August 11, 2006

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 05000346/2006003

Dear Mr. Bezilla:

On June 30, 2006, 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 June 22, 2006, with you and other members of your staff. Additionally, this inspection report documents special inspection activities associated with your compliance with the March 8, 2004, Confirmatory Order (EA-03-214).

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

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

If you contest the subject or severity of 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 made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000346/2006003
w/Attachment: Supplemental Information

cc w/encl: The Honorable Dennis Kucinich
G. Leidich, President and Chief
Nuclear Officer - FENOC
J. Hagan, Senior Vice President of
Operations and Chief Operating Officer
Richard Anderson, Vice President
Director, Plant Operations
Manager - Site Regulatory Compliance
D. Pace, Senior Vice President of
of Fleet Engineering
J. Rinckel, Vice President, Fleet Oversight
D. Jenkins, Attorney, FirstEnergy
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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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2006003
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	April 1, 2006, through June 30, 2006
Inspectors:	J. Rutkowski, Senior Resident Inspector R. Smith, Resident Inspector G. Wright, Project Engineer D. Passehl, Senior Reactor Analyst R. Daley, Senior Reactor Inspector J. House, Senior Radiation Specialist M. Phalen, Radiation Specialist
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000346/2006003; 4/1/2006 - 6/30/2006; Davis-Besse Nuclear Power Station; Fire Protection, Surveillance Testing, Access Control to Radiologically Significant Areas, and Identification and Resolution of Problems.

This report covers a 3-month period of baseline inspection. The inspection was conducted by Region III inspectors and resident inspectors. The inspection identified five findings of very low safety significance, four of which involved Non-Cited Violations (NCVs) of NRC requirements. 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was self-revealed when, with the plant shut down for a planned refueling outage, an uncontrolled 10 degree Fahrenheit heatup of the reactor coolant system occurred over a period of approximately 1 hour. The licensee had remotely closed a degraded air-operated valve to isolate cooling water flow to the in-service decay heat cooler to control plant heatup. However, because the valve was degraded, it could not be remotely opened to control the heatup. The onshift operating crew was unaware that the valve had been identified as degraded and inoperable by a previous operating crew. The licensee restored cooling by manually opening the valve and generated a condition report to enter the issue into its corrective action program. No violation of regulatory requirements occurred.

  The finding was more than minor because it was associated with the configuration control attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance because: 1) the finding did not increase the likelihood of a loss of reactor coolant system inventory; 2) the finding did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory when needed; and 3) the finding did not significantly degrade the licensee's ability to recover decay heat removal once it was lost. The primary cause of this finding was related to the cross-cutting area of Human Performance. (Section 1R22)

- Green. A finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified by the inspectors when the licensee failed to investigate and correct an abnormal noise associated with emergency diesel generator 2 (EDG2) in a timely manner. As a result, EDG2 was inoperable in excess of the time limits prescribed by the associated Technical Specification (TS) Limiting Condition for Operation (LCO). The licensee repaired EDG2 and generated a condition report to enter this issue into its corrective action program.

The finding was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance because the change in core damage frequency calculated through an SDP Phase 3 analysis was less than 1E-6. The primary cause of the finding was related to the cross-cutting area of Human Performance. (Section 4OA2.3)

C Green. The inspectors identified a finding of very low safety significance and a Severity Level IV Non-Cited Violation when the licensee failed to adhere to the requirements of TS Section 6.17, “Technical Specifications Bases Control Program.” By adding TS Bases wording, Surveillance Requirement 4.8.1.1.1.b was reinterpreted to include testing of the fast transfer for the 13.8 kV bus tie transformers. This allowed the licensee to credit the fast transfer capability for offsite power operability without submitting a license amendment for NRC review.

Because the issue affected the NRC’s ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the TS Bases change would have required a license amendment and because the finding affected equipment important to safety. The finding was determined to be of very low safety significance because: 1) it did not represent an actual loss of safety function of a system; 2) it did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; 3) it did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours; and 4) it did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event. (Section 4OA5.3)

Cornerstone: Occupational Radiation Safety

- Green. A finding of very low safety significance and associated Non-Cited Violation of TS 6.12.1.b was self-revealed when, in three separate instances, radiation workers entered posted high radiation areas (HRAs) without appropriate authorization. Specifically, the radiation workers entered posted HRAs although in each instance the radiation work permit used to access the radiologically restricted area did not permit access to HRAs.

  The finding was more than minor because it was associated with the human performance attribute of the occupational radiation safety cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. The finding was of very low safety significance because it did not involve: 1) As-Low-As-Reasonably-Achievable planning or controls; 2) an overexposure; 3) a substantial potential for an overexposure; or 4) an impaired ability to assess dose. The primary cause of this finding was related to the cross-cutting area of Human Performance. (Section 2OS1.1)

Cornerstone: Other

C Green. A finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 72.212, “Conditions of general license issued under §72.210,” was identified by the inspectors when the licensee failed to adequately control transient combustible material near the dry spent fuel horizontal storage modules (HSMs) in accordance with procedures. The proper control of transient combustible material was required by fire protection procedures that prescribed compliance with conditions specified in the NRC-issued Certificate of Compliance for the HSMs. As part of their immediate corrective actions, licensee personnel removed the transient combustible material from the area surrounding the HSMs and generated a condition report to enter this issue in their corrective action program.

This finding was more than minor because it was associated with protection against potential fire damage to the HSMs, which if left uncorrected could lead to a more significant safety concern. The inspectors determined that the finding was not suitable for SDP evaluation because the issue did not involve permanently installed plant equipment. Therefore, the finding was reviewed by Regional Management in accordance with Inspection Manual Chapter 0612, Section 05.04c, and determined to be of very low safety significance. The plant fire brigade could have been dispatched to extinguish a fire before significant damage to the HSMs occurred. The primary cause of the finding was related to the cross-cutting area of Human Performance. (Section 4OA5.2)

B. Licensee-Identified Findings

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 shut down conducting activities in its 14th refueling outage (RFO 14). The licensee commenced restart activities on April 22, 2006, with the reactor being made critical on April 25, 2006. The main generator was placed online and connected to the electrical grid on April 27, 2006. Full power was achieved on May 5, 2006.

On June 23, 2006, the licensee commenced a power reduction to 15 percent for replacement of the master trip solenoid valve on the main turbine. The main turbine was tripped in the early morning of June 24, 2006. The unit remained in Mode 1. The replacement of the valve was completed and the main turbine was placed online and connected to the electrical grid in the early evening of June 24, 2006. Full power was achieved on June 25, 2006.

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 the licensee's restoration of systems from cold weather preparations and the licensee's preparations for hot weather operations with particular emphasis on the readiness of the emergency diesel generator systems for hot weather. This included a review of the requirements and work orders for changing to higher heat resistant oil in the diesel generator air intake filters. Additionally, the inspectors reviewed the licensee's procedural requirements and conducted walkdowns to determine whether ventilation systems and other equipment were properly realigned for hot weather. The inspectors also interviewed operations personnel on their progress towards completion of hot weather preparations.

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

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

a. Inspection Scope

The inspectors conducted partial walkdowns of the systems and trains listed below to determine whether the systems were correctly aligned to perform their designed safety function. The inspectors used licensee system valve line-up documents and system drawings during the walkdowns. The walkdowns included selected switch and valve position checks, and verification of electrical power to critical components. Finally, the inspectors evaluated other elements of system readiness, such as material condition, housekeeping, and component labeling. The documents used for the walkdowns are listed in the Attachment. The inspectors reviewed:

- high pressure injection system train 2 during a planned maintenance outage that rendered train 1 inoperable and unavailable on May 16, 2006;
- emergency diesel generator 1 and associated equipment during a planned outage for work that rendered emergency diesel generator 2 inoperable on June 1, 2006; and
- decay heat and low pressure injection system train 2 during a planned maintenance outage that rendered train 1 equipment inoperable and unavailable on June 13, 2006.

This constitutes three quarterly samples.

b. <u>Findings</u>

No findings of significance were identified.

1R05 <u>Fire Protection</u> (71111.05Q)

a. <u>Inspection Scope</u>

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

- auxiliary feedwater pump 1 room (Fire Area E, Room 237);
- auxiliary building, 585' elevation, corridor (Fire Area V, Rooms 304);
- spent fuel pool pump room and adjoining hatch area (Fire Area U, Room 312 and Room 313);
- reactor coolant pump oil spill collection system including the areas around the reactor coolant pumps;
- boric acid addition tank room (Fire Area G, Room 240);
- service water pump room (Fire Area BF, Room 52); and
- dry spent fuel storage pad.

This constitutes seven quarterly samples.

b. Findings

One specific issue, which was identified as an Non-Cited Violation (NCV), was noted during a routine fire protection inspection of the dry fuel storage pad. This violation is discussed further in Section 4OA5 of this report.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors evaluated the service water pump room for internal flooding hazards. As part of this inspection, the inspectors evaluated the impact of below grade wall penetrations, fire suppression capability, drain and sump locations, and sump pump capability and determined whether the room configuration and equipment capability was accurately depicted in design basis documents and risk assessments. Additionally, the inspectors determined whether the licensee had procedures in place to address flooding from the service water tunnel and if compensatory measures were established during maintenance activities which could increase the potential for internal flooding. The inspectors walked down the intake service structure and the service water pump room to determine whether the licensee had identified all reasonable sources that could flood the rooms and that sufficient plugs were available to address procedural requirements for plugging service water pump room drains in the event of flooding of the service water pipe tunnel. Also, during the licensee's refueling outage, while work was being accomplished in the service water pump room, the inspectors reviewed the licensee's precautions to monitor for and address potential flooding situations.

This constitutes one sample of internal flooding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

On May 23, 2006, the inspectors observed an operating crew during a crew simulator quarterly evaluation and attended the post-session licensee controller 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 reactor coolant system small break with a makeup pump failure and subsequent high turbine vibration.

This constitutes one quarterly sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's response to performance issues associated with the main turbine with emphasis on issues associated with the master trip solenoid valve. The inspectors also reviewed licensee's activities associated with replacing the master trip solenoid valve.

The reviews consisted of evaluating the following activities:

- use of the condition report process in identifying deficiencies and issues with system equipment;
- whether equipment performance issues were correctly categorized per the system's scoping sheet performance criteria for reliability;
- whether the licensee effectively tracked key parameters and identified system trends and monitored for indications of component failures;
- appropriateness of goals and corrective actions associated with long-term reliability;
- whether the physical condition of the system appeared consistent with status as reflected in condition reports and open work orders;
- whether the licensee's corrective actions included extent of condition; and
- appropriateness of maintenance rule system status classification with an emphasis on whether current re-classification appeared appropriate for the equipment's recent history.

This constitutes one quarterly sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the licensee's response to risk-significant activities. These activities were chosen based on their potential impact on overall plant risk. The inspections were conducted to determine whether the planning, control, and performance of the work was done in a manner to minimize overall plant risk and if contingency plans were in place where appropriate. The licensee's daily configuration risk assessments, shift turnover meeting observations, daily plant status meeting observations, and the documents listed at the end of this report were used by the inspectors to determine if the equipment configurations had been properly listed, protected equipment had been identified and was being controlled where appropriate, that significant aspects of plant risk were being communicated to the necessary personnel, and existing work plans were adequately revised to accommodate changes in planned equipment operability. The inspectors evaluated the following activities:

- On April 10, 2006, the licensee drained the reactor coolant system to 54 inches above the reactor vessel hot leg nozzle centerline to accomplish planned refurbishment of reactor coolant pump seals. This evolution was classified as an orange risk evolution.

- On April 23, 2006, the licensee determined that there was leakage past the seat of the pressurizer pilot operated relief valve (PORV) into the quench tank. Cycling of the PORV block valve did not cause the PORV to reseat. The block valve, for a period of time, was maintained in a closed position and was reviewed by the licensee for impact on risk.

- On April 24, 2006, the licensee had an unexpected loss of the 'K' Bus (part of the plant's ring bus) which resulted in a loss of startup transformer 2. The loss of the bus was due to an offsite fault on the Ohio Edison transmission line. The plant was in Mode 3 at the time.

- On May 3 through 5, 2006, while ascending in power from RFO 14 and at about 93 percent power, the licensee suspended power ascension to address numerous high pressure feedwater heater alarms and low level alarms in their deaerator storage tanks that supply the normal feedwater pumps. The loss of the feedwater pumps could cause a plant trip.

- On May 24 through 26, 2006, licensee personnel removed switchyard air circuit breaker 34561 from service to perform 10-year preventive maintenance and testing. This placed the plant in a 72-hour limiting condition for operation and a yellow risk condition.

- On June 5, 2006, the licensee experienced a perturbation in their main generator seal oil system that caused the emergency seal oil pump to start. The event required repairs to the main pump and resulted in an elevated risk condition.

- On June 27 and 28, 2006, licensee personnel conducted excavation activities near the main transformer to locate a large water leak in a firewater spray feeder pipe. The licensee evaluated this activity as contributing to the risk of a plant trip.

This constitutes seven samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors reviewed licensee logs and procedures, plant computer data as appropriate, and licensee performance, to determine if the non-routine actions and responses were in accordance with standards and procedures. The inspectors reviewed the following non-routine events:

- On April 4 through April 10, 2006, the inspectors observed portions of the reassembly of the reactor vessel. This included core plenum movement from the deep end of the refueling canal to the reactor vessel, reactor vessel head placement on the reactor vessel, and tensioning of the reactor vessel head studs. The inspectors also observed selected As-Low-As-Reasonably-Achievable (ALARA) and work planning briefs; rigging methods and load paths established for movement of heavy loads by the containment polar crane; and foreign material exclusion processes associated with reactor vessel reassembly.

- On April 22 and 23, 2006, the inspectors observed portions of the plant heatup to Mode 4 and then to Mode 3. This included drawing a steam bubble in the pressurizer, startup of the reactor coolant pumps, heatup, and maneuvering to maintain the plant within pressure-temperature limits. The inspectors also observed pre-job briefs for various evolutions during the heatup.

- On April 25, 2006, the inspectors observed the approach to criticality and zero power physics testing. This included dilution of the reactor coolant system to the estimated critical boron concentration, withdrawing control rods to criticality, and zero power physics testing to determine acceptability for power escalation for the beginning of the new fuel cycle. The inspectors also observed the pre-job brief for the approach to criticality.

- On April 26 and 27, 2006, the inspectors observed several main turbine rolls and several manually initiated turbine trips due to high vibration associated with the new turbine conditions caused by the low pressure turbine mono-block rotor installation.

This constitutes four samples.

b. Findings

One specific issue that involved a licensee-identified violation was identified during the inspectors' review of the non-routine evolutions and events inspection samples. This licensee-identified violation is further discussed in Section 4OA7 of this report.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected condition reports which identified potential operability issues associated with risk-significant components or systems. These condition reports and applicable licensee operability evaluations were reviewed to determine if the operability of the components or systems were appropriately supported. The inspectors compared the operability and design criteria in the appropriate sections of the Updated Safety Analysis Report to the licensee's evaluation of the issues to determine if the components or systems were operable. Where compensatory measures were necessary to maintain operability, the inspectors determined whether compensatory measures were in place, would work as intended, and were properly controlled.

The following samples were evaluated:

- CR 06-01243, CR 06-01339, CR 06-01342, and CR 06-01400, which addressed foreign material in and around the reactor coolant system;
- CR 06-00779, CR 06-01035, and 06-01471, which addressed the as found and as left conditions of the cooling water side of the containment air coolers;
- condition report mode restraints for entering Modes 6, 5, 4, 3, 2 and 1; and
- CR 06-02382, which addressed a stroke time outside of procedure guidelines for a decay heat cooler outlet valve.

This constitutes four samples.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- makeup pump 2 baseline test on April 14, 2006, following motor replacement;
- auxiliary feedwater pumps 1 and 2 high speed stop and overspeed trip tests on April 17 and 18, 2006;
- containment spray pump 2 quarterly test following mechanical seal replacement from April 16 through 20, 2006;

- control rod assembly insertion time test for all control rods on April 24, 2006, following reassembly of the reactor vessel head during the refueling outage;
- normal operating pressure and temperature leak tests on April 23 and 24, 2006, of various reactor coolant system components and the reactor vessel following component reassembly and restoration during the refueling outage;
- local leak rate testing of valve CV5007 (containment purge exhaust damper inside containment) following the adjustment of seat rings on April 14 through April 16, 2006;
- main turbine overspeed testing on April 27, 2006, following complete replacement of both low pressure turbine rotors during the refueling outage; and
- emergency diesel generator 2 excitation system acceptance test on April 4, 2006, following installation of an upgraded excitation system.

The inspectors determined 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 were 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 constitutes eight samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed activities associated with RFO 14 which began on March 6, 2006. The inspectors reviewed the reactor coolant system (RCS) heatup, configuration management, clearance activities, reduced RCS inventory operations, shutdown risk management, conformance to applicable procedures, and compliance with TS. Select portions of the following major activities were also observed:

- reactor vessel reassembly;
- mode restraint closure;
- spent fuel pool cooling operation;
- reactor coolant system heatup and the transition to placing the RCS loops in service and removing the decay heat system from service;
- containment closure inspection activities including inspection of the containment emergency recirculation sump and the area under the reactor vessel;
- operations with the reactor coolant system in reduced inventory conditions, up to and including deep drain operations (reactor vessel water level below vessel flange level to 54 inches above the hot leg nozzle centerline) for reactor coolant pump seal work;
- reassembly of reactor coolant pump 2-1 and 2-2 with refurbished motors and pumps;

- initial criticality and low power physics testing; and
- control rod drop testing.

Because RFO 14 spanned two inspection periods, the activities noted above were a continuation of the sample that were opened in the previous inspection period (IR 05000346/2006002) and did not constitute a new inspection sample.

b. Findings

No findings of significance were identified from the above activities other than those findings reported elsewhere in this report.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed the surveillance tests listed below and/or evaluated surveillance test data to determine whether equipment tested met TS, USAR, and licensee procedural requirements, and 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; whether the test was conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; whether the test acceptance criteria were met; and whether the results of the test were properly reviewed and documented. The following surveillances were evaluated:

- DB-OP-03013 attachment 1, 2, and 3 and DB-SP-03134, containment closeout on April 16, 2006, of containment lower levels and emergency sump inspection performed prior to entry into Mode 4;
- DB-SC-04052, 4160 volt buses D1 and D2 transfer and lockout test during the period of April 3 through April 16, 2006;
- DP-SP-03001, service water loop 2 integrated flow balance test during the period of April 16 through April 19, 2006;
- DB-SP-03136, decay heat train 1 boron precipitation check valve forward flow test and quarterly pump test on April 19, 2006;
- DB-OP-03013, containment closeout of the containment upper levels, including the 653' elevation, east and west 'D' Rings and the incore tank on April 20, 2006;
- DB-SC-03114, safety features actuation system integrated response test on April 21, 2006;
- DB-SS-04151, main turbine control valve testing on May 4, 2006;
- DB-PF-03072, component cooling water pump 1 quarterly testing on May 9, 2006; and
- DB-PF-06703, reactor coolant system water inventory balance on May 18, 2006.

This constitutes nine samples of which two were quarterly inservice testing (IST) samples and one was an RCS leakage inspection sample.

b. Findings

Introduction: A finding of very low safety significance was self-revealed when, with the plant shutdown for a planned refueling outage, an uncontrolled 10 degree Fahrenheit (EF) heatup of the reactor coolant system occurred over a period of about 1 hour. Licensee personnel had remotely closed a degraded air-operated valve to isolate cooling water flow to the in-service decay heat cooler to control plant cooldown during a surveillance test. However, because the valve was degraded, it could not be remotely opened to control plant heatup. The valve had been identified as degraded and inoperable; however, the onshift crew was not aware of this condition. No violation of regulatory requirements occurred.

Description: On April 19, 2006, with the plant in Mode 5 and in day 44 of RFO 14, and with the reactor coolant loops filled and vented, licensee personnel commenced testing of decay heat system train 1 as prescribed by procedure DP-SP-03136, "Decay Heat Train 1 Pump and Valve Test." At the time, decay heat train 1 was in-service providing reactor core cooling. A section of the procedure that was required to be performed verified flow through check valve DH207 in a boron precipitation control flow line.

The verification of flow through DH207, as required by DP-SP-03136, prescribed that the decay heat cooler bypass valve be closed to obtain a system flow rate of about 3100 gallons per minute (gpm). With the bypass valve closed and in an abnormal alignment, all flow would be directed through the decay heat cooler. The onshift operations crew discussed this alignment and realized that with the small amount of decay heat present, the reactor could be inadvertently cooled down outside of operating temperature limits. To address this issue, operations personnel established a contingency plan to remotely close CC1467, the decay heat cooler 1 component cooling water outlet valve. A previous operations crew had declared valve CC1467 inoperable due to reliability concerns associated with the valve opening. While the previous crew had generated a condition report to document the concern, they did not document the position in the unit's operating log or place tags on the valve hand switch. While log entries and tags were required to be reviewed during shift turnovers, condition reports were not, therefore the onshift crew was not aware of the concern with the valve. Shift personnel also failed to recognize that literal compliance with the component cooling water system operating procedure would have precluded shutting CC1467 while the decay heat cooler was aligned for core cooling.

During the portion of the test that verified flow through DH207, reactor coolant system temperature was observed to be decreasing and the operating crew remotely closed valve CC1467. Following completion of the test, attempts to open CC1467 remotely were unsuccessful. The valve was opened manually about 1 hour later. During this time, with no cooling water flow to decay heat cooler 1, reactor coolant water temperature at the decay heat cooler outlet increased from about 88 EF to about 99 EF.

Subsequent licensee investigation (CR 06-01963) determined that on March 20, 2006, CC1467 had failed to open as required on several attempts and the valve had been declared inoperable (CR 06-01114). Additionally, licensee personnel determined that previous testing of valve DH207 had been performed with decay heat train 1 not in-service to provide core decay heat removal. The licensee also determined that the DH207 flow test had originally been scheduled after decay heat cooling had been transferred from decay heat train 1 to decay heat train 2. Licensee personnel identified that revisions to the outage schedule had removed restraints that would have prevented the DH207 flow test from being performed until after decay train 2 was placed in service.

Analysis: The inspectors determined that the closure of a degraded and inoperable valve during a surveillance test, which isolated cooling water flow to the in-service decay heat cooler and an uncontrolled 10 EF heatup of the reactor coolant system, was a performance deficiency warranting a significance evaluation. The finding was more than minor because it was associated with the configuration control attribute of the mitigating systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. The primary cause of this finding was related to the cross-cutting area of Human Performance because licensee personnel failed to adequately communicate the inoperable and degraded condition of a valve used to establish a shutdown cooling flowpath.

The inspectors determined that the finding was of very low safety significance using Checklist 4, “PWR Refueling Operation: RCS Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 Hours and Inventory in the Pressurizer,” of Inspection Manual Chapter (IMC) 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process - Phase 1 Operational Checklist for Both PWRs and BWRs,” dated May 25, 2004. The inspectors determined that the issue was of very low safety significance, because: 1) the finding did not increase the likelihood of a loss of RCS inventory, 2) the finding did not degrade the licensee’s ability to terminate a leak path or add RCS inventory when needed, and 3) the finding did not significantly degrade the licensee’s ability to recover decay heat removal once it was lost.

Enforcement: No violation of regulatory requirements occurred. Licensee personnel performed a test as prescribed by an approved procedure and adhered to all TS requirements. The issue was considered a finding of very low safety significance (FIN 05000346/2006003-01). The licensee entered the event into their corrective action program as CR 06-01963, “Decay Heat System Temperature Increase During DH207 Check Valve Flow Testing.”

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary modification 06-0016, “Large Temporary Enclosure with Leak Sealing for Valve Reheat Drain 13 Steam Leak,” and the associated 10 CFR 50.59 screening against system requirements, including the USAR, to determine whether there were any detrimental effects on system operability, availability, or transient initiator probability and if consistency with plant documentation and procedures was maintained. The inspectors periodically observed the effectiveness of the temporary modification in eliminating a steam plume. The inspectors sampled the work activities involved in the physical implementation of the temporary modification. Additionally, the inspectors reviewed the work order governing the work.

This constitutes one 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 June 22, 2006. The observations included licensee preparations, evaluation of drill conduct, review of the drill critiques, 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 statistics. The inspectors observed drill activities and personnel performance in the simulator control room and included the communications from the control room to the support facilities. The inspectors evaluated the effectiveness of the licensee's communications, the accuracy of situation evaluations, and the timeliness of required reporting (simulated) of event-related information to the appropriate agencies. Finally, the inspectors reviewed the licensee's control room drill critique to determine if weaknesses and deficiencies were acknowledged and if appropriate corrective actions were identified.

This constitutes one sample.

b. Findings:

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Problem Identification and Resolution

a. Inspection Scope

Corrective action reports related to access controls and high radiation area (HRA) radiological incidents (non-performance indicator occurrences identified by the licensee in HRAs less than 1 Rem/hour) were reviewed. Staff members were interviewed and corrective action documents were reviewed to determine if follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

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

This sample was credited in Inspection Report (IR) 05000346/2006002.

b. <u>Findings</u>

.1 <u>Introduction</u>: A finding of very low safety significance and as associated NCV of TS 6.12.1.b was self-revealed when, in three separate instances, radiation workers entered posted HRAs without appropriate authorization. Specifically, these radiation workers entered posted HRAs although in each instance the radiation work permit (RWP) used to access the radiologically restricted area did not permit access into an HRA.

Example 1:

<u>Description</u>: On April 3, 2006, a contractor radiation worker (radworker) received a pre-job briefing and was directed to report to the 653' elevation of containment for paint stripping of structural steel staged on top of the D-ring. The radiation protection aspects of the work were controlled by RWP 2006-5001, which prescribed dose and dose rate alarm setpoints of 10 millirem and 75 millirem respectively. Radiation Work Permit 2006-5001 did not authorize entry into HRAs. During the work, the radworker's supervisor identified an additional location that required paint stripping. The radworker accompanied the supervisor to the 565' elevation inside the D-ring which was a posted HRA. Shortly after entering the HRA, the radworker's electronic dosimeter (ED) alarmed. Following the procedure, the individual left the area and reported to radiation protection (RP) personnel, who determined the individual had received 11 millirem. Prior to entering the HRA (inside the D-ring), the radworker failed to ensure that he was on the appropriate RWP that permitted entry into an HRA and failed to receive the required RP briefing for entry into an HRA.

Example 2:

<u>Description</u>: On March 30, 2006, a contractor radworker was in containment providing oversight for reactor stud movement. The radiation protection aspects of the work were controlled by RWP 2006-5001, which prescribed dose and dose rate alarm setpoints of 10 millirem and 75 millirem respectively. Radiation Work Permit 2006-5001 did not authorize entry into HRAs. During the work activity, the radworker was requested to assist with the movement of the foreign material exclusion cover from the reactor vessel.

The additional activity required entry onto the catwalk around the refueling canal area which was a posted HRA. While on the catwalk, the worker received an ED dose alarm. The worker exited the area and reported the alarm to RP department personnel. Radiation protection personnel determined that the radworker received a 12 millirem dose. Prior to entering the HRA (the catwalk), the radworker failed to ensure that he was authorized to enter an HRA and failed to receive the required RP briefing for entry into an HRA.

Example 3:

<u>Description</u>: On April 6, 2006, an operations employee entered an HRA located in the auxiliary building train bay to disconnect decant hoses from a high integrity container (HIC) containing spent resin. The job entailed transferring spent resin from the spent resin storage tank to a HIC in preparation for shipment. The radiation protection aspects of the work were controlled by RWP 2006-6001, which prescribed dose and dose rate alarm setpoints of 10 millirem and 75 millirem respectively. Radiation Work Permit 2006-6001 did not authorize entry into HRAs. Following entry into the HRA, the worker received a dose rate alarm of 80 millirem per hour. The individual left the area and reported the alarm to RP personnel who determined the individual had received a dose of 2 millirem. When radiation workers initially connected the decant hoses to the HIC for the resin transfer, the area did not qualify as an HRA. The worker had been briefed by a coworker to disconnect the hoses; however, the area had been posted as an HRA while the workers were not in the area. Prior to entering the HRA (on top of the HIC), the radworker failed to ensure that he was authorized to enter an HRA and failed to receive the required RP briefing for entry into an HRA.

As part of the licensee's immediate corrective actions for all three cases, the radworkers' access to the radiologically restricted area was suspended until the individuals were counseled.

<u>Analysis</u>: In each of the three examples described above, the inspectors determined that the failure of a radworker to adhere to radworker practices described in licensee radworker training such as FENOC Radiation Worker Training, Revision 2, and Plant Access Training, Revision 2, was a performance deficiency that warranted a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612 "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 30, 2005, because radworkers failed to adhere to RWP requirement and therefore the finding was associated with the human performance attribute of the occupational radiation safety cornerstone, and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. The primary cause of this finding was related to the cross-cutting area of Human Performance since in each example, a radworker failed to perform adequate self-checking, which resulted in the failure to follow licensee procedures.

Since the finding involved radiological access control issues and the unauthorized entry into HRAs, the inspectors utilized IMC 0609, Appendix C, "Occupational Radiation Safety SDP" to assess its significance. The finding was determined to be of very low safety significance because: 1) it did not involve ALARA planning or controls; 2) it did not involve an overexposure or a substantial potential for an overexposure since the highest dose received was only 12 millirem; or 3) it did not involve an impaired ability to assess dose. Consequently, the inspectors concluded that the finding was of very low safety significance.

Enforcement: Technical Specification 6.12.1.b required, in part, that access to a high radiation area be controlled through the issuance of an appropriate RWP. Contrary to this requirement, on April 3, March 30, and April 6, 2006, contractor radiation workers entered posted HRAs without appropriate RWPs authorizing HRA entry. As part of their immediate corrective actions, the licensee removed each individual's access to the radiologically restricted area and the individuals were counseled. However, because of the very low safety significance of the issue and because the issue was entered into the licensee's corrective action program (CRs 06-01565, 01458 and 01790), the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2006003-02)

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring Program (REMP) And Radioactive Material Control Program (71122.03)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the most current Annual Environmental Monitoring Reports (2004 and 2005) and licensee assessment results, to determine if the Radiological Environmental Monitoring Program (REMP) was implemented as required by the Radiological Environmental Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM). The inspectors reviewed the reports for changes to the ODCM with respect to environmental monitoring and commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, interlaboratory comparison program, and data analysis.

The inspectors reviewed the ODCM and the Annual Reports for 2004 and 2005 to identify environmental monitoring stations and their locations and evaluated licensee self-assessments, audits, and the licensee's vendor laboratory interlaboratory comparison program results. The inspectors reviewed the Updated Final Safety Analysis Report for information regarding the environmental monitoring program and meteorological monitoring instrumentation. The inspectors also reviewed the scope of the licensee's audit program to determine if it met the requirements of 10 CFR 20.1101c.

This review constituted one sample.

b. Findings

No findings of significance were identified.

.2 Onsite Inspection

a. Inspection Scope

The inspectors walked down six of the air sampling stations (greater than 30 percent) and approximately 20 percent of the thermoluminescent dosimeter monitoring stations to determine whether they were located as described in the ODCM and to determine the equipment material condition. This review constituted one sample.

The inspectors observed the collection and preparation of a variety of environmental samples including surface water and air. The environmental sampling program was evaluated to determine if it provided data that was representative of the release pathways as specified in the ODCM and that sampling techniques were performed in accordance with station procedures. This review constituted one sample.

From direct observations and record reviews, the inspectors determined if the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the annual report, NRC Safety Guide 23, and licensee procedures. The inspectors determined if the meteorological data readout and recording instruments, including computer interfaces and data loggers at the tower, were operable; that readouts of wind speed, wind direction, delta temperature, and atmospheric stability measurements were available on the licensee’s computer system, which was available in the Control Room; and that the system was operable. This review represented one sample.

The inspectors reviewed each event documented in the Annual Environmental Monitoring Report which involved missed samples, inoperable samplers, lost thermoluminescent dosimeters, or anomalous measurements for the cause and corrective actions. The Annual Reports were reviewed for positive sample results (i.e., licensed radioactive material detected above the lower limits of detection) and the licensee’s evaluation of the source of this material. This review constituted one sample.

The inspectors reviewed the ODCM for significant changes resulting from modifications to the land use census or sampling station changes made since the last inspection. This included a review of technical justifications for changed sampling locations. The inspectors determined if the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment. This review represented one sample.

The inspectors reviewed the calibration and maintenance records for five air samplers. There were no calibrations for composite water samplers. The inspectors reviewed calibration records for radiation measurement (counting room) instrumentation that could be used for environmental sample analysis and was used for the free release of liquids or pourable solids from the radiologically restricted area. This included determining if the appropriate detection sensitivities would be achieved for counting samples, in that the instrumentation could achieve the RETS/ODCM required environmental lower levels of detection limits. The inspectors reviewed quality control data used to monitor radiation measurement instrument performance, and actions that would be taken if indications of degrading detector performance were observed.

The licensee does not perform radio-chemical analyses of REMP samples. The inspectors reviewed a licensee audit of the vendor laboratory that analyzed these samples. Corrective actions for deficiencies identified in the audit were evaluated along with the vendor's interlaboratory comparison program to determine if the vendor's analytical and quality assurance programs were adequate.

The inspectors reviewed quality assurance audit results of the program to determine whether the licensee met the TS/ODCM requirements. This review constituted one sample.

b. Findings

No findings of significance were identified.

.3 Unrestricted Release of Material From the Radiologically Restricted Area

a. Inspection Scope

The inspectors observed the access control location where the licensee monitored potentially contaminated material leaving the radiologically restricted area and inspected the methods used for the control, survey, and release of material from this area. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use to determine if the work was performed in accordance with plant procedures. This review represented one sample.

The inspectors determined if the radiation monitoring instrumentation was appropriate for the radiation types present and was calibrated with appropriate radiation sources that represented the expected isotopic mix. The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material and determined if there was guidance on how to respond to an alarm indicating the presence of licensed radioactive material. The inspectors reviewed the licensee's equipment to determine if radiation detection sensitivities were consistent with the NRC guidance contained in IE Circular 81-07 and IE Information Notice 85-92 for surface contamination, and HPPOS-221 for volumetrically contaminated material. The inspectors determined if the licensee performed radiation surveys to detect radionuclides that decay via electron capture.

The inspectors reviewed the licensee's procedures and records to determine if the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters such as counting times and background radiation levels. The inspectors determined whether the licensee had established a "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area. 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, and Special Reports related to the REMP since the last inspection to determine if identified problems were entered into the corrective action program for resolution. The inspectors also determined if the licensee's self-assessment program was capable of identifying and addressing repetitive deficiencies or significant individual deficiencies that were identified by the problem identification and resolution process.

The inspectors also reviewed corrective action reports related to the REMP that affected environmental sampling and analysis, and meteorological monitoring instrumentation. Staff members were interviewed and documents were reviewed 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 NCVs tracked in the corrective action system; and
- implementation/consideration of risk significant operational experience feedback.

This review constituted one 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 last quarter 2004 through the first quarter 2006. 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 constitutes one sample of the PI listed above.

Cornerstone: Mitigating Systems

The inspectors sampled licensee submittals for the PI listed below for the period from the last quarter 2004 through the first quarter 2006. 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.

- Safety System Functional Failures

The inspectors reviewed portions of the operating logs, LERs, and maintenance rule records. The inspectors discussed the methods of compiling and reporting the PI with cognizant licensing and maintenance rule personnel.

This constitutes one sample of the PI listed above.

Cornerstone: Barrier Integrity

The inspectors sampled licensee submittals for the two PIs listed below for the period from the second quarter 2005 through the first quarter of 2006. 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 Activity, and
- Reactor Coolant System Leakage

The inspectors reviewed portions of licensee daily reports to find historical data related to reactor coolant system leakage and reactor coolant system activity and reviewed if the values reported for the PIs were consistent with the daily reported data.

This constitutes two samples of the PIs listed above.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment deficiencies or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program (CAP). This screening was accomplished by reviewing documents entered into the CAP and review of document packages prepared for the licensee's daily Management Alignment and Ownership Meetings.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily CAP item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors' review included the 6-month period of December 2005 through May 2006; the Davis-Besse Oversight Assessment Reports (fourth quarter 2005 and first quarter 2006); and issues documented in the licensee's system health reports, maintenance rule monthly minutes for 2006, and the operations integrated performance assessment (November 2005 through April 2006). The inspectors compared and contrasted the results of the licensee's condition report monitoring for RFO 14 with the results contained in the latest Davis-Besse oversight organization's effort to bin and trend refueling outage condition reports. Corrective actions associated with a sample of the issues identified in the licensee's reports were reviewed for adequacy.

This constitutes one semi-annual trend review sample.

b. Assessment and Observations

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

.3 Emergency Diesel Generator Engine Damaged Due to Improper Torquing of Lock Nut

a. Inspection Scope

The inspector reviewed licensee actions to address significant damage to the left valve bridge of cylinder 4 and a missing left rocker arm lash adjustment screw locknut associated with emergency diesel generator 2 (EDG2) that was identified prior to routine surveillance testing as discussed in Unresolved Item (URI) 05000346/2006002-03.

This does not constitute a new sample.

b. Findings

Introduction: A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" was identified by the inspectors when licensee personnel failed to promptly investigate and correct an abnormal tapping noise identified in EDG2 after a scheduled maintenance overhaul.

Discussion: From January 8 through January 14, 2006, EDG2 was inoperable for preventive maintenance. During the post-maintenance test on January 13, 2006, a tapping noise was heard, which was believed to originate in the vicinity of cylinder 4. No additional investigation of the condition was conducted at the time. Although a work order notification was generated to investigate the condition, the post-maintenance test was considered satisfactory, and EDG2 was declared operable on January 14, 2006.

On January 20, 2006, in preparation for a routine EDG2 surveillance test, licensee personnel investigated the previous issue and identified significant damage to the left valve bridge of cylinder 4 and a missing left rocker arm lash adjustment screw locknut. EDG2 was declared inoperable, repaired, and following successful post-maintenance testing, was declared operable on January 23, 2006. Analysis concluded that, based on the conditions found on January 20, 2006, EDG2 had been inoperable from January 13 until January 23, 2006.

Analysis: The inspectors determined that failure to ensure that the lash adjustment lock nut on cylinder 4 was properly torqued was a performance deficiency warranting a significance determination. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 30, 2005, because it was associated with the equipment performance attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since inadequate maintenance practices led to restoring EDG2 to service in a condition such that the EDG would not have functioned as designed.

The finding was related to the cross-cutting area of Human Performance in that the original condition occurred due to the lack of procedure-required sign-offs for individual work activities and that an individual, other than the individual that performed the work, signed off the procedure to signify completion of the work.

The issue is being considered as NRC-identified because, in accordance with IMC 0612, Section 09.02c, although the issue was initially identified by licensee personnel, the finding was potentially greater than Green and required a Phase 2 and Phase 3 analysis.

Phase 1 Assessment

The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." In accordance with the "SDP Phase 1 Screening Worksheet for IE (Initiating Events), MS (Mitigating Systems), and BI (Barrier Integrity) Cornerstones," the inspectors determined that because the 7-day TS allowed outage time for EDG2 had been exceeded, the finding represented an actual loss of safety function of a single train of safety-related equipment for greater than its TS allowed outage time, and a Phase 2 SDP assessment was required.

Phase 2 Assessment

The inspectors utilized the "Loss of Offsite Power (LOOP)" and the "LOOP with Loss of One Emergency AC (LEAC)" Phase 2 SDP Worksheets and solved only those sequences that involved the EDG2 with a duration of 3-30 days. Based on the results of the SDP worksheets, the inspectors determined that the finding was potentially of low to moderate safety significance (White). A regional senior reactor analyst (SRA) reviewed these results and determined that an SDP Phase 3 assessment was necessary to refine the risk characterization. The licensee determined, and the inspectors agreed, that despite the damage to cylinder 4, EDG2 would have started, loaded, and run for between 10 and 16 hours of its 24-hour design basis mission time.

Phase 3 Assessment

<u>Internal Events</u> - The SRA performed a phase 3 risk evaluation using the Davis-Besse Standardized Plant Analysis Risk (SPAR) Model, Level 1, Revision 3P, Change 3.21, dated October 2005. The SRA assumed the failure of EDG2 to run after 10 hours throughout an exposure time of 280 hours. The exposure time was based on the time that the condition (i.e. inadequate torque) was reasonably known to have existed, and, in accordance with NRC Inspection Manual Chapter guidance documents, included the repair time.

The SRA also determined the initiating event frequency for LOOP events. The SRA used data from NUREG/CR-6890, Reevaluation of Station Blackout Risk at Nuclear Power Plants - Analysis of Loss of Offsite Power Events (1986-2004), and determined the frequency for LOOP events with durations greater than 10 hours to be 1.8E-3/yr.

The SRA also determined the nonrecovery probabilities for offsite power. Using data from NUREG/CR-6890 and the SPAR Model LOOP Event Tree, the SRA generated a curve that represented the probability of nonrecovery for LOOP events that exceed 10 hours. The SRA determined the probability of nonrecovery of offsite power within 1 hour, 2 hours, and 6 hours following a 10 hour LOOP to be estimated at: 0.99, 0.99, and 0.98 respectively.

The SPAR model included nonrecovery probabilities for emergency power; however, the model did not allow a nonrecovery probability to be assigned to a single EDG. To bound the calculation, the SRA set the nonrecovery probabilities for all emergency power sources to 1.0 after 10 hours.

The SRA ran the SPAR model using the above information and obtained a change in core damage frequency (ΔCDF) of 1.3E-7 (Green) for internal events. The dominant sequences involved loss of offsite power, failure of emergency power, failure to recover emergency and offsite power, and loss of reactor coolant subcooling through failure of feed and bleed.

The SRA also performed a hand calculation using the licensee's baseline CDF of 1.22E-5 and risk achievement worth value of 2.09, obtained from the NRC's pre-solved cases (Excel spreadsheet). Using the exposure time of 280 hours, the SRA calculated a ΔCDF of 4.3E-7 (Green).

In summary, the ΔCDF due to internal events was estimated to be 1.3E-7 to 4.3E-7.

External Events - The SRA reviewed the “Individual Plant Examination of External Events” (IPEEE) report for Davis-Besse, dated November 1996, to gain risk contribution insights from external events. Section 4.1.1, “Safe Shutdown Equipment,” of the report stated that equipment for systems relied upon to achieve safe shutdown as a result of a fire in any particular fire area was powered so that it would not be affected by a loss of offsite power. The SRA concluded that events due to fire were not significant contributors to the risk associated with this finding.

The SRA reviewed the IPEEE regarding seismic contributors. The IPEEE discussed success paths to place the plant in a safe shutdown state following a seismic event. In defining the paths, the IPEEE analysis included several assumptions, including one that the plant would be without offsite power for 72 hours and that core cooling and inventory functions would be maintained following a maximum possible earthquake of 0.2g. The SRA concluded that events due to seismic activity were not significant contributors to the risk associated with this finding.

Flooding risk was also incorporated into the licensee's PRA model and was part of the assessment performed by the licensee. The SRA's review of the licensee's risk assessment is discussed below.

Large Early Release Frequency (LERF) - Using IMC 0609, Appendix H, “Containment Integrity Significance Determination Process,” the SRA determined that this was a Type “A” finding for a pressurized water reactor with large dry containment. Using Table 5.1 and the SDP Phase 2 Worksheets, the SRAs concluded that none of the accident sequences were contributors to LERF. The attributes considered in Table 5.1 were inter-system loss of coolant accidents (LOCAs) and steam generator tube ruptures. None of these scenarios were impacted by this finding.

The SRA concluded that the ΔLERF was negligible and did not contribute to the risk associated with this finding.

<u>Overall Conclusion</u> - The SRA concluded that the total ΔCDF considering internal events, external events, and LERF was in the range of 1.3E-7 to 4.3E-7 (Green).

Licensee's Analysis:

<u>PRA Model</u> - Davis-Besse PRA, Revision 3, was the licensee's baseline model used for the analysis. This model was finalized in May 2002. The model includes internal events and flooding and inter-facing system LOCA. Revisions to the PRA EDG model were made based on the PRA Outlier Review instituted by the NRC during development of the Mitigating System Performance Index. These changes increased the risk-significance of the EDGs in the plant PRA. The PRA model used for this condition included the changes made under the PRA Outlier Review.

<u>Failure Determination</u> - The licensee's measurements taken from the exhaust valves of EDG2, cylinder 4, indicated that the rate of degradation would have allowed EDG2 to have started, loaded, and run for an additional 10 to 16 hours of the 24-hour design basis mission time. The licensee's analysis credited EDG2 running for a total of 12 hours as being representative of the 10 to 16 hour estimated range.

<u>Exposure Time</u> - The licensee split the exposure time for this deficiency into periods when the EDG 12 was available to start and run for 12 hours and a period when the EDG was unavailable due to maintenance. Specifically, the licensee assumed failure of EDG 12 to run after 12 hours throughout an exposure time of 6.7 days. This correlated to the time EDG 12 was declared operable on January 14, 2006, at midnight, to the time EDG was tagged out for additional testing on January, 20, 2006, at 3:57 p.m.

The time to repair EDG2 correlated to the time from January 20, 2006, at 3:57 p.m., to January 23, 2006, at 6:42 a.m., when EDG2 was again declared operable. This added an additional 2.6 days to the overall exposure time.

<u>Risk Increase</u> - The licensee calculated the conditional core damage probability (CCDP) values separately for the two periods: the 6.7 days and 2.6 days (total of about 223 hours), and summed them. The core damage frequency for EDG2 failing after running 12 hours was 2.6E-5/yr. The core damage frequency for EDG2 being unavailable for maintenance was 8.3E-5/yr. The baseline model core damage frequency is 1.7E-5/yr. The calculated increase in the CCDP for internal events was 6.4E-7.

The licensee's calculated increase in CCDP due to external events was Incremental CCDP(seismic) + Incremental CCDP(fire) = 6.6E-9 + 3.0E-8 = 3.7E-8. The licensee calculated the increase in large early release frequency (ΔLERF) to be 1.1E-9.

The licensee's total ΔCDF including internal events, external events, and LERF was therefore 6.4E-07 + 3.7E-8 + 1.1E-09 = 6.8E-7. The SRA determined that the licensee's analysis was of sufficient scope, detail, and quality, and the results reasonably matched the results obtained by the SRA.

Significance Determination Conclusion: The SRA concluded that the total ΔCDF for this issue, considering internal events, external events, and LERF was in the range of 1.3E-7 to 6.3E-7 (Green).

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," stated, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment and non-conformances are promptly identified and corrected. Contrary to the above, the source and impact of the condition causing the audible tapping noise heard during the post maintenance test of EDG2 on January 13, 2006, was not promptly identified and corrected until January 23, 2006. (NCV 05000346/2006003-03)

.4 Annual Sample: Review of Issues with Work on Wrong Train of Service Water

a. Inspection Scope

The inspectors reviewed CR 06-01502, "DB-FE11210 Flanges Disassembled with System in Service," and the associated licensee's root cause evaluation. The issue occurred during RFO 14 and involved contractor personnel performing tasks in the outage schedule. Because the licensee utilized a large number of contractors during the outage, the inspectors focused on the control of contractor activities. In addition, the inspectors determined whether additional issues associated with contractor performance existed, and, if so, whether these issues could be attributed to a common cause.

This constitutes one annual sample.

b. Findings and Observations

On March 31, 2006, during RFO 14, a contractor was assigned to disassemble the flanges of flow element DB-FE1121 per instructions in work order (WO) 200155492. This flow element measured service water train 2 flow from a component cooling water heat exchanger. The flow element orifice was scheduled to be replaced with a spacer in accordance with a design change package. However, contractor personnel mistakenly started to disassemble the flanges associated with service water train 1. At that time, service water train 1 was in service providing necessary cooling water flow.

The most significant consequence of the event was that personnel started to work on a train of cooling water that was in service. There was no flooding because the contractors, following accepted site practices, when, in the process of loosening connections and noticing water, assumed that the system had either some isolation boundary leakage or had not been totally drained and only loosened the connections sufficiently to allow slow drainage of the system. The contractors requested assistance from plant personnel to determine the cause of the leakage. Plant personnel did not identify that the wrong component in the wrong train was involved until an oncoming shift reviewed the issue.

The licensee classified the event as a significant condition adverse to quality (CR 06-01502) and performed a root cause analysis. The analysis concluded that the root causes included supervisory methods and ineffective corrective actions for previous worker self-checking issues. The analysis also concluded that the there were other examples of lack of sufficient oversight of contractor work forces. The licensee initiated CR 06-01576 to further investigate a potential adverse trend in human performance associated with contractor personnel.

The inspectors were aware that during RFO 14, the licensee experienced other issues with contractor work. The inspectors did not identify any contractor oversight issues of significance beyond those identified by the licensee. The inspectors also did not find evidence that contradicted the licensee analysis that the occurrence of working on the wrong train or wrong components was an infrequent event at Davis-Besse. The inspectors noted that the licensee identified that procedure NOBP-WM-2502, “Contract Management and Oversight,” which had been developed for contractor control issues in previous outages, was ineffective in preventing the recurrence of issues.

4OA3 Event Followup (71153)

.1 (Closed) LER 05000346/2006-001-00: Emergency Diesel Generator Engine Damaged Due to Improper Torquing of Lock Nut

As discussed in NRC IR 05000346/2006002, on January 20, 2006, the licensee identified damage to cylinder 4 of EDG2. Section 4OA2 of this report documents the inspectors review of this issue. The LER did not identify any additional issues. This LER is closed.

This constitutes one sample.

.2 (Closed) LER 05000346/2005-004-00: Containment Hydrogen Analyzers Inoperable Due to Presence of Check Valves in Moisture Removal System

On November 28, 2005, the licensee submitted LER 05000346/2005-004-00, which documented the inoperability of the containment hydrogen analyzers due to the presence of undocumented check valves in the moisture removal system. The check valves prevented the proper drainage of accumulated condensate. The conditions described in the LER were reviewed in IR 05000346/2005009 and were found to constitute an NCV (NCV 05000346/2005009-02). The inspectors review of the LER did not identify any items of significance that were not addressed in the NCV. This LER is closed.

This constitutes one sample.

.3 (Closed) LER 05000346/2002-009-02: Degradation of the High Pressure Injection Thermal Sleeves

LER 05000346/2002-009-00, “Degradation of the High Pressure Injection Thermal Sleeves,” was previously evaluated and closed by inspectors in IR 05000346/2003010(DRS). No violations of regulatory requirements or findings of significance were identified in that report.

On March 26, 2004, the licensee submitted Revision 01 to this LER. The revision updated the results of the licensee's root cause evaluation and the commitments that addressed this issue. The commitments included: 1) revising the Augmented Inservice Inspection Program to include ultrasonic and radiographic testing of the 2-1 and 2-2 high pressure injection thermal sleeves during RFO 14; 2) revising the Augmented Inservice Inspection Program to include an augmented VT-1 visual examination of the HPI/MU thermal sleeve once every other refueling outage, commencing with RFO 15; and 3) completion of an engineering change request to determine the long-term action for thermal sleeve crack initiation. This revision was previously evaluated and closed by inspectors in IR 05000346/2004007.

On May 17, 2006, the licensee submitted Revision 02 to this LER to update the inspection methods used during RFO 14 and to provide a revision to the schedule of future inspections. Because of a change in the RFO 14 scope, the thermal sleeves of concern were visually examined and the licensee reported that no cracks were found. Visual examination is the licensee's preferred method of examination. Since the visual examination was conducted during RFO 14, the licensee planned to revise the inspection schedule to perform examinations of the high pressure/makeup nozzle thermal sleeve every other refueling outage beginning in RFO 16. No items of significance were identified. This LER is closed.

This constitutes one sample.

.4 (Closed) URI 05000346/2006002-03: Inoperability of EDG2 Due to Valve Rocker Arm Damage

As discussed in NRC IR 05000346/2006002, on January 20, 2006, the licensee, identified damage to the number 4 cylinder of EDG2. Section 4OA2 of this report documents the inspectors review of this issue. The LER did not identify any additional issues. This URI is closed.

This constitutes one sample.

4OA5 Other Activities

.1 Review of Institute of Nuclear Power Operations Report

The inspectors completed a review of the Institute of Nuclear Power Operations (INPO) final Davis-Besse Accreditation Report for operator training programs that was dated January 11, 2006.

.2 Transient Combustibles on the Dry Fuel Storage Pad

a. Inspection Scope

The inspectors, while conducting a quarterly fire program sample, reviewed the fire loading assumptions in the spent fuel dry horizontal storage modules (HSMs) Certificate of Compliance.

b. Findings

Introduction: The inspectors identified a finding involving an NCV of 10 CFR 72.212, “Conditions of general license issued under §72.210,” having very low safety significance (Green) for non-compliance with transient combustible material control procedures required for the Davis-Besse spent fuel dry horizontal storage modules (HSMs).

Description: On April 10, 2006, while performing a routine fire protection inspection of the HSMs, the inspectors noted a Sealand container, that contained combustible materials, within 50 feet of the HSMs. Other Sealand containers were noted within 50 feet of the HSM that had combustible traps on top of the containers. Three additional Sealand containers that were within 50 feet of the HSMs were not on the licensee’s inventory of containers located in the 50 foot area. The licensee also identified that a truck trailer, located on the storage pad, had hydraulic fluid stored in it. The issues identified were not in compliance with the licensee’s procedures, specifically DF-FP-00007 for control of combustible transient material. Control of transient combustible material was required to ensure conformance with temperature limitations for the HSMs as outlined in the NRC-issued HSM Certificate of Compliance.

The licensee generated CR 06-01753 to document these issues and the corrective actions necessary to be in full compliance with their procedures. The combustible material was removed from the Sealand container within 50 feet of the HSMs. The Sealand containers with tarps and containers not appearing on the inventory were moved to an area greater than 50 feet from the HSMs. The issue with the trailer containing hydraulic fluid was turned over to the licensee’s site projects group to have the hydraulic fluid removed. On May 19, 2006, the inspectors questioned the licensee on the status of corrective actions taken for the identified issues and were informed that the hydraulic fluid had not been removed from the trailer. The inspectors informed licensee management and the licensee took immediate action to remove the hydraulic fluid from the trailer. The licensee initiated CR 06-02340 to enter this issue into their corrective action program.

Analysis: The inspectors determined that the failure to follow fire protection procedures developed for control of transient combustible material stored on the dry spent fuel storage pad in close proximity to the HSMs was a performance deficiency that warranted a significance evaluation. This finding was greater than minor because it was associated with the protection against potential fire damage to the HSMs and, if left uncorrected, would become a more significant safety concern since the prolonged presence of combustible material in the vicinity of the HSMs increased the vulnerability of the HSMs to a fire. The inspectors determined that the finding was not suitable for SDP evaluation because the noncompliance did not involve permanently installed plant equipment. Therefore, this finding was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c and determined to be of very low safety significance (Green). 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

Human Performance because licensee personnel failed to adhere to licensee procedures associated with the control of combustibles in the vicinity of the HSMs.

Enforcement: 10 CFR 72.212 "Conditions of general license issued under §72.210," section b(9) stated, in part, that the licensee shall "Conduct activities related to storage of spent fuel under this general license only in accordance with written procedures." Procedure DB-FP-00007, "Control of Transient Combustibles," provided, among other things, controls for limiting transient combustible material in the area around the HSMs. Contrary to the above, transient combustibles were stored on the dry spent fuel storage pad inside the area prohibited by station procedures. Once identified, the licensee initiated actions to remove the transient combustible material and entered the issue into its corrective action program as CR 06-01753, and CR 06-02340 on April 10 and May 19, 2006, respectively. However, because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000346/2006003-04)

.3 (Closed) URI 0500346/2004007-01: Startup and Bus Tie Transformer Fast Transfer Capability Not Reflected in SR 4.8.1.1.1.b

a. Inspection Scope

During the inspection period, the inspectors reviewed URI 05000346/2004007-01.

b. Findings

Introduction: A finding of very low safety significance and an associated NCV of TS Section 6.17, "Technical Specifications Bases Control Program," was identified when licensee personnel inappropriately revised the TS Bases to credit the fast transfer capability of the 13.8 kV bus tie transformers for offsite power source operability without required NRC approval.

Description: In April 2004, while evaluating issues related to CR 04-02511 and CR 04-02522, the inspectors reviewed the electrical configuration of the plant prior to the restoration of breaker ABDD2 to normal operating status. Before returning breaker ABDD2 to service, both divisions of the electrical plant loads were supplied through bus tie transformer AC. While in this configuration, the licensee did not enter Limiting Condition for Operability (LCO) 3.8.1.1, "A.C. Sources," because the licensee considered both offsite sources to be available. This required that a single electrical system failure only caused the loss of a single offsite power source. The inspectors identified that the licensee relied on the fast transfer of electrical loads between 13.8 kV bus tie transformers AC and BD in certain fault scenarios to ensure that a single offsite power source was available. However, the inspectors determined that TS 3.8.1.1 did not address the operability aspects of the fast transfer capability. Additionally, the inspectors determined that TS 3.8.1.1 did not include a surveillance requirement to ensure that the fast transfer capability for the 13.8 kV bus tie transformers was operable. Consequently, although licensee personnel credited the fast transfer capability of the 13.8 kV bus tie transformers for offsite source operability, a TS surveillance requirement did not exist to test the fast transfer function.

Surveillance Requirement (SR) 4.8.1.1.1.b required that offsite sources be demonstrated operable at least once each refueling interval during shutdown by transferring manually and automatically the unit power supply to each of the offsite circuits. Surveillance Requirement 4.8.1.1.1.b addressed testing the transfer from the unit auxiliary transformer (UAT) to the startup transformers, but did not address testing the automatic transfer capability between the two startup transformers and the associated 13.8 kV bus tie transformers. In October 2003, the licensee revised the TS Bases and interpreted SR 3.8.1.1.1.b to include the testing of the 13.8 kV bus tie transformer fast transfer function. Specifically, the revised TS Bases stated that SR 4.8.1.1.1.b was to be performed at least once each refueling interval, during shutdown by demonstrating the capability of manual and automatic transfer of each 13.8 kV bus power supply from the unit auxiliary transformer to each offsite power source, and from each offsite power source to the other offsite power source. The inspectors noted that this change expanded on the scope of SR 4.8.1.1.1.b that only tested the transfer from the UAT to the other offsite power sources, but did not include the testing of the fast transfer capability between the 13.8 kV bus tie transformers. The inspectors determined that this action did not meet the requirements of TS Section 6.17, "Technical Specifications Bases Control Program," which stated, in part, that the licensee may make changes to the TS Bases without prior NRC approval provided the changes do not require a change in the TS incorporated in the license. The inspectors concluded that the addition of the 13.8 kV bus tie transformer fast transfer function as a requirement for operability associated with TS 3.8.1.1 should have resulted in a revision to the TSs.

Additionally, the licensee had performed Regulatory Applicability Determination (RAD) No. 03-01949 to evaluate this TS Bases change. The RAD screening concluded that the activity did not require a license amendment because the TS Bases were not part of the TS. The inspectors determined that this conclusion was incorrect because the revision to the TS Bases required the testing of the 13.8 kV bus tie transfer function, which was not within the scope of SR 4.8.1.1.1.b as it existed. Had the requirements of TS Section 6.17 been adhered to, the licensee should have determined in RAD 03-01949 that a license amendment was required since a TS change was necessary to include a periodic verification of the 13.8 kV bus tie transformer fast transfer function.

Analysis: The inspectors determined that the failure to request a license amendment to incorporate a change to TS 3.8.1.1 to add a surveillance requirement that verified the 13.8 kV bus tie transformer fast transfer function prior to taking credit for this capability was a performance deficiency that warranted a significance determination.

The finding was determined to be more than minor because the TS Bases change required a license amendment that was not submitted; the finding affected equipment important to safety; and the failure to submit the license amendment potentially impeded or impacted the regulatory process. Consequently, this issue was dispositioned using the traditional enforcement process.

The underlying technical issue was evaluated using the SDP to determine the significance of the violation. In this case, the issue was determined to be of very low significance using IMC 0609, Appendix A, "Significance Determination of Reactor

Inspection findings for the At-Power Situations," because (1) it did not represent an actual loss of safety function of a system; (2) it did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (3) it did not represent an actual loss of safety function of one or more non-TS trains of equipment designed as risk-significant per 10 CFR 50.65 for greater than 24 hours; and (4) it did not screen as potentially risk-significant due to a seismic, fire, flooding, or severe weather initiating event. In particular, there were no actual consequences as a result of the issue because the licensee had periodically verified the 13.8 kV bus tie transformer's fast transfer capability. Therefore, in accordance with the Enforcement Policy, the violation was assigned a Severity Level IV significance level.

Enforcement: Technical Specification Section 6.17, "Technical Specifications Bases Control Program," states, in part, that the licensee may make changes to the TS Bases without prior NRC approval provided the changes do not require a change in the TS incorporated in the license.

Contrary to the above, in April 2004, the licensee changed the TS Bases associated with TS 3.8.1.1.1.b to credit the fast transfer function of the startup transformers and associated 13.8 kV bus tie transformers, without prior NRC approval although the change required a change to TS 3.8.1.1 to periodically verify this capability. In accordance with the Enforcement Policy, this violation of TS Section 6.17, "Technical Specifications Bases Control Program," was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance since the licensee had periodically verified the 13.8 kV bus tie transformer fast transfer capability. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (CR 04-02552), the violation is considered to be an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2006003-05).

.4 Confirmatory Order Related Activities

On March 8, 2004, the NRC issued "Confirmatory Order Modifying License No. NPF-3 (EA-03-0214)," which required, in part, that the licensee perform annual independent assessments, for a period of 5 years, in the areas of operations performance; organizational safety culture, including safety conscious work environment; corrective action program implementation; and engineering program effectiveness. The following activities associated with EA-03-0214 were reviewed during this inspection period.

.a Review of Calendar Year (CY) 2006 Operation Performance Independent Assessment Plan.

(1) Inspection Scope

As part of the inspection activities performed to verify the licensee's compliance with the requirements for independent assessments, as described in the March 8, 2004, Confirmatory Order Modifying License No. NPF-3 (EA 03-214), the inspectors verified that the licensee submitted, per letter dated March 14, 2006, the required inspection plan for the Operations Independent Assessment prior to the performance of the CY2006 annual Operations Assessment, which was scheduled for June 2006. As part of the inspection activities, the inspectors reviewed the scope of the Independent Assessment Plan and the qualifications of the team members designated to perform the assessment.

(2) Observations and Findings

After evaluating the Operations Performance Independent Assessment Plan for CY2006, the inspectors determined that the scope and depth of activities outlined in the plan would be sufficient to obtain an appropriate assessment of Operations department performance.

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

.b Observation of CY2006 Operations Performance Independent Assessment

(1) Inspection Scope

The March 8, 2004, Confirmatory Order (EA 03-0214) required independent assessment of Operations performance for CY2006 was conducted on-site from June 12 to June 23, 2006. Prior to this assessment, the inspectors reviewed the qualifications of the Assessment Team and the Operations Performance Assessment Plan.

The inspectors also evaluated the on-site activities. In particular, the inspectors attended licensee debriefs, monitored in-process evaluations, and discussed preliminary findings with assessment team members. Additionally the inspectors observed independent assessment activities to determine the effectiveness of the assessment and the potential impact, if any, of the unexpected unavailability of one of the team members for part of the assessment.

(2) Findings and Observations

No findings of significance were identified.

The inspectors concluded that the assessment team conducted all activities prescribed by the Operation Performance Assessment Plan. A discussion of the results of the independent assessment of Operations was planned, but had not been scheduled at the end of the inspection period. The March 8, 2004, Confirmatory Order required that the licensee provide the NRC Region III Administrator with all assessment results and actions planned to address the assessment results, within 45 days of the completion of the independent assessment.

.c Safety Culture/Safety Conscious Work Environment (SC/SCWE) Independent Assessment, CY2005

(1) Inspection Scope

By letter dated January 27, 2006, (ML060330132), FirstEnergy Nuclear Operating Company (FENOC) submitted the results of the SC/SCWE independent assessment performed between November 1 and December 14, 2005. The letter also provided an Action Plan to address areas for improvement identified during the assessment.

On December 14, 2005, Dr. Sonja Haber of Human Performance Analysis Corp. presented the results of the independent assessment to plant personnel. During the week of April 17, 2006, the inspectors reviewed the assessment report and the actions planned to address the identified issues. In addition, the inspector reviewed a May 26, 2006 FENOC letter, entitled “Change to Completion Date of Action in Response to 2005 Safety Culture Independent Assessment Area for Improvement COIA-SC-05-03,” which amended the Action Plan implementation date for one activity. The implementation date was delayed due to unanticipated length of RFO 14.

(2) Findings and Observations

The inspector concluded that the licensee’s Action Plan had adequately addressed all areas for improvement identified in the SC/SCWE independent assessment report. The areas for improvements were:

(a) “A long-term strategy to ensure the organization’s continued and sustainable commitment to safety still needs additional focus and development.”
(b) “While improvement in values and attitudes have been observed since the 2004 Independent Assessment, they are generally back to the levels obtained in the 2003 Independent Assessment during the long outage. Davis-Besse leadership behaviors need to demonstrate continuing improvement and sustainability across all levels of the organization to ensure the desired outcomes. The top down style of management, previously identified, while effective for short-term results, will not result in long-term sustained success.”
(c) “Efforts to improve Davis-Besse’s performance by learning from its past performance, from industry performance, from internal and external assessments, and from the day-to-day implementation of its own programs and processes, still are not effectively implemented nor recognized to be of high value to the organization.”
(d) “The overall rating of White on Davis-Besse’s Annual Safety Culture Assessment is noted to be a conservative one as their actual numerical calculation was equivalent to a Green rating. While the team recognizes this as a positive step, the results of the 2005 Independent Assessment are more critical of the current status of Davis-Besse’s Safety Culture and Safety Conscious Work Environment and have provided an overall assessment as Marginally Effective.”

In addition, regarding the completion date for COIA-SC-05-03, the inspectors concluded that the delay in the completion date was acceptable given the extended refueling outage.

Overall, the inspectors concluded that the licensee had met requirements contained in the NRC's March 8, 2004, letter, "Approval to Restart the Davis-Besse Nuclear Power Station, Closure of Confirmatory Action Letter, and Issuance of Confirmatory Order" in the SC/SCWE area.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. B. Allen and others members of the licensee management on June 22, 2006. The inspectors asked the licensee whether any material presented should be considered proprietary. No presented material was identified as proprietary.

.2 Interim Exit Meetings

An interim exit associated with NCV 05000346/2006000-04 and closure of URI 05000346/2004007-01 was conducted with Mr. B. Allen and other licensee personnel on June 21, 2006.

An interim exit associated with access control to radiologically significant areas, the radiological environmental monitoring program, and radioactive material control program was conducted with Mr. B. Allen on May 18, 2006.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is 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 6.8.1 required that written procedures shall be established and implemented for applicable procedures recommended in Appendix A of Regulatory Guide 1.33, February 1978. Regulatory Guide 1.33 required that maintenance, potentially affecting the performance of safety-related equipment, be preplanned and performed in accordance with written procedures appropriate to the circumstances. For re-assembly of the reactor vessel and reactor vessel stud tensioning, contractor procedures provided pre-planned steps and were utilized. Contrary to the above, following the identification that the reactor vessel head studs were under-tensioned, licensee personnel used procedure steps other than those specified for correcting the condition. This resulted in the over-tensioning of 36 reactor vessel studs. This issue was entered in the licensee's corrective action program as CR 06-01701. The finding was of very low safety significance because no reactor vessel or reactor vessel stud damage occurred as a result of the performance deficiency.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

B. Allen, Director, Plant Operation
M. Bezilla, Site Vice President
B. Boles, Manager, Plant Engineering
R. Farrell, Director, Site Maintenance
L. Harder, Manager, Radiation Protection
S. Loehlein, Director, Station Engineering
K. Ostrowski, Manager, Plant Operations
C. Price, Manager, Regulatory Compliance
R. Schrauder, Director, Performance Improvement

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open and Closed

05000346/2006003-01	FIN	Unplanned Heatup of RCS Due to Loss of Component Cooling Water to In-Service Decay Heat Cooler During Mode 5
05000346/2006003-02	NCV	Radiation Workers Entered Posted High Radiation Areas Without Proper Authorization
05000346/2006003-03	NCV	Inoperability of EDG2 Due to Untimely Corrective Action to Find and Repair Exhaust Valve Rocker Arm Damage
05000346/2006003-04	NCV	Fire Protection, Transient Combustibles on the Dry Fuel Storage Pad
05000346/2006003-05	NCV	Startup and Bus Tie Transformer Fast Transfer Capability Not Reflected in SR 4.8.1.1.1.b

Closed

05000346/2002-009-02	LER	Degradation of the High Pressure Injection Thermal Sleeves
05000346/2005-004-00	LER	Containment Hydrogen Analyzers Inoperable Due to Presence of Check Valves in Moisture Removal System
05000346/2006-001-00	LER	Emergency Diesel Generator Engine Damaged Due to Improper Torquing of Lock Nut
05000346/2006002-03	URI	Inoperability of EDG2 Due to Valve Rocker Arm Damage
05000346/2004007-01	URI	Startup and Bus Tie Transformer Fast Transfer Capability Not Reflected in SR 4.8.1.1.1.b

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

DB-OP-06316; Emergency Diesel Generator Operating Procedure; Revision 25
DB-OP-06913; Seasonal Plant Preparation Checklist; Revision 14
WO 200147636; PM 6382 - Replace Winter Oil in Filter F108-1
WO 200147637; PM 6383 - Replace Winter Oil in Filter F108-2
Select Computerized Completed Zone 1 and 2 Operator Round Sheets; May 27, 2006 through June 2, 2006

1R04 Equipment Alignment

DB-OP-06011; High Pressure Injection System; Revision 15
DB-OP-06012; Decay heat and Low Pressure Injection System Operating Procedure; Revision 26
DB-OP-06316; Emergency Diesel Generator Operating Procedure; Revision 25
Drawing OS-003; High Pressure Injection System; Revision 28
Drawing OS-004, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 42
Drawing OS-041A, Sheet 1; Emergency Diesel Generator System; Revision 27
Drawing OS-041A, Sheet 2; Emergency Diesel Generator System; Revision 25
Drawing OS-041B; Emergency Diesel Generator Air Start/Engine Air System; Revision 29
Drawing OS-041C; Emergency Diesel Generator Diesel Oil System; Revision 16

1R05 Fire Protection

Davis-Besse Nuclear Power Station Fire Hazard Analysis Report
CR 06-00438; Failure To Meet Requirements For Storing RCP Motor On The Dry Fuel Storage Pad
CR 06-01753; Station Not In Compliance With DB-FP-00007 (NRC Identified)
CR 06-01786; Operability Determination for CR 06-01753 Did Not Address 10CFR72.212 Evaluation
CR 06-02340; Potential Violation Of 10CRF72.122C (NRC Identified)
DB-FP-00007; Control Of Transient Combustibles; Revision 07
DB-NE-03400; Horizontal Storage Module (HSM) Monitoring; Revision 02
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-230F; Fire Protection Intake Structure; Revision 9
Notification 600267243; Store RCP Motors On the Dry Fuel Storage
Notification 600293089; CR 06-00438 Resolution-Revise RAD06-00315
NRC Certificate Number 1004; Certificate of Compliance for Dry Spent Fuel Storage Casks; Revision 00

1R06 Flood Protection

RA-EP-02880; Internal Flooding; Revision 03
Drawing OS-47B, Sheet 3; Fire Suppression System; Revision 5
Drawing OS-53, Sheet 1; Station Drainage; Revision 30
Calculation 15.50; Evaluation of Fire Suppression System Impact on Auxiliary Building and Intake Structure Flooding; Revision 1
Calculation C-CSS-099.20-024; Assessment of Cover Plates and Slab for Flood Loads on Pump Holes at Intake Structure Slab EL 576'-0"; Revision 1

1R11 Licensed Operator Requalification Program

DBBP-TRAN-0017; Conduct of Simulator Training; Revision 02
Davis-Besse Emergency Response Integrated Drill Manual; Revision 00

1R12 Maintenance Effectiveness

D-B System Health Report, Turbine Generator Window; First Quarter, 2006
Maintenance Rule Program Manual; Revision 20
CR 06-02466; Main Generator Seal Oil System Failure
CR 06-01977; No. 1 Steam Packing Exhauster Fan Motor Failure
CR 06-02218; Master Trip Solenoid “A” Failure
DB-PF-00003; Maintenance Rule; Revision 08
WO 200215203; DB-SV2210B: Troubleshoot Failure to Trip
WO 200215216; DB-SV2210B: Replace Master Trip Solenoid Valve

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

CR 05-01152; PORV Leakage
CR-06-02042; PORV Leaking After PORV Cycle Test DB-SP-03363
DBBP-OPS-0003; On-line Risk Management Process; Revision 02
DB-OP-06002; RCS Draining and Nitrogen Blanketing; Revision 13
DB-OP-06904; Shutdown Operations; Revision 21
DB-SC-03023; Off-Site AC Sources Lined Up and Available; Revision 16
NG-DB-00001; On-line Risk Management; Revision 03
NOP-ER-3001-1; Problem Solving Plan, CR06-02042 Pilot Operated Relief (PORV) Leaking After PORV Cycle Test DB-SP-03363; Revisions 00 and 02
NOP-ER-3001-1; Problem Solving Plan, CR05-01152 PORV Leakage; Revision 00
Contingency Plan 14RFO-2; RCS Drain Below Flange Level and Operation Below 8” Above the RCS Hot Leg Centerline; Revision 05
WO 200188115; Repair/Replace 10" Fire line near X1 located west of the Main transformer
WO 200201537/WO 200201538; Installation of TM 06-0004 and Removal of TM 06-0004

1R14 Operator Performance During Non-Routine Evolutions and Events

CR 06-01498; RCP 1-2-1 and 1-2-2 Stud Elongation Exceeded Limits of DB-MM-09117
CR 06-01701; Elongation Readings on Reactor Head Studs Out of Tolerance
CR 06-01724; All AREVA Worked Stopped by the Shift Outage Director
DB-MM-06002; Polar Crane Operation; Revision 11

DB-MM-09089; Reactor Vessel Head Stud Removal and Reinstallation; Revision 02
DB-NE-03212; Zero Power Physics Testing; Revision 06
DB-OP-06900; Plant Heatup; Revision 30
DB-OP-06912; Approach To Criticality; Revision 07
Procedure 03-5016144-00; RV Closure Head Stud Replacement; Revision 00
(Areva/Framatome ANP Proprietary)

1R15 Operability Evaluations

CR 06-00779; Three Tubes Found Plugged in Containment Air Cooler E37-3
CR 06-01035; Tube Plugging Found in Containment Air Cooler #3 (E37-3)
CR 06-01243; Piece of Grid Strap Debris not Removed from Rx Vessel
CR 06-01339; Cain Broke on Reconstitution Rod Manipulator Tool
CR 06-01342; Observed Missing Grid strap Pieces from 14RFO Offlaod and Detailed Visuals
CR 06-01400; Debris Removal Unsuccessful For Reinsert Fuel Assembly NJ10L4
CR 06-01471; Flow Obstructions Found in CTMT Air Cooler E37-1 Tubes
CR 06-01600; Flow Obstruction Found in Ctmt Air Cooler E37-2 Tubes
CR 06-02382; DH14A Stroke Time Outside Of Expected Time Range
DB-PF-03206; ECCS Train 2 Valve Test; Revision 09
DB-OP-06911; Pre-Startup Checklist; Revision 12
NOBP-OM-4010; Restart Readiness For Plant Outages; Revision 03
Document Identifier 51-50173340-002; Reactor Operations with Loose Parts at Davis-Besse; March 29, 2006
ISTB3; Pump and Valve Basis Document Volume III Stroke Time Basis; Revision 27
Operability Evaluation 06-002; DH14A Stroke Time Outside of Expected Time Range; May 26, 2006
Calculation C-NSA-001.01-016; Service Water System Design Basis Flowrate Analysis and Testing Requirements; Revision 00
Calculation C-NSA-065.05-007: CAC Heat Duty at Elevated SW Inlet Temperatures; Revision 2
Specification M-401Q; Technical Specification for Replacement Containment Cooling Coils; Revision 01

1R19 Post-Maintenance Testing

CR 06-01829; Request Design Engineering To Evaluate #2 M/U Pump Baseline Data; April 14, 2006
CR 06-01831; MUP 1-2 Baseline Test, Motor Data DB-PF-05064 Greater Than 100% Full Load Amps; April 14, 2006
CR 06-01920; Evaluation Needed for DB-SP-04153 Acceptance Criteria On AFPT 2 HHS; April 18, 2006
CR 06-02084; NRC Question Regarding Procedure Adherence During DB-SC-03270; April 24, 2006 (NRC Identified)
DB-OP-03013; Containment Daily Inspection & Containment Closeout Inspection; Revision 03
DB-PF-03477; Makeup Pump 2 Baseline Test; Revision 01
DB-SC-03270; Control Rod Assembly Insertion Time Test; Revision 04
DB-SP-03338; Containment Spray Train 2 Quarterly Pump And Valve Test; Revision 14
DB-SP-04152; APFT 1 HSS And Overspeed Trip; Revision 12
DB-SP-04153; APFT 2 HSS And Overspeed Trip; Revision 11

DB-SS-04163; Main Turbine Overspeed Trip Test; Revision 06
DB-OP-06902; Turbine Operating Procedure; Revision 13
DB-SC-10003; Site Acceptance Test Plan for Replacement EDG2 Excitation System; Revision 00
CR 06-01900; LLRT Test Lineup Changes Went Unnoticed
DB-PF-0300 8; Containment Local Leakage Rate Tests; Revision 07
DBBP-DBTS-0002; Use of Leak Rate Monitor Test Equipment; Revision 00
WO 200139525; DB-CV5007: Replace Valve Seat (14RFO)

1R20 Refueling and Outage Activities

DB-OP-01003; Operations Procedure Use Instructions; Revision 06
DB-OP-06002; RCS Drain And Nitrogen Blanketing; Revision 13
DB-Op-06012; Decay Heat And Low Pressure Injection System Operating Procedure; Revision 25
DB-OP-06902; Power Operations; Revision 14
DB-OP-06904; Shutdown Operations; Revision 21

1R22 Surveillance Testing

CR 06-01114; CC1467 Failure to Operate as Expected
CR 06-01695; Breaker ABDC1 Failed to Close
CR 06-01870 DB-SC-04053 D1/D2 Transfer and Lockout Test, Inappropriate Procedure Steps
CR 06-01902; Performance of D1/D2 Lockout Procedure Use and Adherence
CR 06-01911; Procedure Inadequacies During Performance of D1/D2 Lockout Test DB-SC-4052
CR 06-01954; Containment Air Cooler #3 as #2 Flow Less than Flow Balance Acceptance Criteria
CR 06-01963; Decay Heat System Temperature Increase During DH207 Check Valve Flow Testing
CR 06-02322; Component Cooling Water System Procedure DB-OP-06262 Limit & Precaution 2.2.8
DP-OP-03001; Service Water Loop 2 Integrated Flow Balance Procedure; Revision 07
DB-OP-03013; Containment Daily Inspection & Containment Closeout Inspection; Revision 03
DB-OP-06262; Component Cooling Water System Procedure; Revision 13
DB-PF-03072; Component Cooling Water Pump 1 Test; Revision 09
DB-PF-06703; Miscellaneous Operation Curves; Revision 10
DB-SC-03114; SFAS Integrated Response Time Test; Revision 08
DB-SC-04052; 4160V System Transfer and Lockout Test Buses D1 and D1; Revisions 02 and 03
DB-SP-03134; Containment Emergency Sump Visual Inspection; Revision 03
DB-SP-03357; RCS Water Inventory Balance; Revision 09
DB-SP-03136; Decay Heat Train 1 Pump and Valve Test; Revision 11
DB-SS-04151; Main Turbine Control Valve Test; Revision 07
Operations Standing Order 06-003; CREVS Train 1, Design/Licensing Basis; Revision 01

1R23 Temporary Plant Modifications

TM 06-0016; DB-RD13, MSR First Stage Reheat Drain Tank 1 to HP Feedwater Heater 1-5 Check; May 4, 2006

1EP6 Drill Evaluation

Davis-Besse Emergency Response Integrated Drill Manual; 2006
CR 06-02630; E-Plan Drill 6-22-06 Items Noted In Simulator Control Room
CR 06-02631; E-Plan Drill 6-22-06 NRC Comments Following Critique in Simulator
RA-EP-01500; Emergency Classification; Revision 06

2OS1 Access Control to Radiologically Significant Areas

CR06-01565; Worker Received Dose Alarm; dated April 3, 2006
CR06-01790; Dose Rate Alarm, RP Worker; dated April 6, 2006
CR06-01458; NPS Worker Received Dose Alarm; dated March 30, 2006
CR06-01497; Dose Rate Alarm For RWP 2006-6021; dated March 25, 2006

2PS3 Radiological Environmental Monitoring Program and Radioactive Material Control Programs

NUPIC Joint Audit Survey of Environmental, Inc. 19238; dated January 18, 2006
Audit Report DB-C-05-02; dated July 27, 2005
Davis-Besse Annual Environmental Operating Report For 2004
Davis-Besse Annual Environmental Operating Report For 2005
Davis-Besse Offsite Dose Calculation Manual; Revision 20
CR04-07470; ODCM Change Request Condition Report Enhancement; dated December 7, 2004
CR05-02731; PCR: CR Enhancement To DB-CH-04041; dated May 12, 2005
CR05-03872; Meteorological Program Data Reporting Issues; dated July 15, 2005
CR05-04062; Malfunction-REMP Air Sampler; dated July 26, 2005
CR05-03713; Broad Leaf Vegetation Sample Not Available For Collection; dated July 6, 2005
CR06-02289; Both Primary and Secondary Meteorological Tower Differential Temperature Instruments Inoperable; dated May 12. 2006
DB-CN-00015; Radiological Environmental Monitoring Program; Revision 1
DB-CN-03005; Radiological Monitoring Weekly, Semi-monthly and Monthly Sampling; Revision 2
Air Sampler Calibration Records (Selected)
DB-HP-01706; Vehicle And Material Release From The RRAs And The Restricted Area; Revision 8
Germanium Detectors 1 and 2 Efficiency Calibration Confirmation Checks; dated October 12, 2005
LLD Verifications For Germanium Detectors 1 and 2

4OA1 Performance Indicator (PI) Verification

DBBP-RA-0004; NRC Performance Indicator - Mitigating System Cornerstone Safety System Functional Failure; Revision 00
Performance Indicator Data Input Sheets for Safety System Functional Failures January 2005 though March 2005

4OA2 Identification and Resolution of Problems

Condition Report Monitoring 14th Refueling Outage; May 23, 2006
CR 06-00154; #2 EDG Broken Parts In The rocker Arm Area; January 20, 2006
CR 06-01502; DB-FE11210 Flanges Disassembled With System In Service
CR 06-01576; Possible Trend in Human Performance of Non-Station Personnel
Davis-Besse Fleet Oversight Assessment Report DB-C-05-04; February 3, 2006
Davis-Besse fleet Oversight Assessment Report DB-C-06-01; May 4, 2006
Davis-Besse Oversight 14RFO Binning and Trending of Condition Reports; April 20, 2006
ECR 04-0216-01; Service Water "18"-HBC-42 Return Header from CCW Heat Exchangers Annubar Flowmeters; Revision 00
LER 2006-001-01; Emergency Diesel Generator Engine Damaged Due to Improper Torquing of Lock Nut; March 21, 2006
NOP-LP-2001; Corrective Action Program; Revision 13
NOBP-WM-2502; Contract Management and Oversight; Revision 02
OPS IPA 2006-01; Operations Integrated Performance Assessment November 1, 2005 - April 30, 2006; June 2, 2006
Plant Health Committee Issue Summary; May 10, 2006
Plant Health Report 4th Quarter 2005; February 24, 2006
Quarterly Program Health Report 2006-01, Davis-Besse
WO 200155492; ECR 04-02116-01 Remove FE1211 Orifice

4OA5 Other

CR 04-02552; ABDD2 Failure to Trip - Application of LCO 3.8.2.1
CR 1999-0492; NRC Resident Questions Operability of Electrical Distribution System Alignment
PCACR 97-0064; TS Operability Concerning the Offsite Transmission Network and the Onsite Class 1E AC Electrical Distribution System; January 29, 1997

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
ALARA	As-Low-As-Reasonably-Achievable
BI	Barrier Integrity
CAP	Corrective Action Program
CCDP	Conditional Core Damage Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
FENOC	FirstEnergy Nuclear Operating Company
HIC	High Integrity Container
HPI	High Pressure Injection
HRA	High Radiation Area
HSM	Horizontal Storage Module
IE	Initiating Events
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
LCO	Limiting Condition for Operability
LER	Licensee Event Report
LERF	Large Early Release Frequency
LEAC	Loss of Emergency Alternating Current
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MS	Mitigating Systems
NCV	Non-Cited Violation
NRC	United States Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulations
ODCM	Offsite Dose Calculation Manual
PI	Performance Indicator
PORV	Pilot Operated Relief Valve
PSA	Probabilistic Safety Analysis
RAD	Regulatory Applicability Determination
Radworkers	Radiation Workers
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Environmental Technical Specifications
RFO	Refueling Outage
RP	Radiation Protection
RWP	Radiation Work Permit
SC	Safety Culture

SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
SFAS	Safety Features Actuation System
SPAR	Standardized Plant Analysis Risk
SR	Surveillance Requirement
SRA	Senior Reactor Analyst
TS	Technical Specifications
UAT	Unit Auxiliary Transformer
USAR	Updated Safety Analysis Report