March 17, 2006

Mr. Mark B. Bezilla
Vice President-Nuclear, Davis-Besse
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 EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000346/2006006(DRS)

Dear Mr. Bezilla:

On February 10, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Davis-Besse Nuclear Power Station. The enclosed report documents the results of the inspection, which were discussed with Mr. B. Allen, and others of your staff at the completion of the inspection on February 10, 2006 and March 2, 2006.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of the inspection, three NRC identified findings of very low safety significance were identified, which involved violations of NRC requirements. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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,

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000346/2006006(DRS)

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
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:	05000346/2006006(DRS)
Licensee:	First Energy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Plant
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	January 23, 2006, through February 10, 2006, and March 2, 2006
Inspectors:	R. Daley, Senior Reactor Inspector, Team Leader A. Dahbur, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000346/2006006(DRS); 01/23/2006 - 02/10/2006 and 03/02/2006; Davis-Besse Nuclear Power Station; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a two week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by two regional based engineering inspectors. Three Green Non-Cited Violations (NCV) and one Unresolved Item were identified. 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. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance. Specifically, the flow acceptance curves used for two safety-related pumps, the number 2 High Pressure Injection (HPI) pump and the number 2 Low Pressure Injection (LPI) pump, were incorrect and non-conservative. This issue was entered into the licensee's corrective action system and the licensee verified that other safety-related pumps were not similarity affected.

The issue was more than minor because it was associated with the Mitigating System cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, performing required flow testing of safety related pumps with non-conservative acceptance criteria could allow the pumps to operate during an accident with less than acceptable flows. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the failure to apply conservative test acceptance criteria during flow tests for the HPI and LPI pumps could have eventually caused the pumps to be outside of their acceptable flow rates without it being detected, the licensee verified that the most recent test data for these pumps would have still been acceptable if the correct pump curves had been used. (Section 1R02.1.b.1)

Green. The inspections identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance, where the licensee had not evaluated and updated the plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher temperatures in the Diesel Fire Pump Room and other areas in the plant. The issue was entered into the licensee's corrective action program.

The finding was more than minor because it affected the mitigating system cornerstone attribute of “Design Control” and affected the cornerstone objective of ensuring reliability of systems that respond to initialing events to prevent undesirable consequences. Specifically, the licensee did not account for high temperature conditions that adversely affected the ampacity of cabling supplying power to equipment important to safety. This finding was of very low safety significance, because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the licensee was able to demonstrate that even though the higher temperatures decreased the ampacity margins for the effected cabling, it did not decrease the margins to the limit where the cabling would fail if called upon to provide power to equipment important to safety. (Section 1R17.1.b.1)

Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” that was of very low safety significance. Specifically, after performing modifications to implement a new Boron Precipitation Control method for post-LOCA (Loss of Coolant Accident) operations, the licensee failed to both identify and establish testing for the flow instrumentation that the operators would use post-LOCA to ensure minimum flow for proper boron dilution. This issue was entered into the licensee’s corrective action system.

The issue was more than minor because if it was left uncorrected, the finding would become a more significant safety concern. Had this issue not been detected, the instrument could have fallen well out of tolerance in the future leading to inaccurate readings. During post-LOCA operations, these inaccurate readings could have caused operators to establish insufficient Boron Precipitation Control (BPC) flows. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the lack of calibration procedures for the flow instrumentation would have eventually caused the instrument to read inaccurately, the inspection team identified the problem early enough so that the instrument drift (as provided in the vendor instructions) would not be large enough to cause inaccuracies that would adversely affect BPC flows. (Section 1R17.1.b.2)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From January 23 through February 10, 2006, the inspectors reviewed four evaluations performed pursuant to 10 CFR 50.59. The inspectors confirmed that the evaluations were thorough and that prior NRC approval was obtained as appropriate. The team could not review the minimum sample size of five evaluations, because the licensee only performed four evaluations during the biennial sample period. The inspectors also reviewed 13 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

b. Findings

b.1 Non-Conservative Flow Testing Acceptance Criteria for the Number 2 HPI and Number 2 LPI Pumps

Introduction: The inspectors identified a NCV having a very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, safety-related pump flow curves used for test acceptance criteria, which were created in the design basis calculation for the number 2 HPI pump and the number 2 LPI pump, were not correctly translated into the plant procedure used for testing. The acceptance curves used for the pumps, were incorrect and non-conservative.

Description: At the Davis-Besse Nuclear Power Station, flow testing of the HPI and the LPI Pumps were performed quarterly. To ensure proper operation of the pumps, flow and pressure readings taken during the quarterly test were compared to curves from design basis calculations. These curves established the acceptance criteria for the quarterly flow tests.

During the inspection, the inspector identified that the acceptance curves used for two of the pumps, the number 2 HPI pump and the number 2 LPI pump, were incorrect. In the case of the number 2 HPI pump, the licensee's procedure contained a degraded pump curve instead of the correct curve. In the case of the number 2 LPI pump, the procedure contained a curve for the number 2 LPI pump that was really the curve for the number 1 LPI pump. In both instances, the curves in the test procedures contained non-conservative acceptance criteria.

Because of this non-conservative acceptance pump testing criteria, the licensee initiated Condition Report (CR) 06-00219. The licensee subsequently agreed that the correct pump curves, which were created in the design basis calculation for each pump, were not correctly translated into the plant procedure containing the acceptance criteria. Additionally, the licensee reviewed the most recent test data for these pumps and compared the results to the correct (conservative) acceptance criteria for the pumps. This review determined that even though the curves that were used for the test were non-conservative, the pumps would have still tested satisfactory using the correct pump curves.

In addition, since two pumps were found to have non-conservative acceptance values, the licensee, as an immediate corrective action for CR 06-00219, reviewed other safety related pump procedural quarterly acceptance criteria to ensure their accuracy. The review did not identify any other non-conservative acceptance criteria.

<u>Analysis:</u> The inspectors determined that this failure to translate the correct acceptance criteria from the design basis calculation to the testing procedure for the number 2 LPI and number 2 HPI pumps was a performance deficiency warranting a significance determination. The issue was more than minor because it was associated with the Mitigating System cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, performing required flow testing of safety-related pumps with non-conservative acceptance criteria could allow the pumps to operate during an accident with less than acceptable flows.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the failure to apply conservative test acceptance criteria during flow tests for the HPI and LPI pumps could have eventually caused the pumps to be outside of their acceptable flow rates without it being detected, the licensee verified that the most recent test data for these pumps would have still been acceptable if the correct pump curves had been used.

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, sometime prior to 2004 at the Davis-Besse Nuclear Power Station, the correct design basis pump curves for the number 2 HPI pump and the number 2 LPI pump were not correctly translated as acceptance criteria into the testing procedure. Because this

failure to apply appropriate test control measures was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CR 06-00219, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 06000346/2006006-01 (DRS))

b.2 Change to Design Basis Tornado Differential Pressure Design Limit for the Auxiliary Building

Introduction: The inspectors identified an Unresolved Item (URI) involving the adequacy of a 10 CFR 50.59 safety evaluation for Updated Safety Analysis Report (USAR) changes that the licensee had implemented. Specifically, the inspectors questioned the adequacy of the licensee's basis for determining that changes to USAR Sections 3.3.2, "Tornado Criteria" and 3.8.1.1.1, "Auxiliary Building" did not require a license amendment. This issue is unresolved pending further NRC review of Davis-Besse's licensing basis for tornado protection.

Description: Davis-Besse USAR Section 3.3.2.2.1 previously referenced DEPRELLIN as the analytical method/software used to evaluate the differential pressure between the compartments for the Auxiliary Building. This section indicated that by using the DEPRELLIN computer program, the maximum differential pressure between compartments in the Auxiliary Building was 0.5 psi. Condition Report (CR) 03-01132 identified a concern regarding the tornado differential pressure analysis of the auxiliary building, in that the DEPRELLIN methodology did not consider closed doors and hatches, and modeling of these closed barriers could impact the calculated differential pressure. Due to limitation of the DEPRELLIN software, the licensee chose to use the COMPARE computer code as an alternate method to DEPRELLIN to re-determine the maximum tornado depressurization loading. The licensee evaluated the use of the COMPARE computer code in 10 CFR 50.59 Evaluation Number 04-02210 "UCN 03-058 Tornado Subcompartment Pressurization Methodology Change Permitting COMPARE Methodology." The inspectors reviewed the screening/evaluation written on April 5, 2005, and questioned the changes that were made to the USAR per USAR Change Notice (UCN) 03-058. The inspectors were concerned about the adequacy of the licensee's basis, as stated in the 10 CFR 50.59 screening and evaluation, for determining that changes to USAR Sections 3.3.2 "Tornado Criteria" and 3.8.1.1.1 "Auxiliary Building" did not require a licensee amendment.

Prior to the implementation of UCN 03-058, USAR Section 3.3.2 stated, in part, that the structures for the Auxiliary Building were analyzed for tornado loading using an assumed value of 1.5 psi positive differential pressure between the inside and outside with the provision of venting the structure in order to control the differential pressure to within the 1.5 psi limit. Section 3.8.1.1.1 also stated:

> "The Auxiliary Building has been designed with sufficient venting area in order to keep the differential pressure drop within the 1.5 psi design limit. The external walls and the roof slabs are designed to withstand the forces due to a tornado, as described in Section 3.5.1(1). This 1.9×10^{6} cubic foot building has natural venting area on roof slabs, in the external walls, internal walls, and internal slabs. Each floor has an adequate opening or a number of openings to allow the air flow during tornado depressurization. The pressure drop is assumed to be 3.0 psi in 3 seconds, which is quite conservative."

Using the computer program COMPARE, the licensee re-calculated that the maximum deferential pressure between subcompartments in the auxiliary building was 3.0 psi. The licensee re-evaluated the Auxiliary Building for the revised differential pressure loads due to tornado depressurization. This evaluation was documented in Calculation C-CSS-099.20-029. The calculation concluded that the auxiliary building structure is adequate for these increased loads. Based on this, the licensee revised USAR Section 3.3.2 per UCN 03-058, and indicated that the structures for the Auxiliary Building subcompartments were analyzed for tornado loading using an assumed value of 3.0 psi maximum. The UCN also deleted the above discussion from USAR Section 3.8.1.1.1. The licensee justified the revision of these two sections in 10 CFR 50.59 Screen Number 04-02210, Revision 1, which indicated that these sections incorrectly identified the original tornado depressurization load of 1.5 psi as a design limit.

The inspectors were concerned that the licensee might have decreased a margin of a design limit (1.5 psi with venting capability) that was a basis for NRC staff acceptance of the original auxiliary building design for tornados. Additionally, the licensee's new calculation for the auxiliary building was a change in methodology and was not evaluated under 10 CFR 50.59 as required.

Following identification of this issue, the licensee entered the issue into their corrective action program as Condition Reports (CRs) 06-00246 and 06-00472. This issue is unresolved pending further NRC review of Davis-Besse's licensing basis for tornado protection to determine if the licensee had decreased design margin that was used by the NRC staff as a basis for acceptance of the licensee's safety analysis for a tornado event. (URI 05000346/2006006-02(DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From January 23 through February 10, 2006, the inspectors reviewed eight permanent plant modifications that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, one modification was chosen that affected the barrier integrity cornerstone. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements and the licensing bases and to confirm that the changes did not adversely affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

b.1 Failure to Consider Adverse Ampacity Effects of High Temperature Conditions in the Diesel Fire Pump Room

Introduction: The inspectors identified a NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B Criterion III, "Design Control." Specifically, the licensee had not evaluated and updated the plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher ambient temperatures in the Diesel Fire Pump Room.

Description: The licensee revised calculation 67.005, "Service Water Pump Room Ventilation System Capacity," which increased the potential maximum ambient temperature in the Diesel Fire Pump room from 104 degrees F to 120 degrees F. The potential high ambient temperatures could occur when the outside temperature is at the extreme limit of 104 degrees F and the Diesel Fire Pump engine is running.

The licensee used 10 CFR 50.59 Screening Number 04-03226, Revision 1, to evaluate the effects of high ambient temperature on the diesel fire pump engine and the safety-related equipment (i.e., Motor Control Center (MCC) E12C, MCC E12D and small circuit breakers) located in the Diesel Fire Pump Room. The evaluation concluded that the operability of the diesel fire pump engine and the safety-related equipment were acceptable for an ambient room temperature of 120 degrees F. However, the licensee failed to address the effects of these heightened temperatures on the ampacity of electrical cables in the room.

During the inspection, the team identified that Davis-Besse Electrical Ampacity Calculation C-EE-015.07-002 assumed ambient temperatures of 110 degrees F for cables located in the Diesel Fire Pump room. This was clearly non-conservative for this room. Since higher temperatures adversely affect the ampacity of electrical cables, the higher temperatures in the Diesel Fire Pump room had the potential to adversely affect the functionality and/or operability of equipment important to safety fed by cabling in the room. Specifically, the possibility existed that some of the equipment that were fed by cables located in the area may not function due to possible faulting of the supply cables.

The inspectors also reviewed Condition Report Number 1998-0001, which identified that ambient temperature in several areas in the plant (i.e., Diesel Generator Rooms, Low Voltage Switchgear/Battery Rooms, Electrical Isolation Rooms, Component Cooling Water Room and Auxiliary Feed Water Pump Rooms) could exceed the rooms' design temperatures when the outside temperature is above 95 degrees F. The inspectors noted that the licensee did not address the effects of these heightened temperatures on the electrical cables in these areas. Since the licensee's bounding ampacity calculations only evaluated cable ampacity to 110 degrees F, the inspectors were concerned that the possibility existed that some of the equipment that were fed by cables located in these areas may not function due to possible faulting of the supply cables or due to inadequate voltage caused by higher voltage drop. As a result of the inspectors' concerns, the licensee issued CR 06-00327.

After performing a preliminary evaluation that assessed cabling in Diesel Fire Pump room and the above listed areas, the licensee determined that there was no evidence that safety related Structures, Systems, and Components (SSCs) would not function as required. While the higher temperatures decreased the ampacity margins for the effected cabling, the licensee preliminarily determined that it did not decrease the margins to the limit where the cabling would fail if called upon to provide power to equipment important to safety.

Analysis: The inspectors determined that this issue was a performance deficiency warranting a significant determination. The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not account for high temperature conditions in the Diesel Fire Pump Room and other several rooms (i.e., Low Voltage Switchgear, Electrical Isolation, Component Water Pump, and Auxiliary Feed Water Pump Rooms) that adversely affected cables supplying power to equipment important to safety.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. In particular, the licensee's preliminary evaluation determined that the higher temperatures would not prevent pertinent equipment from functioning.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee did not have a design basis calculation for cable ampacity that supported the high temperatures that the Diesel Fire Pump Room and other plant's areas could experience. Hence, an adequate design basis was not translated into applicable documents. The Davis-Besse calculation that did address ampacity was significantly less conservative, since temperatures 110 degrees F were assumed where temperatures in these areas could exceed 120 degrees. This condition was known to have existed since 1999.

Because the failure to address the adverse effects of heightened temperatures on cables located in these rooms was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CR 06-00327, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000346/2006006-03 (DRS))

b.2 Failure to Provide Testing for Boron Precipitation Control Flow Instrumentation

Introduction: Specifically, after performing modifications to implement a new Boron Precipitation Control method for post-LOCA operations, the licensee failed to implement adequate test controls. The inspectors identified a NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion XI. The licensee failed to both identify and establish testing for the flow instrumentation that the operators would use post-LOCA to ensure minimum flow for proper boron dilution.

Description: The licensee performed Engineering Change Packages (ECP) 03-0146-00 and 03-0146-01 to implement a new Boron Precipitation Control method. Boron Precipitation Control ensures that flow is being provided through the core to dilute borated water that has been concentrated by evaporation during post-LOCA operations. The new method required that the LPI pump discharge flow to the reactor vessel via the decay heat drop line. Included in the design change was the refurbishment of an abandoned flow element, FE4909, that would be used for flow indication for the primary BPC flow. During the post-modification test for this new BPC method, the licensee performed a calibration on flow transmitter FT4909 and a flow test to ensure that the minimum BPC flow could be achieved. Davis-Besse operating procedures used for post-accident operations contained steps that measure the BPC flow to ensure that it is greater than the minimum required to establish adequate flow for proper boron dilution. The revised procedures used this flow element to verify the minimum BPC flow.

Because of the importance of this flow element, the inspectors requested that the licensee provide the inspection team with the Preventive Maintenance (PM) task that should have been established to ensure that FT4909 was always in calibration. The licensee was unable to find an existing PM or a PM in the planning stages. Because of this, the licensee issued CR 06-00217. Subsequent investigation determined that FT4909 was last calibrated in September 2003 when the post-modification flow testing was conducted. While FT4909 had not been calibrated for over 28 months, the vendor manual for the instrument established a +/- 2 percent drift for a 30 month period, so it was reasonable to conclude that the flow transmitter would still have been accurate for use in its design basis function.

However, the team noted that had this issue not been detected, the instrument could have fallen well out of tolerance in the future, leading to inaccurate readings. Because of the importance of FT4909 for establishing BPC post-LOCA, inaccurate readings could have led to the establishment of insufficient BPC flows. As an immediate corrective action for CR 06-00217, the licensee intended to develop a PM to calibrate FT4909 with a 24 month periodicity.

Analysis: The inspectors determined that this failure to identify and establish testing that ensured satisfactory operation of the boron precipitation control function was a performance deficiency warranting a significance determination. Specifically, after implementing a modification that established a new method for boron precipitation control, the licensee failed to identify and establish calibration testing for the flow instrumentation that would be used post-accident to ensure sufficient boron dilution flow.

The issue was more than minor because if it was left uncorrected, the finding would become a more significant safety concern. Had this issue not been detected, the instrument could have fallen well out of tolerance in the future leading to inaccurate readings. During post-LOCA operations, these inaccurate readings could have caused operators to establish insufficient BPC flows. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the lack of calibration procedures for the flow instrumentation would have eventually caused the instrument to read inaccurately, the inspection team identified the problem early enough so that the instrument drift (as provided in the vendor instructions) would not be large enough to cause inaccuracies that would adversely affect BPC flows.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, "Test Control" states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the licensee failed to identify and establish calibration testing for the flow instrumentation that would be used post-accident to ensure sufficient boron dilution flow. This flow instrumentation was placed into service in September 2003.

Because this failure to apply appropriate test control measures was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CR 06-00217, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 06000346/2006006-04 (DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From January 23 through February 10, 2006, the inspectors reviewed five Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES**

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. B. Allen and others of the licensee's staff, on February 10, 2006 and March 2, 2006. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Loehlein, Engineering Director
K. Zellers, Engineering Analysis Supervisor - Design Engineering
C. Price, Regulatory Affairs Manager
J. Grabnar, Engineering Design Manager
G. Wolf, Staff Engineer - Regulatory Affairs
D. Nassor, Engineer - Design Engineering
G. LeBlanc, Engineer - Design Engineering

Nuclear Regulatory Commission

C. Lipa, Reactor Projects Branch 4
D. Hills, Chief, Engineering Branch 1
J. Rutkowski, Senior Resident Inspector
R Smith, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000346/2006006-02	URI	Change to Design Basis Tornado Differential Pressure Design Limit for the Auxiliary Building

Opened and Closed

05000346/2006006-01	NCV	Non-Conservative Flow Testing Acceptance Criteria for the number 2 HPI and number 2 LPI Pumps
05000346/2006006-03	NCV	Failure to Consider Adverse Ampacity Effects of High Temperature Conditions in the Diesel Fire Pump Room
05000346/2006006-04	NCV	Failure to Provide Testing for Boron Precipitation Control Flow Instrumentation

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather, that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

10 CFR 50.59 Screenings/RAD

RAD 03-01815; Revision to Technical Requirements Manual sections 3/4.5.2, ECCS Subsystems; Revision 00

RAD 04-03194; Revise USAR Section 9.3.5.1; Revision 00

RAD 04-02770; USAR Clarification for Section 8.3.2.1.3 Battery Chargers; Revision 00

RAD 05-01054; Decay Heat Valve Pit Test (PAF 05-0246); Revision 00

Screen 04-03226-01; Service Water Pump Room Ventilation System Capacity; Revision 01

Screening 04-04000; C153, SBODG Electrical Equipment Room Exhaust Fan 90 Day Review; Revision 00

Screening 04-04675; Minimum Boric Acid Flow for Technical Specification 3.1.1.1; Revision 00

Screening 05-01762; Service Water Flow to CCW Heat Exchanger; Revision 00

Screening 05-03034; Elimination of Main Turbine Vibration Trip; Revision 00

Screen 05-04049; EDG Transient Loading Improvements; Revision 01

Screening 05-05307; Technical Requirements Manual Revision - Section 3.6.4.1, Combustible Gas Control - Hydrogen Analyzer; Revision 00

Screening 05-05611; USAR Clarification for ECCS Sump Pump Design Requirements, UCN 05-049U; Revision 00

Screening 05-06054; Minimum Boric Acid Flow for Technical Specification 3.1.1.1; Revision 00

10 CFR Part 50.59 Evaluations

04-02210; UCN 03-058 Tornado Subcompartment Pressurization Methodology Change Permitting COMPARE Methodology; Revision 02

04-03928; Changes to Containment Vessel Analysis Inputs; Revision 00

05-00908; Calculation C-NSA-060.00-014 and UCN 05-008, Loss of Feedwater Analysis; Revision 00

05-03247; USAR Change Notice for High Energy Line Break (HELB) Compartment Pressurization Analysis outside of Containment; Methodology Change to GOTHIC; Revision 00

IR17 Permanent Plant Modifications (71111.17B)

Modifications

ECP 02-0013-00; Revise Setpoint for Pressurizer High Level Alarm; Revision 00

ECP 02-0693-00; AFP and MDFP Discharge Pressure When Aligned to Service Water; Revision 00

ECP 03-0146-01; Utilization of Hot Leg Injection method as primary Boron Precipitation Control Method; Revision 01

ECP 04-0238-00; Service Water Margin Improvement Hardware Changes _ Hydrogen Analyzer; Revision 00

ECP 05-0021-00; Correct Statements in CL-1 and CL-2 for the Auxiliary Feedwater Pump Low-Low Suction Pressure Interlocks; Revision 00

ECP 05-0023-00; Service Water Flow to CCW Heat Exchanger; Revision 00

ECP 05-0030-00; Remove The Requirement to Tag the Breaker for Service Water Return Valve; Revision 00

ECP 05-0095-00; EDG Loading Improvements; Revision 00

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CR 06-00181; NRC 50.59/MOD Inspection - Untimely Issuance of a USAR Change Notice; dated January 25, 2006

CR 06-00211; NRC 50.59/MOD Inspection - Conflicting Statement In ECR 03-0146-01 Boron Precipitation MOD; dated January 26, 2006

CR 06-00217; NRC 50.59/MOD Inspection - FT4909 Lacks PM for Calibration; dated January 27, 2006

CR 06-00219; Non-Conservative Acceptance Criteria Used For Pump Testing; dated January 26, 2006

CR 06-00221; 10 CFR50.59 Database Inaccuracies; dated January 27, 2006

CR 06-00246; NRC 50.59/MOD Inspection - NRC Concern Regarding UCN 03-058; dated January 27, 2006

CR 06-00292; Inadequate Basis Provided for 50.59 Screen 04-04000; dated February 03, 2006

CR 06-00322; NRC 50.59/MOD Inspection - SD-018 Discrepancy; dated February 07, 2006

CR 06-00327; Cable Ampacity Calculation Has Non-Conservative Ambient Temperature; dated February 07, 2006

CR 06-00336; NRC 50.59/MOD Inspection - ECR 02-0693-00 50.59 Screen No. 04-00560; dated February 07, 2006

CR 06-00340; Omission in USAR Change Notice 03-058; dated February 09, 2006

CR 06-00347; NRC 50.59/MOD Inspection - T.S. 4.8.1.1.2.D.3 Change; dated February 08, 2006

CR 06-00354; NRC 50.59/MOD Inspection - Feedback Regarding Engineering Standards and Expectations; dated February 09, 2006

CR 06-00365; NRC 50.59/MOD Inspection - ECR 02-0693-00 RAD & 50.59 Screen No. 04-00560; dated February 13, 2006

Corrective Action Program Documents Reviewed During the Inspection

CR 03-04763; RFA - Test Criteria and alternative Methodology for TS 4.5.2.H Testing; dated June 17, 2003

CR 03-06870; CATI: NRC Unresolved Issues, Concerns with SW Pump Room HVAC Calc 67.005; dated August 23, 2003

CR 04-04456; Concern with 50.59 Documentation for UCN 04-057; dated July 8, 2004

CR 05-02649; NOP-CC-2003, Engineering Changes, Deficiencies; dated May 6, 2005

CR 05-04691; Acceptance Criteria of ECCS Sump Test below USAR Assumption; dated August 31, 2005

Calculations

Calculation 67.005; Service Water Pump Room Ventilation System Capacity; Revision 5

C-EE-002.01-010; DC Calc - Battery/Charger Size, Short Circuit, Voltage Drop; Revision 29

C-ME-24.03-02; SBO Building Ventilation; Revision 02

Drawings

Drawing E-46B sheet 4A; Elementary Wiring Diagram, Steam and Condensate Aux Feed Pumps Turbines Main Steam Inboard Isolation Valves; Revision 23

Drawing E-46B sheet 4B; Elementary Wiring Diagram, Steam and Condensate Aux Feed Pumps Turbines Main Steam Inboard Isolation Valves; Revision 22

Drawing E-46B sheet 46A; Elementary Wiring Diagram, Steam and Condensate SG AFPT Isolation Valve; Revision 19

Drawing E-46B sheet 46B; Elementary Wiring Diagram, Steam and Condensate SG AFPT Isolation Valve; Revision 17

Procedures

DB-SP-03136; Decay Heat Train 1 Pump and Valve Test; Revision 11

DB-SP-03137; Decay Heat Train 2 Pump and Valve Test; Revision 12

DB-SP-03218; HPI Train 1 Pump and Valve Test; Revision 10

DB-SP-03219; HPI Train 2 Pump and Valve Test; Revision 11

NG-RA-00806; Preparation and Control of USAR Changes; Revision 00

NOP-CC-2003; Engineering Changes; Revision 09

Miscellaneous Documents

License Amendment No. 20; Safety Evaluation of Nuclear Reactor Regulation Supporting Amendment No. 20 to Facility Operating License No. NPF-3; dated October 2, 1979

License Amendment No. 103; Amendment No. 103 to Facility Operating License No. NPF-3; Motor Driven Feedwater Pump System; dated September 2, 1987

UCN 03-083T; Change to Technical Requirement Manual 3/4.5; dated September 5, 2003

Completed OI 92A; Fuel Oil Ordering, Receipt Sampling and Offloading; dated April 4, 2005

Completed OI 92A; Fuel Oil Ordering, Receipt Sampling and Offloading; dated December 7, 2005

Operations Work Plan 2004-033; 1RH-713A and B Torque Determination; dated May 31, 2004

NRC SER dated July 9, 1997; Safety Evaluation Related to Amendment Nos. 174 and 178 to Facility Operating License Nos DPR-24 and DPR-27; dated July 9, 1997

Completed PBF 3005; Blended number 1 and number 2 Fuel Oil Acceptance Criteria; dated April 5, 2002

Completed PBF 3005; Blended number 1 and number 2 Fuel Oil Acceptance Criteria; dated December 28, 2004

Completed PBF 3005a; Quarterly Sampling of Emergency Fuel Oil Tanks – T-30 dated September 29, 2005

Completed PBF 3005a; Quarterly Sampling of Emergency Fuel Oil Tanks – T-32A; dated September 29, 2005

Completed PBF 3005a; Quarterly Sampling of Emergency Fuel Oil Tanks – T-32B; dated September 29, 2005

Completed RMP 9225-2; Defeating/Restoring Containment Personnel and Escape Hatch Door Interlocks; various from 2002 through 2005

Station Log; dated February 4, 2004

Station Log; dated February 6, 2004

TCN 2004-0339; Temporary Change - Isolation of the Containment Ventilation System Using the RMS High Alarm Automatic Trip Functions; dated May 21, 2004

Completed TS 10; Local Leak Test of Containment Airlock Bulkheads and Penetrations; dated March 27, 2005

Completed TS 10A; Containment Airlock Door Seal Testing Unit 2; dated March 31, 2005

Completed TS 80; Sampling of Emergency Fuel Oil Tanks (Quarterly); dated March 29, 2005

VPNPD 90-148; Supplement to 10 CFR Part 50.63, TAC. NOS. 68583 and 68587 Loss of All Alternating Current Power Davis-Besse Nuclear Power Station, Unit 1 and 2; dated March 30, 1990

WEP-89-143; Letter from Westinghouse to Point Beach; Transmittal of Midloop Calculations; dated June 30, 1989

WO 9950688; P-38A AFP Mini Recirc Control; dated January 25, 2002

WO 9950689; P-38B AFP Mini Recirc Control; dated January 25, 2002

WO 9926779; Replace Equalizing Device in Accordance with MR 99-036*A; dated February 21, 2004

WO 9926780; Replace Equalizing Device in Accordance with MR 99-036*B; dated February 21, 2004

WO 0203762001; MOV Actuator Checkout; dated April 14, 2003

WO 0309001; Extend RH and SI Vent Lines per MR 02-011*B; dated October 7, 2005

WO 0403678; Inadvertent Letdown Isolation and Loss of Heaters (All Heaters Tripped Off) Control; dated June 17, 2004

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
ASME	American Society of Mechanical Engineers
BPC	Boron Precipitation Control
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECP	Engineering Change Package
FENOC	First Energy Nuclear Operating Company
HPI	High Pressure Injection
IMC	Inspection Manual Chapter
IR	Inspection Report
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
MCC	Motor Control Center
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
UCN	USAR Change Notice
URI	Unresolved Item
USAR	Updated Safety Analysis Report