April 20, 2001

EA-01-097

Mr. Guy G. Campbell
Vice President - Nuclear
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 INSPECTION REPORT 50-346/01-06(DRP)

Dear Mr. Campbell:

On March 31, 2001, the NRC completed an inspection at your Davis-Besse reactor facility. The results were discussed with you and other members of your staff on April 3, 2001. The enclosed report presents the results of that inspection.

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

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Davis-Besse facility.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas J. Kozak, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 50-346/01-06(DRP)

cc w/encl: B. Saunders, President - FENOC
Plant Manager
Manager - Regulatory Affairs
M. O'Reilly, FirstEnergy
Ohio State Liaison Officer
R. Owen, Ohio Department of Health
A. Schriber, Chairman, Ohio Public
Utilities Commission

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/01-06(DRP)
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 N. State Route 2 Oak Harbor, OH 43449-9760
Dates:	February 14 through March 31, 2001
Inspectors:	K. Zellers, Senior Resident Inspector D. Simpkins, Resident Inspector
Approved by:	Thomas J. Kozak, Chief Projects Branch 4 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process (SDP), and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 50-346-01-06(DRP), on 02/14 through 03/31/2001; FirstEnergy Nuclear Operating Company; Davis-Besse Nuclear Power Station. Maintenance Risk Assessments and Emergent Work.

The inspection was conducted by resident inspectors. The inspection identified one Green finding, which was a Non-Cited Violation. The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED) and was determined by the Significance Determination Process.

Cornerstone: Initiating Events

- Green — The licensee failed to adequately assess the risk to the plant from performing reactor protection system troubleshooting with one train of auxiliary feedwater unavailable.

  The issue, which was a Non-Cited Violation of 10 CFR 50.65 (a)(4), had very low risk significance because the change in core damage probability was very low. This was due to the short time duration that the auxiliary feedwater system train was unavailable (Section 1R13).

Report Details

Summary of Plant Status:

The plant was operated at about 100 percent power throughout the inspection period, except for a planned down power to about 55 percent power from March 30 to April 1 for routine maintenance activities and brief down powers to about 93 percent power for routine testing.

1. REACTOR SAFETY

1R04 Equipment Alignment (Inspection Procedure 71111.04)

a. Inspection Scope

The inspectors conducted partial walk-down inspections by comparing station configuration control documentation with actual system/train lineups on the following trains of equipment to verify the system/train was operable when a redundant system/train was out-of-service:

- #1 High Pressure Injection Train during a #2 High Pressure Injection Train outage (documents reviewed: Operations Schematic OS-003).

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (Inspection Procedure 71111.11)

a. Inspection Scope

The inspectors observed risk-important licensed operator actions and emergency plan implementation for a steam generator tube rupture scenario on the plant simulator to identify deficiencies and discrepancies in the training and to assess operator performance and evaluator critiques.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (Inspection Procedure 71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements, including a review of scope, goal setting, and performance monitoring, short-term and long-term corrective actions, and current equipment performance status for the following components and systems which have had performance problems:

- Containment Spray (documents reviewed: Davis-Besse (DB) Maintenance Rule Program for Containment Spray, Davis-Besse System Health Report - Fourth Quarter, 2000, Unit Logs, Condition Report (CR) 2000-1709, CR 2000-2190, CR 2000-2809, CR 2000-2841, CR 2001-0125, and CR 2001-0502).

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the effectiveness of the risk assessments performed before maintenance was conducted on structures, systems, and components (SSCs) and verified how risk was managed and if maintenance risk assessments and emergent work problems were adequately identified and resolved for the following activities:

- #2 Auxiliary Feedwater (AFW) System Pump not available during testing (documents reviewed: DB-PF-03162 (Forward/ Reverse Flow Test for AF 68), CR 01-147, Maintenance Rule Program Manual for AFW, Risk Assessment Test Data Base, Unit Log, Davis-Besse Weekly Maintenance Risk Summary - Daily Review for 1/17/01, WPG-1 (Work Process Guidelines)).

b. Findings

The inspectors identified a green finding and a related Non-Cited Violation of 10 CFR 50.65(a)(4) which were associated with the licensee’s failure to adequately assess plant risk from concurrently performing reactor protection system troubleshooting and an AFW system reverse flow check valve test.

On January 17, 2001, the licensee planned to conduct reactor protection system troubleshooting, which was a reactor trip initiating activity, and a 4-hour duration AFW system reverse flow check valve test. The inspectors reviewed the unit log on January 18, 2001, and noted that the AFW system reverse flow check valve test caused the AFW train to be unavailable. The inspectors then questioned work management personnel if the risk should have been higher than what was assessed, because the inspectors recalled that in the past AFW unavailability by itself was as high a risk as the evaluation that was performed for the January 17 activities. After a review, work management personnel identified that the risk assessment software input data did not configure the AFW train as unavailable during the test. Risk management personnel then corrected the input data and determined they had been in a higher risk category which per station procedures required that the AFW test be rescheduled to avoid the relatively higher risk. In response to the issue the licensee generated CR 01-0147 and conducted a root cause evaluation. The root cause was determined to be work practice and change management related. Corrective actions taken thus far were to perform an in-depth review of future risk assessments, issue an operator standing order to require independent reviews of risk assessments, and perform independent reviews of input data performed by risk assessment engineers.

This finding was considered to be more than minor because it had a credible impact on safety due to an actual increase in core damage probability. The issue affected the initiating events cornerstone because an initiating event activity (reactor protection system troubleshooting) was performed without properly assessing the risk (one train of AFW unavailable was not included in the risk assessment). A Region III risk analyst (Senior Reactor Analyst) familiar with the licensee's probabilistic risk assessment model and risk assessment tools determined that the licensee's calculation of the instantaneous core damage frequency for the issue (5.16E-4) was appropriate. Although the instantaneous core damage frequency was relatively high, the change in core damage probability was very low (less than 1E-6) due to the short time duration that the AFW train was unavailable (1 hour and 43 minutes) during the troubleshooting activity. Therefore, the finding was assessed as having very low risk significance (Green).

Paragraph (a)(4) of 10 CFR 50.65 states, in part, that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this, on January 17, 2001, the licensee did not adequately assess the increase in risk associated with concurrent maintenance activities on the reactor protection system and the AFW system. Specifically, the licensee failed to include the unavailability of the AFW system train due to surveillance testing in their risk assessment. Since this issue was determined to have very low safety significance and was characterized as Green by the SDP, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A.I of the NRC Enforcement Policy **(NCV 50-346-01-06-01)**. This violation is in the licensee's corrective action program as CR 2001-0147.

1R14 Personnel Performance Related to Non-routine Plant Evolutions and Events

a. Inspection Scope

The inspectors reviewed station personnel preparations and operator performance for a reactor down-power to about 55 percent for planned routine maintenance. This review was to determine if personnel actions were appropriate to the evolution and in accordance with procedures and training. Documents reviewed included the unit log, the work week schedule and plant computer logs.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (Inspection Procedure 71111.15)

a. Inspection Scope

The inspectors reviewed the following operability evaluations affecting mitigating systems and barrier integrity. The reviews considered whether the evaluations were technically justified, the adequacy and functionality of any compensatory measures, and any degradations that might cause a loss of function as described in the Updated Safety Analysis Report (USAR) or Technical Specifications (TSs).

- Emergency Core Cooling System Room Cooler (documents reviewed: CR 2001-0013, CR 2000-3086, Maintenance Work Order (MWO) 00-3575-000, MWO 00-1279-000, Operability Justification 2000-0018, Operability Justification 2001-0001, Unit Logs, USAR Sections 9.2 and 9.4, System Description (SD)-18 (Service Water), NRC Generic Letter 91-18, and DB-OP-06261 (Service Water System Operating Procedure)).

- Station Blackout Diesel Generator and Emergency Diesel Generator Mechanical Friction Clutch Locknut (documents reviewed: CR 2000-2483, Operability Justification 2000-0015, and DB-MM-09320 (Emergency and Station Blackout Diesel Engine Maintenance)).

b. Findings

No findings of significance were identified.

1R16 Operator Work-arounds (Inspection Procedure 71111.16)

a. Inspection Scope

The inspectors reviewed operator work-arounds individually and cumulatively by reviewing documentation and interviewing station personnel to identify any potential effects on the functionality of mitigating systems or the ability of operators to implement abnormal or emergency operating procedures. Operator work-arounds reviewed were: #2 emergency diesel generator fuel oil transfer pump does not auto start, #1 Service Water Pump upper motor bearing oil leak, Electric Fire Pump discharge check valve, Diesel Fire Pump Day Tank fill valve, and weekly drain of AFW pump turbine. Documents reviewed were WPG-2 (Work Process Guideline - Operations Equipment Issues), and the active operator work-around log.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the following plant modifications against the design bases, licensing bases, and performance capabilities to ensure risk significant SSCs had not been degraded and modifications performed during increased risk-significant configurations did not place the plant in an unsafe condition:

- Safety Features Actuation System Containment High-High Pressure Setpoint Modification (documents reviewed: SD-002 (Safety Features Actuation System), DB-MI-03107 (Calibration of PT-2000 Containment Pressure Transmitter to SFAS Channel 1), DB-MI-03108 (Calibration of PT-2001 Containment Pressure Transmitter to SFAS Channel 2), DB-MI-03109 (Calibration of PT-2002 Containment Pressure Transmitter to SFAS Channel 3), DB-MI-03110 (Calibration of PT-2003 Containment Pressure Transmitter to SFAS Channel 4), DB-MI-03111 (Channel Calibration of 59-ISP2000 Containment Pressure Input to SFAS Channel 1), DB-MI-03112 (Channel Calibration of 59-ISP2001 Containment Pressure Input to SFAS Channel 2), DB-MI-03113 (Channel Calibration of 59-ISP2002 Containment Pressure Input to SFAS Channel 3), DB-MI-03114 (Channel Calibration of 59-ISP2003 Containment Pressure Input to SFAS Channel 4), USAR Section 7.3, License Amendment Request 98-005, TS 3/4.3.2.1, TS Amendment 243, Modification 99-0029-00, MWO's 00-4455-000, 00-4455-001, 00-4455-002, 00-4455-003 and 00-4455-004).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (Inspection Procedure 71111.22)

a. Inspection Scope

The inspectors witnessed the following surveillance tests and/or reviewing the test data, to verify that the subject risk-significant SSCs were capable of performing their intended safety function. The inspectors conducted reviews of TS, USAR, and licensee procedure requirements and evaluated the tests for potential preconditioning, effects on plant risk, clear and adequate acceptance criteria, operator procedural adherence, test data completeness, test frequency, test equipment range and accuracy, and post-test equipment restoration:

- AFW Pump 1 Quarterly Test (documents reviewed: DB-SP-03151 (AFP 1 Quarterly Test), OS-17A, OS-10, M-006D, USAR 15.2.8.2.3, SD-015 (AFW System), Pump and Valve Basis Document).

- High Pressure Injection (HPI) Pump 2 Quarterly Test (documents reviewed: DB-SP-03219 (HPI Pump 2 Quarterly Pump and Valve Test), SD-38 (HPI System), OS-003 (HPI System), Maintenance Rule Program Manual, Pump and

Valve Basis Document for HPI Pumps, revision 04), Calculation C-NSA-52.01-003, revision 04 (HPI Pump Acceptance Criterion)).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (Inspection Procedure 71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modification to verify it did not affect the safety functions of important safety systems. The inspectors reviewed the temporary modification and the associated 10 CFR 50.59 screening against the system design basis documentation, including the USAR and TSs to determine if there was any effect on system operability or availability and to verify temporary modification consistency with plant documentation and procedures.

- Radiation Element 4598AA (documents reviewed: DB-MI-03413 (Channel Calibration of RE4597AA, 4598AA, 4597BA and 4598BA Normal Range Radiation Monitor), MWO 00-5017-000, Unit logs, TS 3.7.6.1, DB-SC-03216 (Quarterly Functional Test of RE4598AA, Station Vent Normal Range Radiation Monitor), DB-SC-3200 (Shift Channel Check of the Radiation Monitoring System), DB-SS-3041 (Control Room Emergency Ventilation System Train #1 Monthly Test), USAR Sections 6.2, 9.4, 10.4 and 11.4, SD-17 (Process Radiation Monitoring), and SD-32 (Post-Accident Monitoring)).

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (Inspection Procedure 71114.06)

a. Inspection Scope

The inspectors evaluated the conduct of the following drill the licensee had identified as contributing to the drill/exercise and emergency response organization drill participation performance indicators. The inspectors observed the drill to identify weaknesses and deficiencies in classification, notification and protective action requirement development activities, to compare identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee properly identified failures, and to determine whether licensee assessment of performance was in accordance with the applicable criteria.

- March 30 Emergency Preparedness Drill (documents reviewed: Davis-Besse Nuclear Power Station Emergency Preparedness Integrated Drill Manual for March 30, 2001, and NEI 99-02 Rev 0, “Regulatory Performance Indicator Guideline”).

b. Findings

No findings of significance were identified.

OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (Inspection Procedure 71151)

a. Inspection Scope

The inspectors reviewed unit log entries to determine if the performance indicators for safety system unavailability for heat removal systems (AFW) and emergency AC power systems (emergency diesel generators) were accurately and completely reported to the NRC by the licensee. Since this was the first time this inspection activity was conducted for these performance indicators, the previous 14 months of data (January 2000 - February 2000) were inspected.

b. Findings

No findings of significance were identified.

4OA6 Management Meeting

The inspectors presented the inspection results to Mr. G. Campbell and other members of licensee management on April 3, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

KEY POINTS OF CONTACT

D. M. Andrews, Senior Engineer, Plant Engineering
G. R. Bartek, Senior Nuclear Engineer, Plant Engineering
J. M. Baldwin, Shift Supervisor, Plant Operations
W. J. Bentley, Superintendent, Operation Support
H. A. Bergendahl, Plant Manager
K. W. Byrd, Senior Engineer, Nuclear Engineering
G. G. Campbell, Vice President - Nuclear
E. V. Chimahusky, Senior Engineer, Plant Engineering
T. D. Cobbledick, Superintendent of Operations
D. D. Duquette, Engineer, Plant Engineering
G. K. Estep, Lead Nuclear Advisor, Plant Engineering
J. A. Fehl, Senior Engineer, Plant Engineering
J. J. Jiamachello, Senior Engineer, Plant Engineering
E. W. Johnson, Senior Engineer, Plant Engineering
T. A. Lang, Supervisor, Nuclear Engineering
D. J. Lange, Licensed Operator Training Instructor
P. J. Mainhardt, Senior Engineer, Plant Engineering
W. J. Molpus, Senior Engineer, Plant Engineering
R. W. Pell, Manager, Plant Operations
R. I. Rishel, Maintenance Rule Coordinator
C. M. Steenbergen, Assistant Shift Supervisor, Plant Operations
P. F. Timmerman, Senior Emergency Preparedness Specialist
T. A. Thompson, Senior Plant Engineering Advisor

Items Opened and Closed

01-06-01	NCV	Failure to adequately perform risk assessment required by 10 CFR 50.65 paragraph (a)(4) .

List of Acronyms

AFW	Auxiliary Feedwater
DB	Davis-Besse
CR	Condition Report
DRP	Division of Reactor Projects
HPI	High Pressure Injection
MWO	Maintenance Work Order
NRC	Nuclear Regulatory Commission
OS	Operations Schematic
SD	System Description
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
TS	Technical Specification
USAR	Updated Safety Analysis Report