June 16, 2003

Mr. Lew W. Myers
Chief Operating Officer
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 50-346/03-13

Dear Mr. Myers:

On May 17, 2003, 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 May 16, 2003, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. For the entire inspection period, the Davis-Besse Nuclear Power Station was under the Inspection Manual Chapter (IMC) 0350 Process. The Davis-Besse Oversight Panel assessed inspection findings and other performance data to determine the required level and focus of followup inspection activities and any other appropriate regulatory actions. Even though the Reactor Oversight Process had been suspended at the Davis-Besse Nuclear Power Station, it was used as guidance for inspection activities and to assess findings.

Based on the results of this inspection, the inspectors identified three findings of very low safety significance (Green) that were determined to involve violations of NRC requirements. Because the findings had very low safety significance and were entered into your corrective action program, the NRC is treating the findings as Non-Cited Violations in accordance with Section VI.A of the NRC Enforcement Policy. If you contest any of the Non-Cited Violations in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region III, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at Davis-Besse.

Since the terrorist attacks on September 11, 2001, the NRC has issued two Orders (dated February 25, 2002, and January 7, 2003) and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the February 25th Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) ‘02, and the remaining inspections are scheduled for completion in CY ‘03. Additionally, tabletop security drills were conducted at several licensees to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigation strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY ‘03, the NRC will continue to monitor overall safeguards and security controls, conduct inspections, and resume force-on-force exercises at selected power plants. Should threat conditions change, the NRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial power reactors.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html(the Public Electronic Reading Room).

Sincerely,

/RA by Dave Passehl Acting for/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

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

Enclosure: Inspection Report 50-346/03-013

See Attached Distribution

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/03-013
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, 2003, through May 17, 2003
Inspectors:	S. Thomas, Senior Resident Inspector D. Simpkins, Senior Resident Inspector (Hatch) R. Powell, Senior Resident Inspector (Perry) J. Ellegood, Resident Inspector (Perry) T. Tongue, Project Engineer
Approved by:	Christine A. Lipa, Chief Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000346/2003-013; 4/1/2003 - 5/17/2003; Davis-Besse Nuclear Power Station; Post Maintenance Testing; Maintenance Effectiveness.

This report covered a 7-week period of inspection by resident and Region III inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 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 An NRC identified non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified for the failure to properly implement procedures required for performing equivalency evaluations for components being replaced in safety related equipment. This resulted in the installation of relays into the Safety Features Actuation System (SFAS) cabinets that were not electrically rated for their specific application.

  The inspectors concluded that the finding is more than minor because, if left uncorrected the finding would become a more significant safety concern. By procuring and installing relays into the SFAS cabinets that were not electrically rated for that particular application, if left uncorrected, there was no reasonable assurance that the SFAS would have actuated required safety-related components when called upon. This finding is of very low safety significance because no actual loss of a safety function occurred. Even though SFAS was placed in service (shutdown bypass switches taken out of bypass), the impact to plant risk was negligible due to the fact that, at no time when the incorrect relays were installed, was the SFAS required to be operable to support the operating Mode of the plant. (Section 1R12.b.3)

- Green A Green self-revealing non-cited violation of Technical Specification 6.8.1.a was identified for inadequate component restoration instructions contained in DB-SC-03122, “SFAS Component Testing Procedure,” Revision 01. This resulted in the inadvertent operation, on separate occasions, of Borated Water Storage Tank Outlet Valves DH7A and DH7B during Safety Feature Actuation System (SFAS) individual component testing restoration activities for Core Flooding Tank to Sampling System Valve CF1545 and Nitrogen System to Containment Isolation Valve NN236.

  The finding is more than minor because it could be viewed as a precursor to a more significant event. In other circumstances, the inadvertent opening of the valve could result in a condition adverse to safety, including flooding of the ECCS rooms. In this case, due to the plant configuration, there was no adverse impact on the plant. The finding is of very low safety significance because no actual loss of a safety function occurred. (Section 1R19.b.2)

- Green A self-revealing non-cited violation of Technical Specification 6.8.1.a was identified for the failure to properly implement work instructions during the reinstallation of electrical conduit and the electrical termination of operating power and indication power to Loop 1 Reactor Coolant System High Point Vent Valves RC4608A and RC4608B. This resulted in the electrical power for each valve being swapped.

  The finding is more than minor because it: (1) involved the configuration control attribute of the Mitigating System cornerstone; and (2) affected the cornerstone objective of ensuring the availability, and capability of systems that respond to initiating events to prevent undesirable consequences. (Section 1R12.b.2)

B. Licensee Identified Findings

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

The plant was shutdown on February 16, 2002 for a refueling outage. During scheduled inspections of the control rod drive mechanism nozzles, significant degradation of the reactor vessel head was discovered. As a direct result of the need to resolve many issues surrounding the Davis-Besse reactor vessel head degradation, NRC management decided to implement IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition With Performance Problems." The fuel was removed from the reactor on June 26, 2002, and the plant remained shut down. The plant entered operational Mode 6 on February 19, 2003 and fuel reload was completed on February 26, 2003. The plant entered operational Mode 5 on March 12, 2003. For the entire inspection period, the Davis-Besse Nuclear Power Station was under the IMC 0350 Process. As part of this Process, several additional team inspections continued. The subjects of these inspections included: Containment Health, System Health Assurance, Management and Human Performance, and Program Compliance. The status of these inspections will not be included as part of this inspection report, but upon completion, each will be documented in a separate inspection report which will be made publicly available on the NRC website.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

.1 Quarterly Inspections (71111.04Q)

a. Inspection Scope

The inspectors verified equipment alignment for accessible portions of the 480 VAC and high pressure injection train 2 systems and identified any discrepancies that impacted the function of system components and the associated increase in risk. The inspectors also verified that the licensee had properly identified and resolved any equipment alignment problems that would cause initiating events or impact the availability and functional capability of the mitigating system. Specific aspects of this inspection included reviewing plant procedures, drawings, and the Updated Safety Analysis Report (USAR), to determine the correct system lineup and evaluating any outstanding maintenance work requests on the system or any deficiencies that would affect the ability of the system to perform its function. A majority of the inspector's time was spent performing a walkdown inspection of the system. Key aspects of the walkdown inspection included:

- valves were correctly positioned and did not exhibit leakage that would impact their function;
- electrical power was available as required;
- major system components were correctly labeled, lubricated, cooled, ventilated, etc.;

- hangers and supports were correctly installed and functional;
- essential support systems were operational;
- ancillary equipment or debris did not interfere with system performance;
- tagging clearances were appropriate; and
- valves were locked as required by the licensee's locked valve program.

During the walkdown, the inspectors also observed the material condition of the equipment to verify that there were no significant conditions not already in the licensee's work control system.

b. Findings

No findings of significance were identified.

.2 Semiannual Inspection (71111.04S)

a. Inspection Scope

In addition to the scope of a quarterly inspection, the inspectors performed a complete walkdown of the makeup and purification system to verify equipment alignment and to independently assure the licensee boric acid corrosion control program had adequately identified and corrected discrepancies in the material condition of components within the system. This walkdown will be included in the NRC's decision to address Restart Checklist, Item 2.d, "Extent-of-Condition of Systems Outside of Containment."

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on the availability, accessibility, and condition of fire fighting equipment, the control of transient combustibles, and the condition and operating status of installed fire barriers. The inspectors selected fire areas for inspection based on their overall contribution to internal fire risk, as documented in the Individual Plant Examination of External Events, their potential to impact equipment which could initiate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed at the end of this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use, that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits, and that fire doors, dampers, and penetration seals appeared to be in satisfactory condition.

The following areas were inspected:

- evaluation of the acceptability of epoxy-based floor coatings, primarily in areas which contained safe shutdown equipment, and ensured that the coatings complied with the National Fire Protection Association's (NFPA's) Life Safety Code and did not exceed code requirements for flame spread characteristics for that particular coating material;
- Emergency Core Cooling System (ECCS) pump room 1;
- ECCS pump room 2;
- external fire structures, including outside transformers; and
- turbine deck.

b. Findings

No findings of significance were identified.

1R06 Flood Protection

.1 Evaluation of Underground Cable Insulation Integrity

a. Inspection Scope

The inspectors reviewed the licensee's flood protection program with emphasis on safety related underground cables and drainage of open areas within the protected area. The inspectors reviewed the licensee's program for ensuring the integrity of underground cable insulation while taking into account industry experiences with flooding related failures of underground cable. In addition, the inspectors performed an inspection of the outside area in the protected area to verify drainage paths for surface water run off were clear and visible drains were unobstructed.

b. Findings

No findings of significance were identified.

.2 Evaluation of Internal Flooding for ECCS Pump Rooms

a. Inspection Scope

During the week of March 31, 2003, the inspectors reviewed the ECCS rooms and adjacent areas to verify that the internal flooding vulnerabilities were accurately depicted in design basis documents and risk assessments; that licensee procedures were in place to address flooding; and that compensatory measures were established during maintenance activities which increased flooding potential. Specifically, the inspectors walked down the ECCS rooms, and adjacent areas to verify that the licensee had identified all sources that could flood the rooms and verified compensatory measures were established during service water system hydrolasing and flushing activities.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors reviewed the licensee's overall maintenance effectiveness for risk-significant mitigating systems. This evaluation consisted of the following specific activities:

- observing the conduct of planned and emergent maintenance activities when possible;
- reviewing selected Condition Reports (CRs), open Work Orders (WOs), and control room log entries in order to identify system deficiencies;
- reviewing licensee system monitoring and trend reports; and
- a partial walkdown of the selected system.

The inspectors also reviewed whether the licensee properly implemented the Maintenance Rule, 10 CFR 50.65, for the system. Specifically, the inspectors determined whether:

- the system was scoped in accordance with 10 CFR 50.65;
- performance problems constituted maintenance rule functional failures;
- the system had been assigned the proper safety significance classification;
- the system was properly classified as (a)(1) or (a)(2); and
- the goals and corrective actions for the system were appropriate.

The above aspects were evaluated using the maintenance rule program and other documents listed in the Attachment.

The inspectors reviewed the following components:

- reactor coolant pumps 1-1 and 1-2;
- reactor coolant system high point vent valves RC4608A and RC4608B; and
- safety features actuation system relays.

b. Findings

1. Reactor Coolant Pumps 1-1 and 1-2

No findings of significance were identified. The reactor coolant pump maintenance was further discussed in Section 4OA5.2 (Follow-up of Specific Oversight Panel Issues).

2. Reactor Coolant System High Point Vent Valves RC4608A and RC4608B

Introduction A Green self-revealing non-cited violation of Technical Specification (TS) 6.8.1.a was identified for failing to properly implement work instructions during the reinstallation of electrical conduit and the electrical termination of operating power and indication power to Loop 1 Reactor Coolant System High Point Vent Valves RC4608A and RC4608B. This resulted in the electrical power for each valve being swapped.

Description On September 23, 2002, as part of interference removal in the west D-ring to support loop 1 reactor coolant pump work, RC4608A and RC4608B were electrically disconnected and their associated electrical conduits were removed. On March 24, 2003, after the requisite reactor coolant pump maintenance had been completed, the electrical conduit for RC4608A and RC4608B was reinstalled and electrical connections to each valve were completed. On May 1, 2003, during the performance of DB-SP-03366, "Reactor Coolant System Vent Path Operability Test," Revision 02, RC4608B did not open when HIS 4608B (control room operating switch for RC4608B) was placed in the appropriate position. Troubleshooting conducted per WO 03-003076 determined that electrical cables had been erroneously swapped between SV4608A and SV4608B (the solenoid valves associated with RC4608A and RC4608B).

Analysis The finding was more than minor because it: (1) involved the configuration control attribute of the Mitigating System cornerstone; and (2) affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because no actual loss of a safety function occurred, due to the high point vents not being required to support the current plant operational Mode.

Enforcement The performance deficiency associated with this event is the failure to correctly implement procedures which direct maintenance activities which removed/installed electrical power to safety related equipment. Technical Specification 6.8.1.a requires implementation of procedures required by Regulatory Guide 1.33. Regulatory Guide 1.33 requires procedures for maintenance which can affect the performance of safety-related equipment. The licensee developed WO 02-004137-023 which provided guidance, in part, to disconnect/reconnect power to SV4608A and SV4608B. Contrary to the requirements of TS 6.8.1.a, WO 02-004137-023 was not implemented correctly which resulted in the electrical cables for SV4608A and SV4608B being erroneously swapped. As a result, RC4608B did not function when called upon during the performance of DB-SP-03366. Because of the very low safety significance and because the issue has been entered into the licensee's corrective action program (CR 03-03427 and CR 03-03351) it is being treated as a Non-Cited Violation, consistent with Section VI.A of the USNRC Enforcement Policy (NCV 50-346/03-13-01).

.3 Safety Features Actuation System (SFAS) Relay Replacement

Introduction:

A Green NRC identified non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified for failing to properly implement procedures required for performing equivalency evaluations for components being replaced in safety related equipment. This resulted in the installation of relays into the SFAS cabinets that were not electrically rated for that specific application.

Description:

In the Fall of 1999, the licensee developed plans to acquire replacement relays for their aging SFAS relays. The licensee planned to replace approximately 250 of the 286 relays during 13 RFO. Since the direct replacement for the relays was no longer manufactured, the licensee worked with the relay vendor to acquire a suitable replacement. This replacement relay type was validated by an independent testing lab, to specifications supplied by the licensee, and subsequently installed in the SFAS system. During the pre-installation testing, post installation testing, and return to service of the SFAS system, three of the new style relays failed. Extensive troubleshooting revealed that the design ratings for the new style relay, both in DC and AC applications, were significantly less than the relays that they were meant to replace, and in some applications, significantly less than what was required by the specific SFAS application.

The inspectors evaluated the licensee procurement activities for the new SFAS relays. The licensee procedure which governed this type of activity was EN-DP-01023, "Material Engineering Evaluation," Rev. 00. This procedure described requirements and responsibilities for performing part classification, equivalency evaluation of replacement items and commercial grade procurement and dedication for the Davis-Besse Nuclear Power Station. This procedure also outlined the responsibilities of two key positions in the procurement process.

- Procurement Engineering Supervisor shall be responsible for "reviewing the validity, accuracy, and completeness of the evaluation prepared by the responsible engineer."

- Procurement Engineering Engineer shall be responsible, in part, for "performing equivalency evaluation of replacement items." As part of the equivalency evaluation, the following were required to considered:

    - critical characteristics of original and replacement items shall be evaluated and the differences in characteristics shall be justified;
    - verify the item is satisfactory for the application in terms of form, fit, function, and manufacturing processes, and that there is no degradation to the seismic or environmental qualification of the parent component;
    - obtain missing data from the vendor and reconcile any differences in critical characteristics;

- document and evaluates any differences between critical characteristics; and
- complete the engineering evaluation to determine the acceptability of the replacement item, to include the ability of the parent component or item to perform its intended safety or operational function.

Key to this evaluation was the engineer's requirement to verify that the form (the physical characteristics, design ratings, quality requirements, code applicability, and regulatory requirements), fit (the mounting, attachment, and space that is occupied or required to support operation of an item), and function (the performance characteristics and range of operation of an item) of the new style relay was equal-to or better-than the original relay that it was replacing.

A review of the relay procurement documentation revealed that important information was missing;

- in the document where the relay is described; amperage, contact information, action, and insulation class was described as "NA";
- the standard paragraph (A34) on the procurement package data sheet only listed the part number of the relay and did not provide any amplifying information (on lines 35,36,37,38, or 40) as required by the procurement procedure (NS-MD-00070), which would have ensured component compatibility, form, fit, and function; and
- the equivalency evaluation stated that the new style relay was "the same fit and function as the originally supplied relay," but contained little or no technical justification to support that conclusion.

The licensee's Material Engineering Evaluation procedure clearly defined the expectations of what was required during the performance of an equivalency evaluation;

- "Equivalency Evaluation is defined as a technical evaluation to confirm that an equivalent replacement item, which is not identical to the original item, will satisfactorily perform its intended function once in service. The equivalence of an item can be demonstrated by evaluating fit, form and function considering the design values for the original item, and the comparison of changes in those design values from the original to the replacement item. In order to be an equivalent replacement item, the replacement item's parameters do not need to be identical to the original item, but they must be demonstrated to be equal-to or better-than the original item or it must be demonstrated that the parameters which are not equal-to or better-than the original are acceptable."

No information provided by the licensee documented that an adequate equivalency evaluation was performed during the procurement process for the new style relay or prior to installing the relays into the SFAS cabinets.

<u>Analysis</u>: The inspectors concluded that the finding was more than minor because, if left uncorrected the finding would become a more significant safety concern. By procuring and installing relays into the SFAS cabinets that were not electrically rated for

that particular application, if left uncorrected, there was no reasonable assurance that the SFAS would have actuated required safety-related components when called upon. This finding was of very low safety significance because no actual loss of a safety function occurred. In addition, the impact to plant risk was negligible because at no time when the incorrect relays were installed, was the SFAS system required to be operable to support the operating Mode of the plant.

Enforcement: The performance deficiency associated with this event is the failure to adequately implement procedures required for performing equivalency evaluations for components being replaced in safety related equipment. 10 CFR 50, Appendix B, Criterion V, states, in part, that activities affecting quality shall be prescribed in documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to this, the licensee did not adequately implement procedure NS-MD-01023, “Material Engineering Evaluation,” Revision 00, a procedure affecting quality, during the procurement efforts for the replacement SFAS relays. This resulted in the installation of relays into the SFAS cabinets that are not electrically rated for that specific application. Because of the very low safety significance and because the issue has been entered into the licensee’s corrective action program (CR 03-03232) it is being treated as a Non-Cited Violation, consistent with Section VI.A of the USNRC Enforcement Policy (NCV 50-346/03-13-02).

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (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 increasing overall plant risk. The inspection was conducted to verify the planning, control, and performance of the work were done in a manner to reduce overall plant risk and minimize the duration where practical, and that contingency plans were in place where appropriate. The licensee’s daily configuration risk assessments, observations of shift turnover meetings, observations of daily plant status meetings, and the documents listed at the end of this report were used by the inspectors to verify that the equipment configurations had been properly listed, that protected equipment had been identified and was being controlled where appropriate, and that significant aspects of plant risk were being communicated to the necessary personnel. The following risk significant issues were evaluated by the inspectors:

- D1 (4160 volt) safety related electrical bus outage
- Maintenance on Containment Air Cooler #1; and
- C1 (4160 volt) safety related electrical bus outage.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed Operability Evaluation 2002-0039 (emergency diesel generator maximum room temperature), Revision 1 which discussed potential operability issues for the emergency diesel generators. This operability evaluation was evaluated to determine whether the operability of the component was justified. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and USAR to the licensee's evaluations presented on the issues listed below to verify that the component was operable. Where compensatory measures were necessary to maintain operability, the inspectors verified by review of the documents listed at the end of the report that the measures were in place, would work as intended, and were properly controlled.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors reviewed all of the existing operator workarounds and control room deficiencies to determine whether the cumulative conditions had a significant impact on plant risk or on the operators' ability to respond to a transient or an accident. This involved reviewing the entire list of operator workarounds and control room deficiencies, interviewing operators and staff, and turnover sheets to verify that the licensee had appropriately classified them for significance, that the workarounds were achievable, and that the licensee had made or planned timely and appropriate corrective actions. In addition to evaluating the individual impact of each operator workaround, the inspector evaluated the cumulative affect of all workarounds on plant safety.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance testing activities to ensure that the testing adequately verified system operability and functional capability with consideration of the actual maintenance performed. The inspectors used the appropriate sections of the TSs and the Updated Safety Analysis Report (USAR), as well as the documents listed at the end of this report, to evaluate the scope of the maintenance and verify that the work control documents required sufficient post-maintenance testing to adequately demonstrate that the maintenance was successful and that operability was restored. In addition, the inspectors reviewed CRs to verify minor deficiencies identified during these inspections were entered into the licensee's corrective action system. The inspectors observed and evaluated test activities associated with the following:

- high pressure injection pump 1 Mode 5 baseline test in piggyback mode;
- high pressure injection pump 1 Mode 5 check valve flow test; and
- SFAS component testing of NN(nitrogen supply)236 and CF(core flood)1545.

b. Findings

1. High Pressure Injection Pump 1 Testing

No findings of significance were identified.

2. SFAS Component Testing of NN(nitrogen supply)236 and CF(core flood)1545

Introduction: A Green self-revealing non-cited violation of TS 6.8.1.a was identified for inadequate component restoration instructions contained in DB-SC-03122, "SFAS Component Testing Procedure," Revision 01. This resulted in the inadvertent operation, on separate occasions, of Borated Water Storage Tank Outlet Valves DH7A and DH7B during Safety Feature Actuation System (SFAS) individual component testing restoration activities for Core Flooding Tank to Sampling System Valve CF1545 and Nitrogen System to Containment Isolation Valve NN236, respectively. On each occasion, the control room operators immediately recognized the condition and promptly closed the valve.

Description: On March 31, 2003, following maintenance activities on valve NN236 per work order 03-001812-000, SFAS response time testing was performed. Since multiple SFAS components actuate from the same output module, DB-SC-03122 allowed bypassing or disabling components to prevent unnecessary component repositioning. In this instance, the breaker for DH7B was opened to prevent the valve from opening when the trip signals were provided to output module pair L281/L283. Following completion of the testing, system restoration included shutting the DH7B breaker. When the breaker was shut, the valve immediately began opening.

Licensee review of the event determined that DH7B had a locked in open signal following the SFAS testing. The condition was caused by a design change to the control power pickup point. Originally, the 480 VAC feeding the control power transformer was picked up downstream of the circuit breaker. This pickup was later moved, by a design change, to a point upstream of the circuit breaker. By leaving the control power energized with the valve motor de-energized, the seal-in condition that was created caused the valve to reposition immediately upon breaker closure. The licensee also determined that depressing the local closed pushbutton for DH7B, prior to closing the breaker, would defeat the open signal and prevent the undesired valve opening.

On April 1, 2003, a similar valve opening occurred during post maintenance activities for valve CF1545. DB-SC-03122 allowed bypassing or disabling components to prevent unnecessary component repositioning. In this instance, the breaker for DH7A was opened to prevent the valve from opening when the trip signals were provided to output module pair L282/L284. Following completion of the testing, system restoration included shutting the DH7A breaker. When the breaker was shut, the valve immediately began opening.

Licensee review of the event determined that DH7A had a locked in open signal caused by a similar circuit configuration, as previously discussed for DH7B.

The licensee entered both of these issues into their corrective action program and initiated corrective actions to further review motor-operated valve circuits actuated by an SFAS signal to determine if other motor-operated valves were impacted by the seal-in condition and to modify procedure DB-SC-03122 to provide appropriate guidance to break the seal-in condition prior to closing the circuit breaker for affected motor-operated valves.

Analysis: The inspectors concluded that the finding was more than minor because it could be reasonably viewed as a precursor to a significant event. In other circumstances, the inadvertent opening of the valve could result in a condition adverse to safety, including flooding of the ECCS rooms. In this case, due to the plant configuration, there was no adverse impact on the plant. The finding is of very low safety significance because no actual loss of a safety function occurred.

Enforcement: The performance deficiency associated with this event was the failure to develop adequate procedures for testing SFAS components. TS 6.8.1.a requires implementation of procedures required by Regulatory Guide 1.33. Regulatory Guide 1.33 requires procedures for tests, inspections, and calibrations of the emergency core cooling system. The licensee developed DB-SC-03122 for testing of SFAS components. Contrary to TS 6.8.1.a requirements, this testing procedure was not adequate in that system restoration instructions were incomplete. As a result, on different occasions, DH7A and DH7B began to inadvertently open due to locked in open signals during the testing restoration process. Because of the very low safety significance and because the issue has been entered into the licensee's corrective action program (CR 03-02554 and CR 03-2571) it is being treated as a Non-Cited Violation, consistent with Section VI.A of the USNRC Enforcement Policy (NCV 50-346/03-13-03).

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed the following surveillance test and evaluated test data to verify that the equipment tested met TSs, USAR, and licensee procedural requirements, and also demonstrated that the equipment was capable of performing its intended safety functions. The activity was selected based on its importance in verifying mitigating system capability. The inspectors used the documents listed at the end of this report to verify that the test met the TS frequency requirements; that the test was conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria were met; and that the results of the test were properly reviewed and recorded.

The following test was observed and evaluated:

- DB-SC-03070, Emergency Diesel Generator 1 Monthly Test

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

EP6 Drill Evaluation (71114.06)

a. Inspection Scope:

The inspectors monitored the licensee's emergency preparedness exercises conducted on April 10, 2003, and May 13, 2003, from various locations and perspectives. The observations included licensee preparations, evaluation of drill conduct, adequacy of the drill critiques, and identification of weaknesses and deficiencies. The inspector reviewed the licensee's scenario, preparation, and controller instructions and debriefing. The inspectors noted the reality of the drill scenario and observed personnel performance in the simulator control room, the technical support center, and the emergency control center. The inspectors also noted the communications, accuracy of situation evaluations, and reporting (simulated) to appropriate agencies. Finally, the inspectors observed the licensee's player and controller critiques to assure that weaknesses and deficiencies were acknowledged and appropriateness of corrective actions identified.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA5 Other Activities

One of the key building blocks in the licensee's Return to Service Plan was the Management and Human Performance Excellence Plan. The purpose of this plan was to address the fact that "management ineffectively implemented processes, and thus failed to detect and address plant problems as opportunities arose." The primary management contributors to this failure were grouped into the following areas:

- Nuclear Safety Culture;
- Management/Personnel Development;
- Standards and Decision-Making;
- Oversight and Assessments;
- Program/Corrective Action/Procedure Compliance.

The inspectors had the opportunity to observe the day-to-day implementation that the licensee made toward completing Return to Service Plan activities. Almost every inspection activity performed by the resident inspectors touched upon one of those five areas. Observations made by the resident inspectors were routinely discussed with the Davis-Besse Oversight Panel members and were used, in part, to gauge licensee efforts to improve their performance in these areas on a day-to-day basis.

To better facilitate the inspection and documentation of issues not specifically covered by existing inspection procedures, but important to the evaluation of the licensee's readiness for restart, the Special Inspection for Residents inspection plan was developed and implemented. Inspection Procedure 93812, "Special Inspection," was used as a guideline to document these issues and remains in effect for future resident inspection reports until a time to be determined by the Davis-Besse Oversight Panel. The inspectors performed inspections, as required, to adequately assess licensee performance and readiness for restart in the following area:

- performance of plant activities, including maintenance activities;
- follow-up of specific Oversight Panel Technical issues;
- attended and assessed selected licensee restart readiness meetings;
- evaluated licensee performance in categorizing, classifying, and correcting deficient plant conditions during the restart process;
- reviewed licensee controls, criteria, and assessed licensee performance at meetings associated with work backlogs, including the deferral of work orders, operator work arounds, temporary modifications, and permanent modifications; and
- reviewed activities associated with safety conscious work environment and safety culture.

The following issues were evaluated during this inspection period.

.1 Performance of Plant Activities

During the first week in May, the licensee completed reactor coolant system fill and venting evolutions and established a nitrogen bubble in the pressurizer. This nitrogen bubble was controlled by operations to maintain a reactor coolant system pressure of approximately 45 psig. This established conditions which facilitated the first opportunity to check the integrity of several reactor plant components that were worked on during the current extended outage. The walkdowns revealed several leakage issues, but were primarily limited to valve packing glands that stopped leaking once the packing had been rewetted. The inspectors received several debriefs from the licensee, and had detailed discussions regarding approximately 30 of issues. The inspectors verified that the individual issues were documented in the licensee corrective action program. The licensee plans to perform similar system walkdowns during pressure tests at 250 psig and 2155 psig.

No significant issues were identified during this inspection.

.2 Follow-Up of Specific Oversight Panel Technical Issues

a. Reactor Coolant Pump Maintenance

During RFO 13, the licensee refurbished reactor coolant pumps 1-1 and 1-2. The driving force behind the refurbishment of both pumps at that time was the need to replace the inner and outer case-cover gaskets, as evidenced by coolant leakage past the outer gasket. Since the replacement of the gaskets required complete disassembly of each pump, in addition to replacing the inner and outer gaskets on each pump, the licensee chose to replace both pump motors, inspect the pump covers for cracking, refurbish or replace the rotating elements, and replace the pump seals. This work has been completed. Pump testing will be completed when reactor coolant system conditions permit.

The licensee decided not to perform similar maintenance of reactor coolant pumps 2-1 and 2-2 during the current outage. The corrective action team inspection is reviewing the licensee's decision in regard to these pumps and will discuss it in Inspection Report 2003-010.

.3 Classification, Categorization, and Resolution of Restart Related Issues

The resident inspectors continued to monitor the licensee activity related to classifying, categorizing and resolving the backlog of work orders, corrective actions, and modifications required to be completed prior to transitioning to Mode 4. To accomplish this, the inspectors:

- attended and assessed licensee management meetings;
- monitored the management of open Mode 4 and 3 restraints;
- evaluated the licensee classification of emergent deficient conditions; and
- evaluated closed mode restraints.

As part of this inspection, the inspectors attended selected Mode Change Readiness Review meetings, Senior Management Team meetings, Management Review Board meetings, and Restart Station Review Board meetings where classification of condition reports, prioritization of work activities, and setting of work completion dates took place.

The inspectors evaluated a sampling of condition report downgrade documentation forms associated with the (Significant Conditions Adverse to Quality (SCAQs) that were downgraded to Conditions Adverse to Quality (CAQs), a sampling of completed mode resolution forms, and a sampling of items that were reclassified as Mode 2 (previously classified as Mode 4 or 3 items) items.

No significant issues were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Lew Myers, and other members of licensee management on May 16, 2003. The licensee acknowledged the findings presented. No proprietary information was identified.

ATTACHMENT: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Bezilla, Site Vice President
G. Dunn, Outage Manager
R. Fast, Director, Organizational Development
J. Grabnar, Manager, Design Engineering
D. Imlay, Superintendent Electrical Maintenance
M. Marler, Manager, Nuclear Training
P. McCloskey, Manager, Regulatory Affairs
G. Melssen, Maintenance Rule Coordinator
L. Myers, Chief Operating Officer, FENOC
W. Mugge, Manager, Nuclear Security
R. Farrel, Manager, Chemistry and Radiation Protection
J. Powers, Director, Nuclear Engineering
R. Rishel, PRA Specialist
M. Roder, Manager, Plant Operations
R. Schrauder, Director Support Services
A. Schumaker, Supervisor Access Control (Acting)
A. Stallard, Operations Support Supervisor
M. Stevens, Director, Maintenance
J. Vetter, Quality Assurance Supervisor
G. Wolf, Senior Licensing Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-346/03-13-01	NCV	Failure to Properly Implement Work Instructions During the Reinstallation of Electrical Conduit and the Electrical Termination of Operating Power and Indication Power to RC4608A and RC4608B (loop 1 reactor coolant system high point vent valves)
50-346/03-13-02	NCV	Inadequately Implementation procedure NS-MD-01023 (Material Engineering Evaluation) During the Procurement Efforts for Replacement SFAS elays
50-346/03-13-03	NCV	Inadvertent Operation of DH7A and DH7B Caused By Inadequate SFAS Component Testing Procedure

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

Operational Schematic OS 59, 480 Volt System, Sheet 1, Rev. 4; Sheet 2, Rev. 2; Sheet 3, Rev. 7; Sheet 4, Rev. 10; and Sheet 5, Rev. 3

Operational Schematic OS-003, High Pressure Injection System, Rev. 19

Operational Schematic OS-002, Makeup and Purification System, Sheet 1, Rev. 18; Sheet 2, Rev. 16; Sheet 3, Rev. 26; and Sheet 4, Rev. 13

1R05 Fire Protection

Fire Hazards Analysis Report

Ratings on Keller and Long Coating Systems, July 20, 1992

Specification Number A-024N, Rev. 8

Specification Number A-024N; Specification For Operational Phase For Field Painting Outside Containment; Rev. 7

Fire Protection Schematic A230F, Rev. 9; A224F, Rev. 17; A225F, Rev. 12; and A226F, Rev. 10

1R06 Flood Protection

CR 03-01156, IN 2002-12- Submerged Safety-Related Electrical Cables, February 12, 2003

CR 02-08185, De-energized Motor and Cable Testing, October 15, 2002

CR 01-0489, Degraded Cable Splice in Manhole (MH 3027), February 16, 2001

USAR 8.3.1.2.8, AC Electrical Systems

EMPAC Database, April 3, 2003

Contingency Plan 13RFO-22; Work in the ECCS Rooms, Including Service Water Hydrolasing and Flushing Activities, With RCS Level Greater Than or Equal to 80 Inches; Rev. 0

RA-EP-02880; Internal Flooding; Rev. 2

Identification of Flood Initiating Events for the Davis-Besse Individual Plant Examination; dated May 1992

Davis-Besse Station Flooding Program Collective Significance Review; dated February 18, 2003

CR 03-02391; Issues Concerning the ECCS 1 Sump Overfill Event of 3/25/2003; dated March 25, 2003

1R12 Maintenance Effectiveness

CR 03-03427; RC4608A and RC4608B Are Not Wired Properly

CR 03-03351; RC4608B Failed to Open During DB-SP-0036

WO 03-003076-00; Trouble Shoot RC4608B Failure to Open During DB-SP-0036

WO 02-004137-23; Determ/Reterm Electrical connections to SV4608A and SV4608B as Part of Interference Removal For Loop 1 Reactor Coolant Pump Work

DB-SP-03366; Reactor Coolant System Vent Path Operability; Rev. 02

Electrical Drawing E52B, Sheets 71A and 71B

Procurement Package 9033821 (SFAS Relay Procurement); Revision 00

NS-MD-01023; Material Engineering Evaluation; Revision 00

Procurement Package 86656506 (SFAS Relay Procurement); Revision 04

CR 02-03083; Wyle Labs Failure Analysis Report

CR 03-03232; Inadequate Approval of Replacement SFAS Output Relays: Deutsch 4CP36AF

CR 03-02725; New Style SFAS Output Relay Potential Design Deficiency

CR 03-01679; Confirmed Failure of a Second SFAS Output (Deutsch) Relay From the New Batch

CR 01-3470; Relay Failed Bench Test

NS-MD-00070; Procurement; Revision 03

NPE-03-00047; Reactor Coolant Pump Status to August 9, 2002 White Paper; Revision 01

NPE-02-00227; Reactor Coolant Pump Issues

FLOWSERVE Letter; Subject: Inspection of Davis-Besse Reactor Coolant Pumps 2-1 and 2-2; Dated September 16, 2002

FLOWSERVE Letter; Subject: Return to Service of Davis-Besse Reactor Coolant Pumps 2-1 and 2-2; Dated February 4, 2003

1R13 Maintenance Risk and Emergent Work

Contingency Plan for D1 Bus Outage With RCS Loops Filled; Number 13 RFO-27; Rev. 0

13 RFO Weekly Risk Summary for the Week of May 5, 2003; Rev. 0

Contingency Plan for C1 Bus Outage With RCS Loops Filled; Number 13 RFO-28; Rev. 0

13 RFO Weekly Risk Summary for the Week of May 19, 2003; Rev. 0

1R15 Operability Evaluations

Operability Evaluation 2002-0039; Emergency Diesel Room Temperature; Rev. 01

1R16 Operator Work Arounds

WPG-2 R05, Work Process Guideline -2, Operations Equipment Issues, dated February 21, 2003

1R19 Post-Maintenance Testing

DB-PF-03407; HPI Pump 1 Mode 5 Baseline Test in Piggyback Mode; Revision 00

DB-PF-03307; HPI Pump 1 Mode 5 Check Valve Flow Test; Revision 00

DB-SC-03122; SFAS Component Tests; Rev. 1

Drawing E52B, Sheet 19A; Reactor Cooling System BWST Outlet Vlv; Rev. 16

CR 03-02554; DH7B Opened Unexpectantly; dated March 31, 2003

CR 03-02571; DH7A Opened After Testing Was Complete

DB-SC-03122; SFAS Component Tests; Revision 01

Davis-Besse Elementary Wiring Diagram E52B, Sheet 19A; Reactor Cooling System BWST Outlet Valve; Revision 16

1R22 Surveillance Testing

Surveillance Test Procedure DB-SC-03070 Emergency Diesel Generator 1 Monthly Test, Rev. 4

Operations Directives GP-03 Conduct of Pre-job Briefs and Post Job Reviews, Rev. 04

EP6 Drill Evaluation

Emergency Plan Drill and Exercise Program (Portions), RA-EP-00200, Rev. 03

Emergency Preparedness Dry Run Manual, May 13, 2003

Emergency Plan Implementing Procedure, RA-EP-02720, Rev. 04

LIST OF ACRONYMS USED

ADAMS	Agency-wide Document Access and Management System
CAQ	Condition Adverse to Quality
CFR	Code of Federal Regulations
CR	Condition Report
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FENOC	FirstEnergy Nuclear Operating Company
ICM	Interim Compensatory Measures
IMC	Inspection Manual Chapter
IR	Inspection Report
LER	Licensee Event Report
NFPA	National Fire Protection Association
NRC	United States Nuclear Regulatory Commission
PARS	Publicly Available Records
RFO	Refueling Outage
SCAQ	Significant Condition Adverse to Quality
SFAS	Safety Features Actuation System
SDP	Significance Determination Process
SRB	Station Review Board
TS	Technical Specifications
URI	Unresolved Item
USAR	Updated Safety Analysis Report
WO	Work Order