction performance test in a manner sufficient to ensure that simulator fidelity had been demonstrated was a performance deficiency warranting a significance determination.

Analysis: The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." This finding affected the Mitigating Systems cornerstone of Reactor Safety because it could affect the capability of the simulation facility to adequately meet the requirements to administer initial operator license examinations and provide continuing training of licensed operators in accordance with 10 CFR Part 55, "Operators' Licenses." The safety significance of this issue was more than minor due to potential negative training. The realistic potential of providing negative training based on significant simulator deficiencies compared to the actual plant, including inadequate testing of the simulator to assure that the simulator appropriately replicated the actual plant, could potentially affect operator actions on the actual plant.

The inspectors reviewed this issue in accordance with Manual Chapter 0609, "Significance Determination Process (SDP)," Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspectors found that the finding was related to simulator fidelity and that the simulator did not meet the performance requirements of 10 CFR 55.46 that had the potential impact of providing negative training. Based on this SDP, the inspectors determined that this finding was of very low safety significance (Green), because although the potential for negative training was apparent, the discrepancy was on the simulator, and no event occurred on the actual plant due to the potential negative training. This issue was referred to the NRC Headquarters Operator Licensing Branch and Headquarters concurred with the above analysis.

Enforcement: 10 CFR 55.46(d)(1) required the licensee to conduct simulator performance testing in a manner sufficient to ensure that simulator fidelity was demonstrated and met throughout the life of the simulator. The licensee committed to follow ANSI/ANS-3.5-1998, "American Nuclear Society Nuclear Power Plant Simulators for Use In Operator Training and Examination," as the way they would meet 10 CFR 55.46 (d)(1). The ANSI/ANS-3.5-1998 standard required testing under Section 4.1.4, "Malfunctions." Section 4.1.4 required the performance of simulator malfunction performance testing in accordance with Section 3.1.4, "Malfunctions." In Section 3.1.4, the licensee was required to perform testing associated with item number (16), "Generator Trip." Contrary to the above, the inspectors identified that the licensee's simulator testing procedure, associated with Section 3.1.4, item number (16), "Generator Trip," did not meet the requirements of 10 CFR 50.46(d)(1) to provide continued assurance of simulator fidelity. In particular, the "Generator Trip" test procedure "Davis-Besse Nuclear Power Station Simulator Training Certification Test T16, Generator/Turbine Trip," Revision 4, did not provide a demonstration of a generator trip evolution as an initiating event as required.

This finding is considered a violation of 10 CFR 55.46(d)(1). However, because of the very low safety significance and because the issue has been entered into the licensee's corrective action program (CR 06-10403), the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2007002-01). Unresolved Item 05000346/20060005-01 is closed.

.3 Operation of an Independent Spent Fuel Storage Installation (ISFSI) - Routine Monitoring Activities (60855.1)

a. Inspection Scope

The inspectors reviewed the licensee's surveillance procedure associated with monitoring the Horizontal Storage Modules (HSMs) performance to ensure the requirements specified in the Certificate of Compliance (CoC), the TSs, and the Site Safety Analysis Report were met. The inspectors reviewed select surveillance logs to verify that the temperatures of the concrete remained within long-term storage limits and no blockage of vents occurred in the past. The inspectors performed a walk down of the ISFSI pad and the HSMs to evaluate their condition. The inspectors observed the licensee perform routine surveillance activities, including inspection of the vent screens and taking thermocouple readings.

The inspectors reviewed plant procedures to verify that the licensee had adequate controls in place to monitor the dose resulting from operation of the ISFSI. The inspectors reviewed a plant procedure associated with performing surveys. The inspectors also reviewed a number of routine surveys performed in the vicinity of the HSMs to ensure the radiation dose levels were within the TSs limits. The inspector reviewed the plant procedure for monitoring dose from plant day-to-day activities, including the use of thermo-luminescent dosimetry (TLD) and the results of the TLD quarterly readings, to verify the radiation dose levels from the HSMs were within regulatory limits. The inspectors also reviewed the results of the 2006 Annual Radiological Report.

The inspectors reviewed the licensee's procedures associated with material control, accountability and records management. The inspectors verified that the licensee maintained duplicate records of fuel stored in Dry Shielded Canisters (DSCs). The inspectors reviewed the licensee's site annual inventory record for 2006 to verify the spent fuel in the DSCs was controlled and accounted for. The inspectors also verified that the licensee had a procedure to unload a canister if necessary.

The inspectors evaluated a number of 10 CFR 72.48 screenings associated with temporary storage of reactor coolant pump motors and sealand containers on the ISFSI pad to ensure no adverse changes were made that could affect the design or function of the HSMs. The inspectors also reviewed the plant Emergency Plan and select emergency procedures to evaluate their adequacy in regard to ISFSI.

b. Findings

No findings of significance were identified.

.4 Operation of an Independent Spent Fuel Storage Installation (ISFSI) - Corrective Actions Taken for Storage of Combustible Material on the ISFSI Pad (60855.1)

a. Inspection Scope

The inspectors reviewed the corrective actions that the licensee implemented in response to a violation issued in the August 11, 2006 Integrated Inspection Report, No. 05000346/2006003. The violation involved the failure to control combustible material near the HSMs in accordance with procedures.

b. Findings

Introduction: The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of very low safety significance for failure to comply with the requirements of the NRC issued Certificate of Compliance by failing to implement adequate corrective actions in response to a violation to control transient combustible material on the dry fuel storage pad. As of the date of the inspection associated with dry fuel storage, the licensee did not take adequate corrective actions to comply with its fire protection procedure.

Description: During a routine inspection associated with operation of the ISFSI, the inspectors reviewed the licensee's corrective actions taken in response to a previous NRC-identified violation associated with the dry fuel storage area. On April 10, 2006, the resident inspectors identified a violation involving the licensee's failure to follow the plant fire protection procedure for control of transient combustible material stored on the ISFSI pad in close proximity to the HSMs (IR 05000346/2006003). During the routine dry fuel storage inspection, the inspectors toured the ISFSI pad and inquired about the content of sealands that were stored on the pad as close as 15 feet from the HSMs. Upon further investigation, the licensee identified five additional sealands that contained combustible material and were located within the 50-foot restricted zone of the HSMs. As of February 28, 2007, the licensee did not remove all combustible material within the 50-foot restricted zone as specified in its plant fire protection procedure. The licensee generated condition report CR 07-15336 to document the issue and the corrective actions needed to be in full compliance with the NRC issued CoC and the plant fire protection procedure. The licensee took immediate corrective actions which included: 1) relocation of all sealands outside of the 75-foot perimeter; 2) thorough physical inspection and inventory of the relocated sealands and verification of the inventory database; and 3) limited access to the ISFSI pad and control of material stored on the pad.

Analysis: After further discussion and review of documents, the inspectors determined that the licensee did not take adequate corrective actions to correct a condition adverse to quality in order to restore full compliance with the 10 CFR Part 72 licensee (CoC) and its plant fire protection procedure as required by the site Quality Assurance Program. The lack of thorough corrective actions was a performance deficiency that warranted a significance evaluation. This finding was greater than minor because the lack of adequate corrective actions resulted in a more significant safety concern since the prolonged presence of combustible materials within 50 feet of HSMs for approximately 10 months increased the vulnerability of the HSMs to a fire. In addition, the lack of adequate corrective actions had the potential to become a programmatic issue and could have adversely affected the NRC regulatory oversight and enforcement processes, as the agency relied on the licensee's adequacy of corrective actions to correct an NRC-identified violation. The inspectors determined that the finding was not suitable for SDP evaluation because the noncompliance involved 10 CFR Part 72 dry fuel storage activities. Therefore, this finding was reviewed by Regional Management and dispositioned using traditional enforcement. The finding was determined to be of very low safety significance. The combustible material was contained within metal containers which could have mitigated the spread of a potential fire. Also, the plant fire brigade could have been dispatched to extinguish a fire, involving the transient combustible material, before the HSMs incurred significant damage. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution because licensee personnel failed to thoroughly evaluate the problem (P.1(c)).

Enforcement: Certificate of Compliance, No. 1004, Revision 0, Condition 1.1.3 states, in part, that "activities at the ISFSI shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 50, Appendix B, and which is established, maintained, and executed with regard to ISFSI." 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that "measures shall be established to assure that conditions adverse to quality such as failures are promptly identified and corrected."

Contrary to the above, from April 2006 to February 28, 2007, the licensee failed to adequately implement corrective actions to correct a condition adverse to quality. Specifically, the licensee failed to ensure that there were no containers holding combustible materials within 50 feet of the dry fuel storage modules as described in licensee's procedure DF-FP-00007 for a period of approximately 10 months. The results of this violation were determined to be of very low safety significance; therefore, this violation was classified as a Severity IV Violation of the Certificate of Compliance, No. 1004, Condition 1.1.3, Quality Assurance and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." Upon identification of the presence of combustible material on the pad, the licensee entered this issue into the corrective actions program as CR 07-15336 on February 28, 2007. Because this violation was of very low safety significance, non-willful, non-repetitive, and documented in the licensee's corrective actions program, this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000346/2007002-02).

.5 (Closed) VIO 0500346/2003010-02 Service Water Discharge Check Valve Test Acceptance Criteria

In 2002, NCV 05000346/2002014-03 was issued to Davis-Besse Nuclear Power Station for failure to take proper corrective actions. The licensee did not address an inadequate acceptance criteria used for the service water pump discharge check valves in-service full flow test when it was determined to be non-conservative.

This issue was entered in the licensee's corrective action program as CR 02-07657. As a result of this condition report, the licensee determined that the valves were full-open at flows greater than 7270 gallons per minute (gpm) and no corrective actions to the procedure were necessary. The inspectors noted that the licensee's evaluation of the flow rate at which the valves were fully open could not be reproduced as it relied on verbal information provided by the vendor. The inspectors concluded that the licensee's evaluation was inadequate to ensure that the check valves were in the full open position. As a result, in 2003, NRC issued a Notice of Violation (VIO 05000346/2003010-02) of 10 CFR 50.55(a) which requires, in part, that licensees perform in-service testing of valves per ASME Operation and Maintenance Code and applicable addenda. The licensee's corrective actions were to revise the procedure and document the basis for the check valve test acceptance criteria.

During this inspection period, the inspectors reviewed the licensee's response to the Notice of Violation and the additional information requested. The inspectors noted that the licensee established a flow acceptance criteria for the service water discharge check valves which met the design basis requirements. In addition, procedures were changed to permit throttling of flow to obtain the higher flow rates in accordance with the TSs.

Based on this review, this violation is closed.

.6 (Closed) Unresolved Item (URI) 05000346/2006006-02: Change to Design Basis Tornado Differential Pressure Design Limit for the Auxiliary Building

During the 2006 NRC inspection for Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications, the inspectors identified a URI concerning the adequacy of the licensee's basis, described in the 10 CFR 50.59 safety evaluation, for determining that changes to USAR Sections 3.3.2, "Tornado Criteria," and Section 3.8.1.1.1, "Auxiliary Building," with respect to the ability to withstand the effects of differential pressure during a tornado did not require a license amendment. The inspectors questioned whether the licensee's new calculation for the auxiliary building was a change in methodology that was not properly evaluated in accordance with 10 CFR 50.59.

The licensee entered the inspectors' concern into the corrective action program as CR 06-00246 and CR 06-00472. The licensee's evaluation concluded that the new calculation, C-CSS-099.20-029, was approved using Standard Review Plan (SRP) Section 3.3.2 tornado load combinations in lieu of the Davis-Besse USAR described load combination methodology. The methodology used in the new calculation did not conform to that described in Davis-Besse licensing and design bases. The calculation evaluated the auxiliary building for the revised loads, Section 6.4 of the calculation evaluated the exterior reinforced concrete walls using a tornado load combination methodology of 1.0 tornado wind + 0.5 tornado differential pressure + 1.0 tornado missile as specified in SRP Section 3.3.2. The Davis-Besse USAR Section 3.3.2.2 described the combination of these loads as 1.0 tornado wind + 1.0 tornado differential pressure. As a result of this issue, the licensee revised calculation C-CSS-099.20-029 to conform with the USAR described methodology for the combination of tornado wind and depressurization loads. The revised calculation determined that the subject walls were adequate for the applied loads.

Based on the inspectors' review of the licensee's evaluations and corrective actions, the inspectors determined that the licensee's failure to evaluate the change in methodology used per calculation C-CSS-099.20-029 in accordance with the requirements of 10 CFR 50.59 was a performance deficiency. After discussion with the Office of Nuclear Reactor Regulation staff, the inspectors determined that a license amendment was not required in that the changes remained encompassed by the plant specific licensing acceptance criteria. Because the methodology used in the new calculation was conservative compared to the methodology described in the USAR, the inspectors determined that the finding was minor. Therefore, this URI is closed and no further enforcement action is necessary.

Because the inspection was counted in another inspection report, these inspection activities do not represent an inspection sample for this report.

.7 Evaluation of the 2006 Independent Corrective Action Program Assessment Final Report (95003)

a. Inspection Scope

On September 23, 2006, the licensee submitted the 2006 Independent Assessment Report for the Corrective Action Program (CAP) that included an action plan to address areas identified as needing improvement. The inspectors reviewed the report for consistency with assessment results presented at the assessment exit and debrief meetings and with original drafts of the report. Additionally, the inspectors verified that the report adequately covered areas identified in the assessment plan, that conclusions were consistent with and adequately supported by information in the report, and that the licensee developed action plans to properly address any Areas For Improvement.

b. Findings and Observations

The independent assessment of the CAP and the final report from that assessment addressed the following areas:

- Status of corrective actions from the 2005 and 2004 independent assessment of the CAP;
- Review of CRs for accuracy of identification, classification, and categorization;
- Evaluation and resolution of problems;
- Corrective action implementation and effectiveness;
- Effectiveness of program trending;
- Effect of program backlogs;

- Effectiveness of internal assessment activities; and
- Evaluation of open corrective actions taken in response to NRC corrective action team inspection items (NRC IR 05000346/2003010).

The independent team concluded that the licensee's overall implementation of the CAP was Effective. "Effective" has the meaning that performance, programs, and processes are sufficient to obtain the desired results with consistency and effectiveness, but that there may be several specific areas where improvement is needed and potentially other items that need additional attention. Of the eight general areas assessed, seven we rated effective with the remaining area being rated as highly effective.

The assessment report also identified one continuing "Areas For Improvement." An Area For Improvement (AFI) is defined as an identified performance, program, or process element that requires improvement to obtain the desired results with consistency and effectiveness. The AFI was:

- Because implementation of the corrective actions to address equipment trending was not due until the end of February 2007, the independent team held the AFI open, made some progress since the last assessment, and equipment trending remained behind the industry in the ability to determine common equipment failure issues as well as predicting and preventing future equipment failures.

The licensee issued CR 06-06723 to further address the equipment trending issue. The inspector reviewed the CR and the associated corrective actions. Based on discussions with licensee site and corporate personnel, the corrective actions are on track to be implemented by the end of February 2007.

c. Conclusions

The licensee complied with the year 2006 requirement for an Independent Assessment of the Corrective Action Program as described in the NRC's March 8, 2004, Confirmatory Order. The results of the assessment, including the overall assessment, appear consistent with the information reviewed and documented in the final report. The licensee's action plan for the open AFI appear reasonable and, if implemented properly, could increase the effectiveness of the licensee's equipment trending program. The overall independent assessments of the CAP and the assessment of individual areas of the CAP appear consistent with NRC inspection findings associated with the licensee's CAP. No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

On March 29, 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:

- Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Program inspection with Mr. C. A. Hawley, Manager of Site Projects, and other staff members on January 26, 2007.

- Reviewing the 2006 Independent Corrective Action Program Assessment Final Report with C. Price, Director-Performance Improvement, and other staff members on February 1, 2007.

- Reviewing URI 05000346/2006005-01 with Mr. C. Steenbergin on February 9, 2007.

- NRC onsite inspection and in-office review associated with dry fuel storage with Mr. C. Price, Director-Performance Improvement, and Mr. J. Grabnar, Director-Engineering, and other staff members on March 28, 2007.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and was a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Technical Specification 6.12.1.b requires that access to each high radiation area shall be controlled by means of a Radiation Work Permit (RWP) that includes specification of radiation dose rates and other appropriate radiation protection equipment and measures. Contrary to this, on February 8, 2007, a maintenance electrician entered the mechanical penetration room 2 using a RWP that did not address entry to a high radiation area. Mechanical penetration room 2 was, at the time of the entry, posted as a high radiation area. This event was documented in CR 07-14230 and CR 07-14436. The finding was of very low safety significance because it did not involve a substantial potential for an overexposure and the licensee's ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

<u>SUPPLEMENTAL INFORMATION</u>

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>

M. Bezilla, Site Vice President
B. Boles, Director, Maintenance
J. Grabnar, Director, Engineering
R. Hruby, Manager, Nuclear Oversight
V. Kaminskas, Director, Plant Operation
J. Rinckel, Vice-President, Fleet Oversight
S. Plymale, Manager, Plant Engineering
C. Price, Director, Performance Improvement
T. Stallard, Superintendent, Operations Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Open and Closed</u>

05000346/2007002-01	NCV	Failure to Conduct Simulator Malfunction Performance Testing in a Sufficient Manner to Demonstrate Fidelity
05000346/2007002-02	NCV	Inadequate Corrective Actions (ISFSI)

<u>Closed</u>

05000346/2006-001-02	LER	Emergency Diesel Generator Engine Damaged Due to Improper Torquing of Lock Nut
05000346/2006005-01	URI	Simulator Malfunction Test Potentially Not Being Performed
05000346/2006006-02	URI	Change to Design Basis Tornado Differential Pressure Design Limit for the Auxiliary Building
05000346/2003010-02	VIO	Service Water Discharge Check Valve Test Acceptance Criteria

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.

1R01 Adverse Weather Protection

CR 07-14519; Loss of Offsite Communications Capability
CR 07-14523; Radwaste Ventilation Supply Fan C13 Won't Start
RA-EP-02870; Station Isolation; Revision 03

1R04 Equipment Alignment

Clearance EDB-SUB065-01-013; BF1168, Make-Up Pump 2 Auxiliary Gear Lube Oil Pump P372D, AUXL7-565-236 (F11C)
Clearance EDB-SUB065-01-014; MP37-2A, Motor for Make-Up Pump 1-2, AUXL7-565-225
Clearance EDB-SUB065-01-015; PSMU102A, Reactor CLNT MK-UP 2 GR BRNG Lube Oil Press Switch, AUXL7-565-225
Clearance EDB-SUB065-01-016; MU36, Make-Up Pump 2 Seal Water Vent, AUXL7-565-225 MUP RM
CR 07-13482; Abandoned Clevis on CCW Line 6" HBC-27 Requires Removal (NRC Identified)
DB-MM-09266; Mechanical Maintenance Procedure, Torquing; Revision 07
DB-OP-06006; Makeup and Purification System; Revision 17
DB-OP-06012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 28
DB-OP-06233; Auxiliary Feedwater System; Revision 20
DB-OP-06316; Diesel Generator Operating Procedure; Revision 33
Drawing E302A, Sheet No. 257; Class 1 Conduit Support; Revision 10
Drawing M-31A; P&ID Make Up and Purification System; Revision 46
Drawing M-31B; P&ID Make Up and Purification System; Revision 29
Drawing M-31C; P&ID Make Up and Purification System; Revision 37
Drawing OS-041A SH 2; Operational Schematic Emergency Diesel Generator Systems; Revision 25
Drawing HL-236H; Component Cooling System Supply and Return Emergency Diesel Generators; Revision 03
Drawing OS-041B; Operational Schematic Emergency Diesel Generator Air Start/Engine Air System; Revision 31
Drawing OS-041C; Operational Schematic Emergency Diesel Generator Diesel Oil System; Revision 16
Drawing OS-004, Sheet 1; Operational Schematic Decay Heat Removal/Low Pressure Injection System; Revision 43
Drawing OS-17A, Sheet 1; Auxiliary Feedwater System; Revision 20
Drawing OS-17B, Sheet 1; Auxiliary Feedwater Pumps and Turbines; Revision 23
DB-OP-06316; Emergency Diesel Generator 1 Checklists; Revision 33

1R05 Fire Protection

CR 07-15527; Barrier 500N/EXT Degraded (NRC Identified)
Davis-Besse Nuclear Power Station Fire Hazard Analysis Report
DB-FP-00007; Control Of Transient Combustibles; Revision 08
DB-FP-00009; Fire Protection and Fire Watch; Revision 08
DB-ME-09323; Emergency Lighting System Preventive Maintenance; Revision 01
Drawing A-221F; Fire Protection General Floor Plan EL 545'-0" & 555' -0"; Revision 08
Drawing A-222F; Fire Protection General Floor Plan EL 565'-0"; Revision 13
Drawing A-223F; Fire Protection General Floor Plan El. 585'-0"; Revision 18
Drawing A-224F; Fire Protection General Floor Plan El. 603'-0"; Revision 21
Drawing A-225F; Fire Protection General Floor Plan El. 623'-0"; Revision 14
PFP-AB-237; Protected Area Pre-Fire Plan, Room 237; Revision 3

1R06 Flood Protection

Calculation 58.8; Flood Level in AFP [Auxiliary Feed Pump] Rooms Due to Serious Line Break; June 1984
RA-EP-02880; Internal Flooding; Revision 03
DB-OP-02010; Feedwater Alarm Panel 10 Annunciators; Revision 08
DB-OP-06226; Startup Feed Pump Operating Procedure; Revision 09
Drawing M-006D; Auxiliary Feedwater System; Revision 51
Drawing OS-022A; Turbine Plant Cooling Water System; Revision 19
Drawing OS-012A, Sheet 1; Main Feedwater System; Revision 23
Drawing HL-206G; Auxiliary & Start-Up Feed Pumps Suction and Recirculation Hanger Locations; Revision 06
Davis-Besse Letter to USNRC Serial 1070; Startup Feed Pump Operation; October 18, 1984
WO 200117130; PM 1696 - Inspect/Verify AF70 & AF71 Operability

1R07 Heat Sink Performance

Document ESM-99-002; Effect of UHS [Ultimate Heat Sink] Pond Siltation on Service Water Intake Structure; Revision 00
CR 07-13945; Lost Connection Between Lake Erie and the Intake Forebay Due to Frazil Ice
CR 07-13947; Equipment Problems During Frazil Ice Compensatory Actions
Calculation C-ICE-009.01-002; Ultimate Heat Sink Level; July 11, 2003
SD-018; System Description for Service Water System; Revision 03
Drawing TECW 201 BI, Sheet 8; Intake Crib; December 20, 1972
USAR Section 9.2.5.1; Loss of Intake Canal; Revision 25
Calculation 12501-M-001; UHS Pond Thermal Performance Analysis; Revision 00
DB-OP-06913; Seasonal Plant Preparation Checklist; Revision 16

1R11 Licensed Operator Requalification Program

DBBP-TRAN-0017; Conduct of Simulator Training; Revision 02
DBBP-OPS-0034; Control Room Crew Composition; Revision 01
ORQ-EPE-S123; Failed NI [Nuclear Instrument], Loss of Secondary SW [Service Water] and SGTR [Steam Generator Tube Rupture] with Failed Open TBV [Turbine Bypass Valve]; Revision 08

1R12 Maintenance Effectiveness

Calculation C-ME-050.01-004; Component Level Review Calculation for AOV MS5889A/B; Revision 02
Calculation C-NSA-50.03-014; Effects of Silt in AFP2 Service Water Suction Piping; Revision 00
Calculation C-NSA-099.16-025; MOV [Motor-Operated Valve] Risk Ranking; Revision 01
CR 03-08893; Reduction In Motor Operated Valve (MOV) Output Thrust/Torque Capability
CR 06-01075; Improperly Open Torque Switch Bypass For SW1379
CR 06-01676; C-ME-050.01-004, Component Level Review Calculation for MS58889A/B and Test Discrepancy
CR 06-8728; MOV Program Snapshot Self Assessment Area For Improvement
CR 07-15029; MS107 Set Outside Target Thrust Range (NRC Identified)
CR 07-15030; Quality Record Incomplete/Inaccurate (NRC Identified)
ISTB3; Pump Valve Basis Document Volume III Stroke Time Basis; Revision 29
WO 7-95-0355-18; CC1411B CCW To Containment Isolation Valve
NOBP-ER-3601A; Motor Operated Valve Program Torque/Thrust Requirement and Actuator Capability; Revision 00
NOP-ER-3601; Motor Operated Valve Program Overview; Revision 01
NOP-SS-2101; Engineering Program Management; Revision 02
NORM-ER-3601; Valve-Motor Operated; Revision 05
WO 99-003801-000; MV106A Main Steam Line 2 to AFW Pump Turbine 1-1 Isolation Valve
WO 01-002204-000; HP2B HPI Line 2-2 Isolation Valve
WO 02-002313-000; DH7B BWST Isolation Valve (Line 1)
WO 02-005850-000; DH9B DH [Decay Heat] Pump 1-1 Suction From Emergency Sump Valve
WO 200001169; SW1399 TPCW Heat Exchanger Inlet Header Isolation
WO 2000553669; DB-HVDH63; Decay Heat Pump 1-2 Discharge To HPI Pump 1-2
WO 200062581; SW1395 TPCW Heat Exchanger Inlet Header Isolation
WO 200116941; DB-MV-1407A CCW Outlet Isolation Valve 1 From Containment
WO 200117051; DB-MV107 Main Steam Line 2 to Auxiliary Feed Pump Turbine 1-2 Isolation
Quarterly Program Health Report, Motor Operated Valve Program; 2006-1-DB
Quarterly Program Health Report, Motor Operated Valve Program; 2006-2-DB
Quarterly Program Health Report, Motor Operated Valve Program; 2006-3-DB
Quarterly Program Health Report, Motor Operated Valve Program; 2006-4-DB
Drawing M-006D; Auxiliary Feedwater System; Revision 51
Davis-Besse Nuclear Power Station Motor Operated Valve Program Documentation
DB System Health Report; Auxiliary Feedwater System Window; Fourth Quarter, 2006
Maintenance Rule Program Manual; System Scoping Sheets for Auxiliary Feedwater; Revision 22

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

C-NSA-099.16-093; Small Break LOCA Accident Sequence Criteria “Cases 3 & 4" (0.003 & 0.02 Ft^2 (1HPI(no PB; 60 min delay), 1HPR, No PROV, with and without AFW)
CR 07-15275; Loose Connections Found In Cabinet S618
CR 07-15335; NRC Concern With Available Operator Actions Use (NRC Identified)
CR 07-15441; K5-1 Electrical Connection Tightness Checks
CR 07- 16753; EDG #1 Fuel Oil Transfer Pump Failed to Start
CR 07-16792; BE1298 Installed Starter Found to be Different than Required by Drawing
DBBP-OPS-0003; On-Line Risk Management Process; Revision 05
Maintenance Risk Summaries for the Week of January 21, 2007; Revisions 00 and 01
Maintenance Risk Summaries for the Week of February 11, 2007; Revision 00

Maintenance Risk Summaries for the Week of February 25, 2007; Revision 00 through 02
Maintenance Risk Summaries for the Week of March 18, 2007; Revision 00
Operations Evolution Order; Stroke DH7B, WO200192721; February 13, 2007
Work Implementation Schedule, Subsystem Sort; January 21, 2007
Work Implementation Schedule, Subsystem Sort; February 11, 2007
Work Implementation Schedule, Subsystem Sort; February 25, 2007
Work Implementation Schedule, Subsystem Sort; March 18, 2007
DB-SC-03023; Off-Site AC [Alternating Current] Sources Lined Up and Available; Revision 18
DB-SC-04000; Station Blackout Diesel Generator Lined-Up To Supply Essential Bus; Revision 02

1R15 Operability Evaluations

Calculation C-CSS-024.02-006; Evaluation of EDG Outlet Ventilation Dampers for Past Operability of Exposure to Tornado Differential Pressure Loads; Revision 00
Calculation C-NSA-049, 02-024; DHP O/B Bearing Reservoir Allowable Oil Leakage; Revision 01
Calculation C-NSA-52.01-012; Maximum Allowable Leak Rate Through HP31/32 or ECCS Systems; Revision 00
Calculation C-NSA-059.01-019; Water Level Inside Containment Post LOCA; Revision 01
CR 06-11269; CDBI-EDG Vent Dampers May Not Be Structurally Adequate For Design Tornado Differential Pressure
CR 06-11421; CDBI-LVSGR Vent Dampers May Not Be Structurally Adequate For Design Tornado DP [Differential Pressure]
CR 06-11483; CDBI-CCW Vent Dampers May Not Be Structurally Adequate For Design Tornado DP
CR 06-11740; CDBI-LVSGR Exhaust Vent Dampers May Not Be Structurally Adequate For Design Tornado DP
CR 07-12189; Unexpected Trip of CREVS [Control Room Emergency Ventilation System] Train 2 Compressor During Monthly Test
CR 07-13103; Oil Leak from MP2-2 Sight Glass
CR 07-15233; Post-LOCA Containment Water Level
Davis-Besse Nuclear Power Station Quality Classification List; Revision 17
Davis-Besse Nuclear Power Station Lubrication data Sheet; Decay Heat Pumps and Motors; Revision 22
DB-PF-03020; Service Water Train 1 Valve Test; Revision 20
DB-PF-03011; ECCS Integrated Train 1 Leakage Test; Revision 05
DB-PF-03012; ECCS Integrated Train 2 Leakage Test; Revision 07
DB-OP-06015; Borated Water Storage Tank Operating Procedure; Revision 11
DB-OP-06316; Diesel Generator Operating Procedure, Attachment 13; Revision 32
Drawing C-0903; Emergency Sump Plan and Sections; Revision 00
Drawing C-0906; Perforated Plate Details; Revision 00
Drawing HL-241D; Service Water System, Auxiliary Building - Return Piping; Revision 11
Drawing OS-20, Sheet 1; Service Water System; Revision 70
Drawing OS-0032B; Control Room Emergency Ventilation System; Revision 16
ECR 05-0086-00; Containment Air Coolers Service Water Modifications; Revision 12
NOP-OP-1009; Immediate and Prompt Operability Determination; Revision 00
RA-EP-02810; Tornado; Revision 06
SD-018; Service Water System; Revision 03
SD-029B; Control Room Emergency Ventilation; Revision 03
Pump and Valve Basis Document Volume III Stroke Time Basis
Manual ISTB2, Volume II; Pump and Valve Basis Document

1R17 Permanent Plant Modifications

Calculation C-ICE-045.01-001; Davis-Besse Heat Balance Uncertainty Calculation; Revision 01
CR 02-03233; Modification Package 99-0047 Deficiencies
CR 07-16283; Vendor Identified Additional Uncertainty For Caldon LEFM [Leading Edge Flow Meter] System
CR 07-16466; Caldon Main Feedwater Transducers Path Failure
Davis-Besse NPS Evaluation for LAR [License Amendment Request] 05-007 for Measurement Uncertainty Recapture (MUR) Power Uprate
Davis-Besse MUR Summary Report Attachment A; dated March 16, 2007
DB-OP-02010; Feedwater Alarm Panel 10 Annunciators; 10-4-A, MFW Flow CALDON SYS TRBL; Revision 09
Design Report For Normal Modification (MOD) 99-0047-00 Install Feedwater Flow Rate Caldon (LEFM) Check Plus System; Revision 02
Design Report For Normal Modification (MOD) 99-0047-01 Install Feedwater Flow Rate Caldon (LEFM) Check Plus System Supplement 01; Revision 02 and 03
ECP No. 06-0118-00; Modify Caldon Leading Edge Flow Meter Alarm Circuit; Revision 00
Engineering Report; ER-202; Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using the LEFM Check Plus System; July 2004

1R19 Post-Maintenance Testing

CR 07-12498; Packing Loads on AF3871 Exceed Current Design Values
CR 07-12506; AFPT [Auxiliary Feed Pump Turbine] #2 Speed Variations
CR 07-12963; Work Week 0702 AFW Train 2 Outage Critique
CR 07-12823; DB-SC-03255 SFRCS Overall Response Time Calculation Performed Incorrectly
CR 07-12947; NRC Comments On Order 200209529 (NRC Identified)
CR 07-13331; MU32 Is Stuck Open Slightly
CR 07-13530; EDG-1 Governor Oil Level Low
CR 07-15275; Loose Connections Found in EDG-2 Cabinet 3618
CR 07-15369; Minor Oil Leak on EDG #2
Drawing E-30B, Sheet 18; Control Switch and Pushbutton Details; Revision 8
Drawing E-65B, Sheet 13; Control Rod Drive Mechanism (CRDM) Control System CRDM Trip Breaker - B (Channel 1); Revision 2
Drawing E-65B, Sheet 13A; Control Rod Drive Mechanism Control System CRDM Trip Breaker - B (Channel 1); Revision 3
DB-MI-03011; Channel Functional Test of Reactor Trip Breaker B; Revision 15
DB-MI-03203; Channel Functional Test and Calibration of SFRCS Actuation Channel 1, Steam Generator Differential Pressure Inputs PDS-2686A, PDS-2686B, PDS-2685C and PDS-2685D
DB-MI-03223; Response Time Test of SFRCS Actuation Channel 1, Steam Generator Differential Pressure Inputs; Revision 06
DB-OP-06316; Diesel Generator Operating Procedure; Revision 33
DB-SC-03076; Emergency Diesel Generator 1 184-Day Test; Revision 14
DB-SC-03081; Emergency Diesel Generator 2 Overspeed Test; Revision 2
DB-SC-03255; SFRCS Overall Response Time Calculation; Revision 03
DB-SP-03160; AFP2 Quarterly Test; Revision 16
DB-SS-03042: Control Room Emergency Ventilation System Train 2 Monthly Test; Revision 04
WO 200055170; Adjust Packing for Valve AF-3871
WO 200188866; Main FW/SG2 Pressure Differential Switch
WO 200195533; Main FW/SG2 Pressure Differential Switch

WO 200209529; Control Room Safety Features Actuation System
WO 200233589; Turbine Drive Auxiliary Feed Pump 1-2
WO 200247062; CTRM Emergency Cond Unit 2 Low Oil Press
WO 200204767; Replace EDG 2 KPD13 Relays, CR 03-07197
WO 200249820; Normal Make-Up Line Valve Operator

1R22 Surveillance Testing

CR 07-13589; DB-SC-03111, SFAS Channel 2 Functional Test, Improvement Opportunity (NRC Identified)
CR 07-15812; Inspection Of Porcelain-Insulated Cable Clamps On Station Batteries (NRC Identified)
CR 07-15911; NRC Question - Maintenance Programs For 345 kV Switchyard Batteries (NRC Identified)
DB-ME-03001; Station Batteries Quarterly Surveillance (2N Battery); Revision 13
DB-ME-03001; Station Batteries Quarterly Surveillance (2P Battery); Revision 13
Drawing OS-004, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 43
DB-HP-01210; Neutron Dosimetry: Issue, Use and Dose Calculation; Revision 4
DB-MI-03059; RPS Channel 3 Calibration of Overpower, Power/Imbalance/Flow, and Power/Pumps Trip Functions; Revision 20
Davis-Besse Nuclear Power Station TSs; Section 3.3.2.1 and 3.8.1.1
DB-MI-03354; Channel Functional Test of Main Feed Pump 1 and 2 Turbine Hydraulic Oil Trip and Main Turbine Oil Trip ARTS Channel 4, Revision 5
DB-OP-01101; Containment Entry; Revision 6
DB-OP-03013; Containment Daily Inspection and Containment Closeout Inspection; Revision 4
DB-SC-03111; SFAS Channel 2 Functional Test; Revision 12
DB-SP-03137; Decay Heat Train 2 Pump and Valve Test; Revision 14
DB-SP-03357; RCS [Reactor Coolant System] Water Inventory Balance; Revision 9; dated February 24 and 25, 2007;

1R23 Temporary Plant Modifications

CR 02-09632; Request for Assistance for Engineering Evaluation of Permanent Platform in CCW [Component Cooling Water] Pump Room
CR 03-06902; Semi-Permanent Scaffold Installation - Request for Assistance
CR 06-9674; Annual Scaffold Inspections Not Performed - Audit MS-C-06-10-07
CR 07-12352; NRC SRI Observation: Unused Scaffolding Installations in Plant
CR 07-16979; DB-MS-01637 Review of Permanent Scaffold (NRC Identified)
CR 07-17807; NRC Senior Resident Inspector Identified: Long-Term Use of Temporary Hoses or Tubing (NRC Identified)
CR 07-17820; NRC Senior Resident Inspector Identified: Ladder Left In Place for Valve WM13 Access (NRC Identified)
DB-MS-01637; Scaffolding Erection and Removal; Revision 10
DB-OP-00016; Temporary Configuration Control; Revision 10
NG-EN-00313; Control of Temporary Modifications; Revision 11
WO Notification 600353527; Engineering Evaluation of Long Term Scaffolding

1EP6 Drill Evaluation

CR 07-17160; EP Drill - Evaluation Of Timeliness GE [General Emergency] Declaration
Davis-Besse Emergency Response Integrated Drill Manual; 2007
RA-EP-01500; Emergency Classification; Revision 06

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

Updated Safety Analysis Report Section 11; Revision 23
Groundwater Flow Characteristic Report Davis-Besse Nuclear Power Station; FENOC; dated January 16, 2007
2005 Annual Radiological Environmental Operating Report; Including the Radiological Effluent Release Report
2004 Annual Radiological Environmental Operating Report, Including the Radiological Effluent Release Report
FENOC Groundwater Protection Initiative Plan; Updated September 6, 2006
FENOC Oversight Quality Field Observation; Effluent Monitoring Program; performed April 18, 2005 - June 10, 2005; Effluent Monitoring Program
FENOC Oversight Quality Field Observation; Effluent Monitoring Program; performed November 29, 2004 - December 17, 2004; ODCM [Offsite Dose Calculation Manual] Revision
FENOC Oversight Quality Field Observation; Weekly Station Vent Releases; performed October 17, 2006 - October 17, 2006; Chemistry
DB-M103425; Channel Functional Test of 69D-1SFI700B; Chemical Waste Outlet 3" Flow; Revision 6
DB-M103435; Channel Functional Test of 20A-ISF2729, Service Water Outlet Flow Collection Box; Revision 7
DB-M103412; Calibration of Channel One and Two for RE 4597AA, RE 4597BA, Normal Range Radiation Monitor; Revision 1
DB-SS-03251; Emergency Ventilation System Train 2 Monthly Test; Revision 4
DB-OP-06503; Containment Purge System Procedure; Revision 14
DB-SS-03042; Control Room Emergency Ventilation System Train 1 Monthly Test; Revision 4
DB-SS-03250; Emergency Ventilation System Train 1 Monthly Test; Revision 4
DB-MI-03444; Channel Calibration of 32C-ISF5090A Station Vent Flow; Revision 5
DB-CH01804; Tritium Determination; Revision 4
DB-OP-03012; Radioactive Gaseous Batch Release; Revision 8
CR 07-13271; A Previous Week Attachment 3 Entry for Sample Start Time had a Date that was Written Over
CR 07-13405; On March 6, 2006, the Station Vent Monitor Detected 2.06E-06 uCi/ml [microcuries per milliliter] of Xenon-133 Gas
CR 06-02133; An Overdue Monthly Check Source Test on DB-SC-03228; a Radiation Monitor to Close Purge Containment Isolation Valves
CR 06-02843; The Storm Sewer Drain Line Radiation Monitor (Re-4686) Alarm Response Instructions in the ODCM are Inadequate
CR 05-04891; Auxiliary Boiler; Tag Out NDB-Sub085-08-027 Issue; Draining Auxiliary Boiler Water into Sewer Drain

4OA1 Performance Indicator (PI) Verification

NOBP-LP-4012; NRC Performance Indicators; Revision 01
NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 04

Initiating Events Cornerstone Indicators Data Input Forms; April 2006 through February 2007
Reactor Coolant System Leakage Data Input Forms; April 2006 through February 2007

4OA2 Identification and Resolution of Problems

Calculation C-EE-006.01-027; Safety-Related Motor Contactor Control Circuit Voltage Drop; Revision 03
CR 02-08884; MU59C - EOC/EQ Inspection
CR 05-00986; Increase In opening stroke Time for MU59C
CR 07-13923; Lost of Power to MU59C
CR 07-14083; Bad Fuse Clip in Breaker BE1177 Cubicle for MU59C
Drawing E-008-00087; Type W Control Center Wiring Diagram and Schematic; Revision 05
MWO 1-97-0442-00; Replace Fuse Block/Channel; Breaker for Clearwell Transfer Pump
NOBP-ER-3010; FENOC Hardware Resolution; Revision 01
NOBP-LP-2010; CREST Trending Codes; Revision 05
NOBP-LP-2011; FENOC Cause Analysis; Revision 06
NOP-LP-2001; Corrective Action Program; Revision 15
WO 01-007063-000; Breaker For CTMT [Containment] Air Sample Return Valve MV5011E
WO 01-009086-000; Breaker for CTMT Emergency Sump Outlet Valve Motor MVDH9A
WO 02-000115-000; Breaker For SG [Steam Generator] 2 To AFPT [Auxiliary Feed Pump Turbine] 1 Isolation Valve Motor MV106A

4OA3 Event Followup

DB-MM-09320; Emergency and Station Blackout Diesel Engine Maintenance; Revision 12
DB-MM-09345; Emergency and Station Blackout Diesel Engine 6 Year Maintenance; Revision 00

4OA5 Other Activities

Davis-Besse Nuclear Power Station Simulator Training Certification Test T16 - Generator/Turbine Trip; dated November 6, 2005
Davis-Besse Nuclear Power Station Simulator Training Certification Test T16 - Generator/Turbine Trip (at less than 40 percent power); dated November 29, 2006
Davis-Besse Nuclear Power Station Simulator Training Certification Test T16 - Generator/Turbine Trip (at 100 percent power); dated November 30, 2006
FENOC Letter dated April 2, 2004; Licensing Basis for Service Water System Discharge Flow Path
FENOC Letter dated April 5, 2004; Reply to a Notice of Violation from Davis-Besse Nuclear Power Station NRC Inspection Report No. 50-346/03-010 (EA-04-049 and EA-04-050)
FENOC Letter dated 5/15/2006: Submittal of the Independent Assessment Plan for the Davis-Besse Nuclear Power Station Corrective Action Program Implementation - Year 2006
FENOC Letter dated 10/23/2006; Submittal of Independent Assessment Report of Corrective Action Program Implementation and Action Plan for the Davis-Besse Nuclear Power Station, Year 2006
CR-02-07802; Basis For PSH 2929 and PSH 2930 Not Found; dated October 10, 2002
CR 03-07656; Forward Flow Rate of 10,000 GPM [gallons per minute] Not Attained for SW 19 During DB-PF-03232; dated September 12, 2003
CR 04-06372; COIA-ENG-2004 Dry Fuel Storage Pad Transient Combustible Control
CR 06-00246; NRC 50.59/MOD - NRC Concern Regarding UCN 03-058; January 27, 2006

CR 06-00472; NRC 50.59/MOD - Methodology Used to Evaluate Tornado Differential Pressure; February 24, 2006
C R 06-01753; Station not in Compliance with DB-FP-00007
CR 06-02340; Potential Violation of 10CFR72.122C
CR 06-6427; NRC Non-Cited Violation - 10CFR72.212; Combustibles Stored Near Dry Fuel Storage
CR 06-06723; COIA-CAP 2006: Equipment Trending Below Industry Standard
CR 06-09223; COIA-CAP 2006: Closing CAQ [Condition Adverse to Quality] CRs to Notifications
CR 06-09228; COIA-CAP 2006: Evaluation of CR 05-00288 was Determined to be Incomplete
CR 06-09229; COIA-CAP 2006: CR Evaluation Weaknesses
CR 06-09230; COIA-CAP 2006: Some Delays in the Internal reviews of Incoming OEs
CR 06-09231; COIA-CAP 2006: Few Resources Directed Towards Backlog Reduction
CR 06-09232; COIA-CAP 2006: Lack of CR Cognitive Binning During 14RFO
CR 06-09233; COIA-CAP 2006: Open SAP Items Were Not Incorporated in Department and Section IPAs
CR 06-09235: COIA-CAP 2006; Identification of Repeat Occurrences are Not Retrievable
CR 06-09236: COIA-CAP 2006; 05-00738 Repeat Occurrence due to Untimely Corrective Action
CR 06-09238: COIA-CAP 2006; CR 05-05395 Additional Training on RCE, EOCC, and CRs Closed Without All Targeted Audience Receiving Training
CR 06-09239; COIA-CAP 2006; The Number of SCAQ [Significant Condition Adverse to Quality] Items Open Over 135 Days
CR-06-10403; NRC 7111.11 Unresolved Issue With Simulator Generator Trip Test; dated November 21, 2006
CR 07-15336; Combustible Material Found in Sealand Containers on Dry Fuel Storage Pad
Procedure DB-NE-03400; Horizontal Storage Module (HSM) Monitoring; Revision 2
Procedure DBBP-RP-1010; Routine Radiological Surveys; Revision 9
Procedure DB-HP-04004; Area and Spiked TLD Checks; Revision 4
2005 Annual Radiological Environmental Operating Report
Surveillance Logs and HSM Radiation Surveys
Task DB-NE4103-00; SNM [Source Nuclear Material] Inventory; dated April 11, 2006
NOP-SS-3300; FirstEnergy Enterprise Records Management Program; Revision 1
DB-NE-00100; Fuel Handling Procedure; Revision 6
DB-NE-06471; Dry Fuel Storage Unloading; Revision 1
10 CFR 72.48 Screen; Temporary Staging Of Reactor Coolant Pump Motors Within the Dry Fuel Storage Facility; dated February 7, 2006
10 CFR 72.48 Screen; Extend Temporary Staging Of Reactor Coolant Pump Motors Within the Dry Fuel Storage Facility; dated July 21, 2006 and September 26, 2006
10 CFR 72.48 Screen; Temporary Storage of Sealands Within the Dry Fuel Storage Facility; dated June 30, 2005
10 CFR 72.48 Screen; Extension of Temporary Sealand Storage on Dry Fuel Storage Facility Pad; dated July 28, 2006
Emergency Plan Implementing Procedure; RA-EP-01500; Emergency Classification; Revision 6
Emergency Plan Off Normal Occurrence Procedure RA-EP-02810; Tornado; Revision 6
Emergency Plan Off Normal Occurrence Procedure RA-EP-02820; Earthquake; Revision 5
Procedure DB-FP-0007; Control of Transient Combustibles; dated June 6, 2006
Procedure DB-OP-02529; Fire Procedure; dated February 2, 2007
Quality Assurance Program Manual; Revision 8; dated April 18, 2006
Nuclear Quality Assurance Program NOPL-LP-2001; Revision 2; dated September 5, 2006
NG-EN-00372; Dry Fuel Storage; dated January 27, 2006
NOP-LP-2001; Corrective Action Program; Revision 15, dated February 21, 2007

Calculation 03-014; Service Water System Performance With Discharge Line Blockage; dated March 27, 2003
Calculation C-ME-099.16-010; Check Valve Design Basis Analysis; Revision 01
Calculation C-NSA-011.01-16; Service Water System Design Basis Flow Rate Analysis and Testing Requirements; Revision R00

LIST OF ACRONYMS USED

ADAMS	Agency-wide Document Access and Management System
AFI	Area For Improvement
ALARA	As-Low-As-Reasonably-Achievable
ANSI	American National Standards Institute
ANSI/ANS	American National Standards Institute/American Nuclear Society
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CR	Condition Report
DH	Decay heat
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DSC	Dry Shielded Canister
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FENOC	FirstEnergy Nuclear Operating Company
FSAR	Final Safety Analysis Report
HEPA	High Efficiency Particulate Air
HSM	Horizontal Storage Module
IE	Initiating Events
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISFSI	Independent Spent Fuel Storage Installation
LER	Licensee Event Report
LLD	Lower Limit of Detection
MS	Mitigating Systems
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PI	Performance Indicator
RETS/ODCM	Radioactive Effluent Technical Specification/Offsite Dose Calculation Manual
RPS	Reactor Protection System
SDP	Significance Determination Process
SRP	Standard Review Plan
TLD	Thermo-Luminescent Dosimetry
TS	Technical Specification
USAR	Updated Safety Analysis Report
URI	Unresolved Item