August 3, 2004

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 EVALUATIONS OF CHANGES, EXPERIMENTS, OR TESTS
AND PERMANENT PLANT MODIFICATIONS INSPECTION
REPORT 05000346/2004010(DRS)

Dear Mr. Bezilla:

On July 2, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed a routine baseline inspection at your Davis-Besse Nuclear Power Station. The enclosed report documents the inspection findings, which were discussed on July 2, 2004 and on July 23, 2004, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Specifically, this inspection focused on the baseline biennial inspections for evaluations of changes, tests, or experiments (10 CFR 50.59) and permanent plant modifications. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Because of the large number of plant modifications and change evaluations performed during the recent Davis-Besse shutdown, the team reviewed a larger than normal sample size of plant changes and modifications.

Based on this inspection, the team identified two Severity Level IV violations of NRC requirements associated with the failure to perform an adequate safety evaluation review as required by 10 CFR 50.59. Because the violations were non-willful and non-repetitive and because they have been 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 these Non-Cited Violations, 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, 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publically 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/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

Docket Nos. 50-346
License Nos. NPF-3

Enclosure: Inspection Report 0500346/2004010(DRS)
w/Attachment: Supplemental Information

cc w/encl: The Honorable Dennis Kucinich
G. Leidich, President - FENOC
J. Hagan, Senior Vice President
Engineering and Services, FENOC
L. Myers, Chief Operating Officer, FENOC
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Public Utilities Commission of Ohio
President, Board of County Commissioners
of Lucas County
C. Koebel, President, Ottawa County Board of Commissioners
D. Lochbaum, Union Of Concerned Scientists
J. Riccio, Greenpeace
P. Gunter, N.I.R.S.

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Sincerely,

/RA/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-346
License Nos:	NPF-3
Report No:	05000346/2004010(DRS)
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	June 28 through July 2, 2004
Inspectors:	R. Daley, Senior Reactor Engineer, Team Lead C. Acosta Acevedo, Reactor Engineer T. Bilik, Reactor Engineer J. Jacobsen, Senior Inspector P. Lougheed, Senior Reactor Engineer S. Sheldon, Reactor Engineer
Approved by:	J. Grobe, Chairman Davis-Besse Oversight Panel

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000346/2004010(DRS); 6/28/2004 - 7/2/2004; Davis-Besse Nuclear Power Station; Evaluations of Changes, Experiments or Tests, and Permanent Plant Modifications.

This report covers a five day period of announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by Region III inspectors. Two Green Severity Level IV Non-Cited Violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

No findings of significance were identified.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Severity Level IV Non-Cited Violation associated with the failure to perform an adequate safety evaluation review as required by 10 CFR 50.59 for changes made to the facility as described in the Updated Safety Analysis Report. Specifically, the licensee failed to perform a safety evaluation in accordance with 10 CFR 50.59 for changes made to Section 9.2.7.3.c of the Updated Safety Analysis Report concerning the low-low pressure interlock for the auxiliary feedwater pumps. The changes made by the licensee adversely affected an Updated Safety Analysis Report-described function in that a previously described automatic feature of the steam inlet valve to the auxiliary feedwater pump was changed to clarify that this automatic feature was not available under certain conditions.

  Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation, that is, the failure to evaluate the changes made to Section 9.2.7.3.c of the USAR, were assessed using the Significance Determination Process.

  This finding was determined to be more than minor because the inspectors could not determine reasonably that the change would not ultimately require NRC approval. The inspectors determined that this issue was of very low safety significance, because the design basis safety-related function of the auxiliary feedwater system, to remove reactor decay heat following a loss of normal feedwater, was not adversely affected, and because the team determined from the mitigating systems evaluation in the Phase 1 Screening Worksheet that all the questions were answered "No." Therefore, the results of the violation were determined to be of very low safety significance and the violation was classified as a Severity Level IV Violation. (Section 1R02)

- Green. The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59, “Changes, Tests, and Experiments,” based on the licensee performing an inadequate evaluation of a proposed change to the plant, regarding tornado missile protection of the diesel generator exhaust stacks and plant doors. Specifically, the licensee's response to the question posed in 10 CFR 50.59(c)(2)(vi) did not demonstrate that the proposed change did not create the possibility of a malfunction of equipment important to safety with a different result than any previously evaluated in the Final Safety Analysis Report (as updated).

  Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation, that is, the failure to demonstrate that the proposed change did not create the possibility of a malfunction of equipment important to safety with a different result, were assessed using the Significance Determination Process.

  This finding was determined to be more than minor because the inspectors could not determine reasonably that the change would not ultimately require NRC approval. The finding was determined to be of very low safety significance based on a significance determination process analysis of a loss of offsite power concurrent with loss of one emergency diesel generator and the violation was classified as a Severity Level IV Violation. (Section 1R02)

Cornerstone: Barrier Integrity

No findings of significance were identified.

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

Davis-Besse operated at or near full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

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

.1 Review of 50.59 Evaluations and Screenings

a. Inspection Scope

From June 28, 2004 through July 2, 2004, the inspectors reviewed eight evaluations performed pursuant to 10 CFR 50.59 requirements. The evaluations related to permanent plant modifications, setpoint changes, procedure changes, conditions adverse to quality, and changes to the Updated Safety Analysis Report (USAR). The inspectors reviewed the evaluations for thoroughness and to determine if prior NRC approval was obtained as appropriate. The inspectors also reviewed 16 screenings where the licensee 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 reviewed the changes to determine if they met the threshold to require a 10 CFR 50.59 evaluation. These evaluations and screenings were chosen based upon a consideration of the risk significance of samples from the different cornerstones. The list of documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

b.1 Inadequate Evaluation of USAR Change for Steam Inlet Valve to Auxiliary Feedwater Pumps

Introduction: The inspectors identified that the licensee failed to perform an adequate safety evaluation in accordance with 10 CFR 50.59 before making changes to the USAR associated with the low-low pressure interlock for the steam inlet valve to the auxiliary feedwater (AFW) pumps. The issue was considered to be of very low safety significance (Green) and was dispositioned as a Severity Level IV Non-Cited Violation (NCV).

Description: On April 24, 2004, the licensee completed USAR Change Notice (UCN) 04-021, “Clarification to the Operation of the Interlocks Associated with the Auxiliary Feedwater Pump Turbine Steam Supply Valves.” UCN 04-021 changed statements in the USAR that described the operation of the low-low pressure interlock for the AFW Pumps while a SFRCS signal is also present. Specifically, the original USAR wording in Section 9.2.7.3.c stated, “If suction pressure remains low (1psig for 60 seconds) the steam supply valves will close to protect the AFP [Auxiliary Feedwater Pump] from cavitation and the SFRCS signal to the valves will be locked out to prevent valve motor damage. The steam supply valves to the AFPT [Auxiliary Feedwater Pump Turbine] will be opened automatically when a suction pressure is re-established and an SFRCS signal remains present." This wording was changed to the following: "The steam supply valves to the AFPT will be opened automatically when adequate suction pressure is re-established and an SFRCS signal remains present provided the steam pressure remained above the AFPT inlet steam pressure interlock setpoint. If steam pressure dropped below the AFPT inlet steam pressure interlock setpoint, manual action will be required to open the steam supply valves."

In Regulatory Applicability Determination (RAD) No. 04-00747, the licensee evaluated UCN 04-021, and determined that the changes made in UCN 04-021 were already evaluated and accepted by a previous safety evaluation. When the inspectors reviewed the referenced safety evaluation, they were not able to find a specific evaluation of this change to this USAR described function. The inspectors determined that this change adversely affected a USAR-described function in that, a previously described automatic feature of the steam inlet valve to the AFW pump was changed, to clarify that this automatic feature was not available under certain conditions. Because of these adverse effects to the USAR, the change should have been evaluated in accordance with the requirements contained in 10 CFR 50.59. Based upon the inspectors' concerns, the licensee initiated CR 04-04338.

<u>Analysis</u>: Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP). In this case, the licensee failed to perform a safety evaluation for changes made to the USAR concerning the low-low pressure interlock for the AFW pumps in accordance with 10 CFR 50.59.

This finding was determined to be more than minor because the inspectors could not reasonably determine that the change would not ultimately require NRC approval. The inspectors determined that even though the change was not adequately evaluated in accordance with 10 CFR 50.59, this violation of the regulatory requirements was of very low safety significance, because the design basis safety-related function of the AFW system, to remove reactor decay heat following a loss of normal feedwater, was not adversely affected. The inspectors completed a significance determination of this finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined from the mitigating systems evaluation in the Phase 1 Screening Worksheet that all the questions were answered "No," therefore the finding was determined to be of very low safety significance (Green).

<u>Enforcement</u>: On July 2, 2004, while performing the baseline procedure 71111.02, the inspectors identified a NCV of 10 CFR 50.59(d)(1). Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written

evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, the licensee failed to provide a basis in RAD 04-00747 for the determination that changing requirements in USAR Section 9.2.7.3.c associated with the low-low pressure interlock for the AFW pumps (as documented in UCN 04-021 dated April 24, 2004), was acceptable without a license amendment. The results of this violation were determined to be of very low safety significance. Therefore, this violation was classified as a Severity Level IV Violation. The licensee entered this issue into the corrective action program (CR 04-04338). This Severity Level IV violation is being treated as a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000346/2004010-01(DRS)).

b.2 Inadequate Evaluation of Missile Protection for Diesel Generator Exhaust Stacks and Plant Doors

Introduction: The inspectors identified an inadequate evaluation performed pursuant to 10 CFR 50.59 associated with the vulnerability of plant doors and the emergency diesel generator (EDG) exhaust stacks to tornado driven missiles. Specifically, the licensee did not provide an adequate response to the question posed in 10 CFR 50.59(c)(2)(vi) and did not demonstrate that the proposed change did not create the possibility of a malfunction of equipment important to safety with a different result than any previously evaluated in the final safety analysis report (as updated). This issue was considered to be of very low safety significance (Green) and was dispositioned as a Severity Level IV NCV.

Description: The inspectors reviewed 10 CFR 50.59 evaluation, 02-1740, Revision 1. In this evaluation, the licensee accepted an as-found plant condition that the EDG exhaust stacks, as well as certain plant doors, were not physically protected from the effects of tornado missiles as stated in the USAR. On page 8 of evaluation 02-1740 in Section 4.6, the licensee responded to the question posed in 10 CFR 50.59(c)(2)(vi). This question asked, "Does the proposed activity create a possibility for a malfunction of a system, structure, or component important to safety with a different result than any previously evaluated in the final safety analysis report (as updated)?" The licensee identified that a non-USAR-described malfunction was introduced. However, the licensee's evaluation attempted to justify why there was not a credible effect on any system, structure or component such that any potential malfunction was eliminated. The licensee referred to a "threshold of credibility" on the order of 1E-6 per year, which the licensee stated was in accordance with Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, as supported by the FENOC 10 CFR 50.59 Guidance Document. To support the credibility threshold determination, the licensee performed a calculation which used a computer code to calculate the probability of tornado missile damage to safety related systems, structures and components without physical protection.

The inspectors were unable to locate a definition of a "threshold of credibility" within NEI 96-07, which the NRC endorsed in Regulatory Guide 1.187. Furthermore, the inspectors determined that NEI 96-07 did not discuss any numerical value below which potential malfunctions were eliminated. Instead, the NEI document stated that possible malfunctions were limited to those as likely to happen as those described in the USAR and that a proposed change or activity which increased the likelihood of a malfunction previously thought to be incredible to the point where it becomes as likely as the malfunctions assumed in the USAR could create a possible malfunction with a different result.

The inspectors noted that tornado generated missiles were evaluated as a potential failure mechanism in the USAR. Therefore, the inspectors determined that the possibility of the EDG exhaust stacks being hit by a tornado generated missile was as likely (on approximately the same order of magnitude) as any other plant feature being struck by a tornado generated missile. The inspectors also noted that failure of the EDGs due to a tornado generated missile was previously considered to be incredible, as the USAR stated that they were physically protected from the effects of tornadoes. Consequently, the inspectors concluded that the change being evaluated appeared to have increased the likelihood of a malfunction previously thought to be incredible to the point where it becomes as likely as the malfunctions assumed in the USAR. The inspectors consulted with the staff in the Office of Nuclear Reactor Regulation (NRR). The NRR staff determined that the inspector's conclusions were in accordance with the staff's understanding of both 10 CFR 50.59 and NEI 96-07, Revision 1.

Based on the above, the inspectors determined that the licensee had not provided sufficient justification to answer 10 CFR 50.59(c)(2)(vi) as "No." Specifically, the inspectors concluded that the possibility of a malfunction with a different result appeared to have been created due to the change, because the possible malfunction was at least as likely to happen as those described in the USAR and because the proposed change increased the likelihood of a malfunction previously thought to be incredible to the point where it becomes as likely as the malfunctions assumed in the USAR.

<u>Analysis</u>: This issue was determined to involve a performance deficiency because the licensee misapplied the criteria of 10 CFR 50.59 and consequently concluded that prior NRC approval was not required when such a conclusion could not be supported by the documented 50.59 evaluation. Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP) described in Inspection Manual Chapter (IMC) 0609, "Significance Determination Process."

The inspectors determined that the finding was more than minor because physical barriers were degraded or missing and because those barriers being degraded could result in one or more of the diesel generators failing to fulfill their design function during a tornado. This was a design issue which affected the Mitigating Systems Cornerstone objective of equipment reliability. Additionally, this finding was determined to be more than minor because the inspectors could not reasonably determine that the change would not ultimately require NRC approval.

The inspectors then assessed the issue through Phase 1 of the SDP. The inspectors answered the question, "Does this issue involve an actual loss of safety function," as

"Yes," because under a design basis tornado, the EDG exhaust stacks were not physically protected. Based on this premise, the inspectors entered Phase 2 of the SDP.

The inspectors determined that the only event tree affected was a loss of offsite power concurrent with a loss of one EDG. This was based on the assumption that a tornado missile hitting both EDG exhaust stacks would be an incredible event. The team decreased the initiating event frequency from a "5" (once in 100,000 years) to a "3" (once in 1,000 years) based on the fact that the Davis-Besse switchyard was struck by a tornado in 1998. Based on these credible assumptions, this issue was determined to have very low safety significance or Green. Because the issue was of very low safety significance, the 10 CFR 50.59 violation was categorized as Severity Level IV.

Enforcement: On July 2, 2004, while performing the baseline procedure 71111.02, the inspectors identified a NCV of 10 CFR 50.59 (d)(1). Title 10 CFR 50.59(d)(1) requires the licensee to maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to 10 CFR 50.59(c). It further requires that these records include a written evaluation which provides the bases for the determination that the change, test or experiment does not require a license amendment pursuant to 10 CFR 50.59(c)(2).

Contrary to the above, on October 16, 2003, the licensee approved a 10 CFR 50.59 evaluation (02-1740) incorporating a change in the design basis to accept the lack of physical protection for the EDG exhaust stacks from tornado missiles. However, this evaluation did not provide an adequate basis for why a possibility for a malfunction of the diesel generators due to impact on the diesel generator exhaust stacks by a tornado driven missile did not produce a different result than any previously evaluated in the USAR as required by 10 CFR 50.59(c)(2).

The failure to provide a written evaluation which described the basis for concluding a license amendment was not needed was a violation of 10 CFR 50.59(d)(1). This issue had been entered into the licensee's corrective action program as CR 04-04685. This Severity Level IV violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000346/2004010-02).

1R17 Permanent Plant Modifications (71111.17)

.1 Review of Recent Modifications

a. Inspection Scope

From June 28, 2004 through July 2, 2004, the inspectors reviewed 16 permanent plant modifications. The modifications were chosen based upon a consideration of probabilistic risk analysis (PRA) significance in the licensee's Individual Plant Evaluation (IPE). The inspectors reviewed these 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 affect any system's 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 list of documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Condition Reports

a. Inspection Scope

From June 28, 2004 through July 2, 2004, the inspectors reviewed a selected sample of condition reports associated with Davis-Besse Nuclear Power Station's permanent plant modifications and concerning 10 CFR 50.59 evaluations and screenings. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. In addition, condition reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Bezilla and other members of licensee management on July 2, 2004, and by a telephone exit on July 23, 2004. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Blakely, Design Engineer
J. Grabner, Design Engineering Manager
P. Jacobsen, Design Engineer
S. Osting, Design Engineer
G. Wolf, Regulatory Affairs
K. Zellers, Design Engineer

Nuclear Regulatory Commission

J. Grobe, 0350 Chairman
D. Hills, Region III Materials Engineering Branch Chief
J. Lara, Region III Electrical Engineering Branch Chief
C. Lipa, Reactor Projects Branch 4
C. Thomas, Davis-Besse Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000436/2004010-01	NCV	Inadequate Safety Evaluation for Changes to the Plant made as Described in the USAR Concerning the low-low pressure interlock for the AFW Pumps
05000436/2004010-02	NCV	Inadequate 10 CFR 50.59 Evaluation Regarding Tornado Missile Protection for EDG Exhaust Stacks

Closed

05000436/2004010-01	NCV	Inadequate Safety Evaluation for Changes to the Plant made as Described in the USAR Concerning the low-low pressure interlock for the AFW Pumps
05000436/2004010-02	NCV	Inadequate 10 CFR 50.59 Evaluation Regarding Tornado Missile Protection for EDG Exhaust Stacks

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

Corrective Action Documents Generated During the Inspection

CR 04-04293; NRC MOD/50.59: AOV SS598/SS607 Past Operability/Reportability; dated June 30, 2004

CR 04-04321; Date Discrepancies Between Modification Order Signoff and Post Mod. Test Date; dated July 1, 2004

CR 04-04327; NRC/50.59: Unable to Find Record of SC-3122 for Mod 03-0299; dated July 1, 2004

CR 04-04338; USAR Change Documentation Deficiency; dated July 1, 2004

CR 04-04347; PCR/CR: NOP-CC-3002, Calculations; dated July 1, 2004

CR 04-04348; NRC MOD/50.59: Engineering Assessment Issues; dated July 1, 2004

CR 04-04685; NRC MOD/50.59; Use of 1E-6 in 50.59 Evaluation Questioned; dated July 21, 2004

Corrective Action Documents Reviewed During Inspection

CR 00-0271; Review of ANO-1 Due to Low Temperature Service Water; dated February 11, 2000

CR 00-1683; Evaluation of Makeup Pump Operation with Reduced Component Cooling Water Temperatures; dated June 26, 2000

CR 01-2855; Untimely Equivalent Change Evaluation for Train 1 HPI Motor Replacement; dated March 23, 2001

CR 01-3224; Auxiliary Feedwater to Steam Generator Stop Valves; dated December 4, 2001

CR 02-00784; Collective Review the Nuclear Fuel Related CR's for Common Causes; dated March 11, 2002

CR 02-06178; Spacer Grid Damage Observed During Fuel Inspections; dated September 22, 2002

CR 02-08201; Improper Metal Composition in Valves; dated November 16, 2002

CR 02-09187; Discrepancy Discovered in Safety Evaluation 01-0023; dated November 8, 2002

CR 02-09895; PR/MOD: Inadequate Verification of Vendor Products; dated December 5 2002

CR 02-09909; PR/MOD: Inadequate Design Verification and Review; dated December 5, 2002

CR 02-09938; PR/MOD: Deficiencies in Design Interface Evaluations (DIE); dated December 6, 2002

CR 03-00890; SS 598/607 Do Not Meet AOV Acceptance Criteria; dated February 1, 2003

CR 03-04648; 10 CFR 50.59 Review Not Performed as Required for Category B Software Change; dated June 12, 2003

CR 03-04705; RFA- Open Assumption on Calculations; dated June 14, 2003

CR 03-04838; EDG Air Start Mod Drawing Discrepancies; dated June 19, 2003

CR 03-05586; Engineering Change Request Issued Without Required Inservice Inspection Review; dated July 14, 2003

CR 03-05382; PCR:DB-OP-00016, Temporary Configuration Control; dated July 8, 2003

CR 03-01492; Fuel Assembly NJ127T Damaged Spacer Grid; dated November 10, 2003

CR 02-09975; PR/MOD Test Requirements not Specified by DBE as Required by NOP-CC-2003; dated December 7, 2002

CR 03-06128; Loss of Power to Boric Acid Line Heat Tracing After a Tornado; dated July 31, 2003

CR 03-10499; High Pressure Injection Pump Hydraulic Performance; dated December 4, 2003

CR 04-02511; Failure of ABDD2 to Open; dated April 5, 2004

CR 04-02522; Breaker A2000Q03 Failed DB-OP-01000 Testing; dated April 6, 2004

Calculations

C-EE-004.01-049; 4.16 kV Bus C1/D1 Degraded Voltage, Loss of Voltage, and 27X-6 Relay Setpoints; Revision 15

C-EE-004.01-057; Setpoint Determination for EDG Breaker Closure Timer 27X-6/C1(D1); Revision 00

C-ME-063.01-123; Component Level Review for AOVs SS 598 - 607; Revision 0

C-ME-065.01-156; Accumulator Sizing Calculation for AOV MU38 and MU3; dated June 6, 2003

C-CSS-099.20-026; Probability of Tornado Missile Damage to Davis-Besse Missile Exposed Targets; Revision 0

Modifications

EWR 01-0072-00; Replace Check Valve IA 501 with a More Reliable Valve; dated March 14, 2001

EWR 02-0374; Modification of Pipe Supports; dated February 6, 2003

ECR 02-0558-00; SV 578 and SV 578A Replacement; Revision 0

ECR 02-0657-00; Removable Insulation Panels for Reactor Flange O-Ring Monitoring Lines; Revision 00

ECR 02-0658-00; Install 4" HBD-171 Service Water Line Isolation Valve; Revision 0

ECR-03-0091-00; Replacement of Time Delay Pickup Relays 27X-6/C1 and 27X-6/D1; Revision 1

ECR 03-0096-00; SS 598 and SS 607 SG 1-2 and 1-1 Sample Line Containment Isolation Valve 180^o Rotation; dated March 28, 2003

ECR 03-0112-00; Replace MU 38 Containment Isolation Valve; dated May 2, 2003

ECR 03-0287-00; Replacement of Motor Operated Potentiometer with Digital Reference Unit for EDG K5-1 Speed Control; Revision 0

ECP 03-0299-00; Add Air Accumulators for Fail Safe Positioning of Component Cooling Water Heat Exchanger Service Water Outlet Valves; Revision 9

ECP 03-0341-00; PORV (RC2A) Conformal Coating of the Terminations; dated July 3, 2003

ECR 03-0409-00; Modification to Base Plates and Anchorage for EDG Exhaust Stack Missile Shields; dated July 22, 2003

ECP 03-0463-00; Equivalent Replacement for DBC2PN Input Circuit Breaker; dated September 29, 2003

MOD 97-0029-00; Fail Valves CC1471 and CC1474 in the Open Position; Revision 0

MOD 98-0041-00; Lower Setpoint of PIC 2796, Condensate Pumps Runout Pressure Indicator Controller; Revision 0

Mod 99-0020-00; Pressurizer SCR Heater Bank #1 and Non-Essential Heater Bank #4 Reconfiguration; dated October 5, 2002

<u>10 CFR 50.59 Evaluations</u>

02-01740; TORMIS Approach for Tornado Missile Risk Evaluation; Revision 1

02-02231; Criteria for Establishing Closure Times for Containment Isolation Valves; Revision 0

03-00642; Evaluation of Proposal Modification to Containment Isolation Valve MU38 to Change it from a “Spring to Close” Valve to an “Air to Close Valve”; Revision 1

03-01367; Revise 4160V Degraded Voltage Relay Setpoint; dated August 2, 2003

03-01375; Replace Obsolete Motor Operated Potentiometer on the EDG with an Equivalent Unit; Revision 2

03-01398; Re-Power MCC “F13" from Bus “F7;" dated August 11, 2003

03-02048; HPI Pump Upgrade to Minimize Debris Damage During Suction From the Containment Sump; dated November 25, 2003

03-02673; Change to High Pressure Injection (HPI) Pump Curve Acceptance Criteria; Revision 0

<u>10 CFR 50.59 Screenings</u>

01-00619; DB-OP-02512 Loss of RCS Makeup C-4 to R-3; Revision 00

02-00882; Instrument AC System Procedure; Revision 04

03-00087; Use of Fuel Assemblies With Spacer Grid Damage in Cycle 14 Core - Modes 3, 4, 5, 6; dated February 25, 2003

03-00124; SRFCS/ICS High Steam Generator Level Setpoint Change; Revision 0

03-00216; DB-OP-02527, R3, Rad Loss of Decay Heat Removal; Revision 0

03-01017; Revise Technical Specification Bases to Remove Discussion of DH Valve Watertight Enclosure; dated June 16, 2003

03-01235; ECR 03-0341-00, Coating of the PORV Solenoid Terminals with EGS Conformal Coating; dated July 2, 2003

03-01467; Modification to Base Plates of the EDG Exhaust Stack Missile Shield; dated July 22, 2003

03-01930; USAR Changes Associated With LAR 03-0010; dated September 12, 2003

03-02048; HPI Pump Upgrade to Minimize Debris Damage During Suction From the Containment Sump; dated November 25, 2003

03-02205; Correct SFRCS Logic to Prevent Block on Restoration of Power; Revision 0

03-02481; DB-OP-06012, R13, Decay Heat and Low Pressure Injection System Operating Procedure; Revision 0

03-02850; DB-OP-06233, R12, Auxiliary Feedwater System; Revision 0

04-00998; DB-OP-02521, Loss of AC Bus Power Sources SM04-0831; dated May 6, 2004

04-01008; Revise TS Bases to Describe AFW System Interlocks Logic and Provide Guidance on Surveillance Requirements with Exceptions to TS 4.0.4; dated May 11, 2004

04-01136; Editorial Changes to the USAR; Revision 0

Procedures

EN-DP-01072; Modification Test Requirements; Revision 5

NOBP-LP-4003A; FENOC 10 CFR 50.59 User Guidelines; Revision 0; dated April 16, 2004

NOP-CC-2003; Engineering Changes; Revision 3

NOP-CC-3002; Calculations; Revision 1

NOP-LP-4003; Evaluation of Changes, Tests and Experiments; Revision 1

USAR Change Notices

UCN 01-061; Revise USAR Section 15.4.4.1 and 15.4.4.2.3.3; dated May 27, 2004

UCN 02-044; Revise Section 9.2.7.3.c to Clarify that the AFP Steam Supply Valves Will Automatically Reopen when a Suction Pressure Is Re-established Only if there Is a SFRCS Signal Present; dated October 10, 2002

UCN 03-022; Revision of UFSAR Section 3.8.2.1.11

UCN 03-114; Add FLUS Leak Detection System Description to USAR; dated April 6, 2004

UCN 04-021; Clarification to the Operation of the Interlocks Associated with the AFP Turbine Steam Supply Valves; dated April 24, 2004

Work Orders

WO 02-002142-000; Auxiliary Feedwater to Steam Generator 2 Line Stop; dated September 9, 2003

WO 02-002145-000; Aux Feed to Stm Gen 1-1 Line Stop Valve (E11E BE1160); dated September 9, 2003

WO 03-001802-000; RCP Seal Return Isolation Valve; dated July 17, 2003

WO 200036283; Install Accumulator and Modify Control Circuitry per ECR 03-0299; dated September 12, 2003

WO 200036280; Install Accumulator and Modify Control Circuitry per ECR 03-0299; dated September 11, 2003

WO 200036281; Install Accumulator and Modify Control Circuitry per ECR 03-0299; dated September 12, 2003

Miscellaneous Documents

DB-PF-03008; Local Leak Rate Test of Makeup MU 38; dated June 23, 2003

DB-PF-03008; Local Leak Rate Test of Instrument Air Valve IA 501; dated December 3, 2003

DB-PF-03386; Makeup System Valve Testing; dated May 22, 2003

DB-PF-05010; Electrical Continuity Check for ECR 03-0299; dated September 5, 2003

Framatome ANP Engineering Information Record 51-1263633-02; Evaluation of Spacer Grid Damage at DB, EOC 13; dated February 25, 2003

Framatome ANP QA Data Package 23-5023500-00; Recon/Loose Rod Inspection; dated January 10, 2003

Framatome ANP QA Data Package 23-5020102-00; FA Recon/Recage (Mk-B10A, Mk-B10K, Mk-B10M); dated September 5, 2002

Framatome Licensing Doc Approval 43-2149A-00; Evaluation of Replacement Rods in BWFC Fuel Assemblies; dated January 1, 2001

NEI 96-07; Guidelines For 10 CFR 50.59 Implementation; Revision 1

Safety Evaluation Report (SER) - Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probablistic Risk Assessment (PRA) Methodology; dated October 26, 1983

USAR 9.2.2; Component Cooling Water; Revision 23

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
AFP	Auxiliary Feedwater Pump
AFPT	Turbine Driven Auxiliary Feedwater Pump
AFW	Auxiliary Feedwater
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
ECP	Engineering Change Package
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
EWR	Engineering Work Request
FENOC	First Energy
IMC	Inspection Manual Chapter
IPE	Individual Plant Evaluation
IR	Inspection Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	United States Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PRA	Probabilistic Risk Assessment
RAD	Regulatory Applicability Determination
RG	Regulatory Guide
SDP	Significance Determination Process
SE	Safety Evaluation
SFRCS	Steam Feedwater Rupture Control System
TLCO	Technical Requirements Manual Limiting Consideration for Operability
TRM	Technical Requirements Manual
TSR	TRM Surveillance Requirement
UCN	USAR Change Notice
USAR	Updated Safety Analysis Report
WO	Work Order
wpd	WordPerfect Document
www	World Wide Web