May 1, 2000

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 INSPECTION REPORT 50-346/2000001(DRP)

Dear Mr. Campbell:

On April 1, 2000, the NRC completed an inspection at your Davis-Besse site. The enclosed report presents the results of that inspection.

During this period, the plant was operated safely and in a conservative manner. Engineering personnel conducted a thorough evaluation of a potential common mode failure of the decay heat removal system pumps once plant management was informed of an event that occurred at another nuclear facility. However, all possible accident scenarios were not considered during the initial operability evaluation for a decay heat removal system valve that was found mispositioned.

Based on the results of this inspection, the NRC has determined that two violations of NRC requirements occurred. The first violation concerned operator's identification that valve DH-62 was open when its required position for the mode of operation was to be closed. The second violation concerned an event where an operator manipulated the wrong control rod drive cooling system pump suction valve. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the NRC Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violations or the severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html.

Sincerely,

Original signed by
Thomas J. Kozak, Chief

Thomas J. Kozak, Chief
Reactor Projects Branch 4

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

Enclosure: Inspection Report 50-346/2000001(DRP)

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

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/s/Thomas J. Kozak

Thomas J. Kozak, Chief
Reactor Projects Branch 4

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/2000001(DRP)
Licensee:	Toledo Edison Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 N. State Route 2 Oak Harbor, OH 43449-9760
Dates:	January 26 through April 1, 2000
Inspectors:	K. Zellers, Senior Resident Inspector D. Simpkins, Resident Inspector
Approved by:	Thomas J. Kozak, Chief Reactor Projects Branch 4 Division of Reactor Projects

EXECUTIVE SUMMARY

Davis-Besse Nuclear Power Station
NRC Inspection Report 50-346/2000001(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers the last period before the revised reactor oversight program begins on April 2, 2000.

Operations

- The rate of configuration control problems during the months of January and February more than doubled the rate of configuration control problems per month during the past year (Section O1.2).

- An operator mistakenly closed the wrong control rod drive cooling system booster pump suction valve which resulted in a trip of the booster pump. The pump and cooling flow were recovered shortly and control rod drive mechanism stator temperature only rose approximately two degrees as a result of this error. One Non-Cited Violation was identified (Section O1.2).

Engineering

- The licensee did not evaluate all possible accident scenarios associated with the mispositioning of valve DH-62 during the initial operability evaluation for this condition. After inspector questioning, the licensee determined that operator action would have been necessary to close valve DH-62 to ensure that the low pressure injection system would have performed its design function during a postulated small break loss of coolant accident. The licensee's calculations were reviewed and determined to be appropriate in assumptions used and the conclusions made. The issue is considered have low risk significance. One Non-Cited Violation was identified (Section O1.2).

- Licensee engineering personnel performed a thorough evaluation of the potential for a common cause failure of the decay heat removal system pumps and determined that the pumps at Davis-Besse would perform their safety function during cold weather conditions (Section E1.1).

Report Details

Summary of Plant Status

The plant was operated at nominally 100 percent power throughout the inspection period, except for the following time frames: on March 12, power was reduced to about 92 percent for turbine valve testing; on March 22; reactor power was reduced at a rate of about 1 percent per day due to fuel depletion until March 31 when a plant shutdown was commenced. On April 1, the plant was shut down to commence the 12th refueling outage.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors frequently observed operator performance in the control room and in the plant, attended plant management communication meetings and shift turnover meetings, and reviewed problem reports. Control room operators performed their duties without error. Plant management controlled plant activities and properly prioritized problem reports for resolution. However, some configuration control items were noted during the inspection period (Section O1.2).

O1.2 Negative Plant Equipment Configuration Control Trend

a. Inspection Scope (71707, 37551)

The inspectors followed the guidance in Inspection Procedures (IP) 71707 and 37551 to review the conduct of operations and surveillance testing activities.

b. Observation and Findings

General Observations

The inspectors noted that there was an increase in the rate of plant equipment configuration control issues during this inspection period. Four equipment configuration control issues occurred in January and seven occurred in February. This more than doubled the rate of configuration control issues which occurred per month during the past year. Examples of equipment configuration control problems included valves and switches being found in the wrong position and doors being left open. In addition, an operator failed to identify that a valve was in the wrong position during a valve lineup verification. Two of these equipment configuration control problems affected equipment that is important-to-safety. The first issue involved the licensee's identification that decay heat removal (DHR) system valve DH-62 was open when its required position was to be closed, and the second issue involved an operator closing the wrong control rod drive (CRD) system booster pump suction valve which caused the pump to trip.

Valve DH-62 Found Open

The DHR system functions as the low pressure injection (LPI) system during postulated loss of coolant accidents (LOCAs). At the Davis-Besse plant the DHR pump and LPI pump are different names for the same pump.

Valve DH-62, a normally closed valve, serves to isolate flow from the DHR system to the spent fuel pool and makeup and purification cooling systems. During a routine DHR system surveillance test on February 16, 2000, operators found DH-62 open. Operators immediately shut the valve and Condition Report (CR) 2000-0307 was generated. The licensee did not positively identify the cause or the time when the valve was mispositioned. With valve DH-62 open, some system flow would be diverted from the reactor coolant system during postulated accident scenarios. However, the licensee determined that the diversion of flow from the reactor coolant system degraded the DHR system but did not render it inoperable because the minimum design system flow requirements to the reactor were still maintained.

The inspectors reviewed the operability evaluation for this condition and determined that the licensee had not evaluated all potential accident scenarios associated with the valve being in an open position. Specifically, during a postulated small break LOCA scenario, the LPI system pumps are manually aligned to provide suction pressure to the high pressure injection (HPI) system pumps. In this configuration, with DH-62 open, about 100 gallons per minute would be diverted from the emergency sump to the reactor coolant drain tank (RCDT). Without operator action, enough emergency sump water inventory would be lost such that a loss of suction pressure would occur in both LPI system pumps. This, in turn, would result in a loss of core cooling. However, the RCDT has a high level alarm which activates in the control room. Therefore, it is likely that operators would have identified that DH-62 was open and closed the valve well before a significant water inventory loss occurred thereby ensuring continued core cooling via the LPI/HPI pumps. The licensee generated CR 2000-0603 to document the oversight identified by the inspectors in the operability evaluation.

The licensee added operator action to the probabilistic risk analysis (PRA) logic for small break LOCAs or transient initiating events that require sump recirculation. Final calculations resulted in a core damage frequency (CDF) associated with the failure to close valve DH-62 of 1.6E-7/year. This represents less than a one percent increase in the Davis-Besse CDF of 1.8E-05/year. Because valve DH-62 may have been open for approximately eight months, the conditional core damage probability (CCDP) was calculated to be 1.1E-07. Both results indicate that the change to the total core damage probability is small and would not be considered risk significant.

Criterion V to 10 CFR Part 50, Appendix B, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Procedure DB-OP-06012, “Decay Heat and Low Pressure Injection System Operating Procedure,” Attachment 2, is an instruction used for the valve lineup during operation of the DHR and LPI systems, an activity affecting quality. The instruction required that valve DH-62 be closed during mode 1 operations. Contrary to this, as of February 16, 2000, valve DH-62 was found open during mode 1 operations which is a violation of Criterion V to 10 CFR Part 50, Appendix B. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CRs 2000-0307, and 2000-0603 **(NCV 50-346/2000001-01(DRP))**.

Control Rod Drive Cooling (CRDC) System Booster Pump Trip

The component cooling water (CCW) system provides cooling water to, among other systems, the CRDC system. On February 23, during restoration from a CRDC system booster pump #1 maintenance activity with CRDC system booster pump #2 operating and providing cooling flow to the control rod drive mechanisms (CRDMs), an operator intended to throttle shut CRDC system booster pump #1 suction valve CC-1328; however, he mistakenly started to close CRDC system booster pump #2 suction valve CC-1338, which caused CRDC booster pump #2 to automatically trip. The subsequent degradation of CRDM cooling caused control room annunciator 5-6-D, "CRDC booster pump low flow" to alarm, and the control room operators entered the CCW system malfunction procedure. Within 4 minutes, control room operators determined the cause for the low flow annunciator alarm, opened valve CC-1338 from the control room, and CRDC booster pump #2 automatically started. The CRDM stator temperature increased 2 degrees during the brief loss of cooling and returned to its normal temperature shortly after the booster pump was restarted.

Criterion V to 10 CFR 50, Appendix B, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Clearance 16-04-001A, a procedure used for maintenance activities associated with the CRDC system, an activity affecting quality, required that valve CC-1328 be manipulated to fill and vent CRDC pump #1. Contrary to this, on February 23, an equipment operator closed valve CC-1338, which caused the operating CRD booster pump to trip. The failure to accomplish this maintenance activity in accordance with Clearance 16-04-001A is a violation of Criterion V to 10 CFR 50, Appendix B. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 2000-0365 **(NCV 50-346/2000001-02(DRP))**.

c. Conclusions

The rate of configuration control problems during the months of January and February more than doubled the rate of configuration control problems per month during the past year.

The licensee did not evaluate all possible accident scenarios associated with the mispositioning of valve DH-62 during the initial operability evaluation for this condition. After inspector questioning, the licensee determined that operator action would have been necessary to close valve DH-62 to ensure that the low pressure injection system would have performed its design function during a postulated small break loss of coolant accident. The licensee's calculations were reviewed and determined to be appropriate in assumptions used and the conclusions made. The issue is considered have low risk significance. One Non-Cited Violation was identified.

An operator mistakenly closed the wrong control rod drive cooling system booster pump suction valve which resulted in a trip of the booster pump. The pump and cooling flow were recovered shortly and control rod drive mechanism stator temperature only rose approximately two degrees as a result of this error. One Non-Cited Violation was identified.

O2 Operational Status of Facilities and Equipment

O2.1 System Walkdowns (71707)

The inspectors walked down the accessible portions of the following engineered safety features (ESF) and important-to-safety systems during the inspection period.

- high pressure injection system
- low pressure injection system
- makeup and purification system
- auxiliary feedwater system
- main steam system
- containment spray system
- service water system
- station and instrument air systems

No substantive concerns were identified as a result of the walkdowns.

O8 Miscellaneous Operations Issues (71707))

O8.1 Review of Evaluation of Davis-Besse Nuclear Power Station by Institute of Nuclear Power Operations (INPO)

The inspectors reviewed the subject report and did not identify any significant safety or training issues. No additional NRC follow-up inspections are required.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Observations (62707)

The inspectors reviewed the scheduling and conduct of maintenance activities to determine if overall plant risk was considered. The majority of the time, risk significant activities were evaluated for risk impact. However, on one occasion, the inspectors identified that the risk impact of carryover maintenance on the start-up feedwater system pump was not considered. After further review, the licensee determined that there was no significant risk increase due to the start-up feedwater system pump maintenance continuing into a new work week.

M1.2 Maintenance and Surveillance Activities (61726, 62707)

The following maintenance and surveillance testing activities were observed/reviewed during the inspection period:

- Miscellaneous Instrument Shift Check, DB-OP-03006
- Reactor Protection System Daily Heat Balance Check, DB-NE-03230
- Channel Functional Test and Calibration of Steam and Feedwater Rupture Control System (SFRCS) Actuation Channel 1 Steam Generator Differential Pressure Inputs, DB-MI-03203
- CCW System Relay Replacement, Maintenance Work Order 99-006410-002
- Station Battery Maintenance Guidelines, DB-ME-09200

No deficiencies were identified.

M8 Miscellaneous Maintenance Issues (92700)

M8.1 (Closed) Licensee Event Report (LER) 50-346/97-009-02: Safety Features Actuation System Sequence Logic Channels Surveillance Testing. This is an update to an LER that was discussed in Inspection Report (IR) 50-346/1997004 and closed in IR 50-346/1997009. No safety-significant issues were identified during the review of the additional information in this LER.

III. Engineering

E1 Conduct of Engineering

E1.1 Evaluation of Potential for Common Cause Failure of Decay Heat Removal (DHR) Pumps

a. Inspection Scope (37551)

The inspectors reviewed the licensee's evaluation of a potential common cause failure of the DHR system pumps.

b. Observations and Findings

On February 5, at Arkansas Nuclear 1 (ANO), Unit 1, shortly after initiation of shutdown cooling at the start of a refueling outage, both DHR system pumps became inoperable due to high inboard pump bearing temperatures (reference LER 313-00002). The root cause of the bearing overheating was determined to be high axial loading on the bearing. The bearing housing was modified from the original cast iron material to a stainless steel material in 1992 which had a much higher coefficient of thermal expansion. With about 42 degree water cooling the bearing, and with tight bearing to housing tolerances, the housing caused an interference fit that restricted the movement of the bearing. When the pump shaft started to expand due to 280 degree water going through the pump, an axial load was imparted to the bearing, causing it to generate heat faster than it could be removed. Also contributing to the event was a change to a higher viscosity of oil that had lower heat transfer capabilities.

Davis-Besse has the same model pumps (Hayward Tyler model KSMK 10x12x12), the same size bearings, used the same viscosity oil, and would experience similar environmental conditions to those described above during a LOCA. A significant difference between the pumps used at the two facilities is that the bearing housing used at Davis-Besse is original issue cast iron which has a much lower coefficient of thermal expansion. In addition, at Davis-Besse, under normal conditions, the bearings are cooled by CCW at about 90 degrees. During the injection phase of a LOCA, CCW temperatures could reach about 38 degrees under the most severe environmental conditions. When the decay heat pump water supply would be changed to the emergency sump, hot coolant would go through the pump; however, CCW would heat up to greater than 50 degrees within 8 minutes.

The licensee developed an analytical model to simulate the effects on the pump for postulated scenarios down to initial bearing temperatures of about 50 degrees and determined that the pump would continue to operate. The licensee also hired a pump contractor who had been involved with the root cause investigation at ANO to assist in the evaluation. The results of the evaluation were that the pumps would reasonably be expected to operate on a continuous basis for all expected CCW temperatures. However, the licensee changed the bearing oil to a lower viscosity oil to provide greater heat transfer ability, and performed a test of both decay heat pumps during a plant shutdown for the 12^{th} refueling outage with CCW water temperature to the bearings adjusted to about 50 degrees. During the test, both pump inboard bearing temperatures stabilized at less than 90 degrees at about 50 degrees CCW temperature, which demonstrated that the pumps would operate long term at the lowest expected CCW temperature during the recirculation phase of a postulated LOCA.

c. Conclusions

Licensee engineering personnel performed a thorough evaluation of the potential for a common cause failure of the decay heat removal system pumps and determined that the pumps at Davis-Besse would perform their safety function during cold weather conditions.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on March 23, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. E. Ackerman, Supervisor, Quality Assessment
M. C. Beier, Manager, Quality Assessment
A. W. Bless, Assistant Engineer, Regulatory Affairs
K. Byrd, Risk Analyst
T. J. Chambers, Supervisor, Quality Assurance
R. B. Coad, Jr., Manager, Plant Operations
S. A. Coakley, Supervisor, Outage Management
L. M. Dohrmann, Manager, Administrative Services
D. L. Eshelman, Manager, Plant Engineering
K. A. Filar, Senior Engineer, Chemistry
J. L. Freels, Manager, Regulatory Affairs
D. C. Geisen, Manager, Technical Services
D. H. Lockwood, Supervisor, Compliance
J. Messina, Director, Work Management
J. L. Michaelis, Manager, Maintenance
S. P. Moffitt, Director, Technical Services
B. H. Robinson, Nuclear Evaluator, Quality Assessment
J. W. Rogers, Manager, Maintenance
P. Shultz, Manager, Radiation Protection
G. A. Skeel, Manager, Security
H. W. Stevens, Jr., Manager, Nuclear Safety & Inspections
M. Stevens, Manager, Work Management
J. M. Vetter, Supervisor, Quality Assessment
G.M. Wolf, Engineer, Regulatory Affairs
L. W. Worley, Director, Support Services

NRC

S. Burgess, Senior Reactor Analyst
T. J. Kozak, Chief, Reactor Projects Branch 4
K. S. Zellers, Senior Resident Inspector, Davis-Besse
D. S. Simpkins, Resident Inspector, Davis-Besse

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 61726: Surveillance Observations
IP 62707: Maintenance Observation
IP 71707: Plant Operations
IP 92700: Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities

ITEMS OPENED AND CLOSED

Opened

50-346/2000001-01	NCV	10 CFR 50 Appendix B, Criterion V procedure violation. Valve DH-62 found mispositioned.
50-346/2000001-02	NCV	10 CFR 50 Appendix B, Criterion V procedure violation. Operator closed the wrong valve causing CRD booster pump trip.

Closed

50-346/2000001-01	NCV	10 CFR 50 Appendix B, Criterion V procedure violation. Valve DH-62 found mispositioned.
50-346/2000001-02	NCV	10 CFR 50 Appendix B, Criterion V procedure violation. Operator closed the wrong valve causing CRD booster pump trip.
50-346/97-009-02	LER	Safety Features Actuation System Sequence Logic Channels Surveillance Testing

LIST OF ACRONYMS AND INITIALISMS USED

ANO	Arkansas Nuclear 1
CCDF	Conditional Core Damage Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
CRDC	Control Rod Drive Cooling
CRDM	Control Rod Drive Mechanisms
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
HPI	High Pressure Injection
I&C	Instrumentation and Controls
IFI	Inspection Followup Item
INPO	Institute of Nuclear Power Operations
IR	Inspection Report
LER	Licensee Event Report
LOCA	Loss-of-Coolant-Accident
LPI	Low Pressure Injection
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PERR	Public Electronic Reading Room
PRA	Probabilistic Risk Assessment
RCDT	Reactor Coolant Drain Tank
SBLOCA	Small Break Loss-of-Coolant-Accident
SFRCS	Steam and Feedwater Rupture Control System
TS	Technical Specification
USAR	Updated Safety Analysis Report
VIO	Violation