

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-346/95007

FACILITY

Davis-Besse Nuclear Power Station  
License No. NPF-3

LICENSEE

Toledo Edison Company  
300 Madison Avenue  
Toledo, OH 43652


DATES

June 26, 1995 through August 23, 1995

INSPECTORS

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APPROVED BY

  
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9/28/95  
\_\_\_\_\_  
Date

AREAS INSPECTED

A routine, unannounced inspection of operations, maintenance, engineering, plant support, and preparations for dry cask storage of spent fuel was performed. Safety assessment and quality verification activities were routinely evaluated. Followup inspection was performed for certain previously identified items.

## RESULTS

### Assessment of Performance

Performance within the area of **OPERATIONS** was excellent overall. Control room activities continue to be conducted in a controlled conservative manner. Adherence to procedures was very good with no substantive concerns noted. Operators were attentive to their panels and remained generally cognizant of system and equipment status. However, operators were unaware that some plant electrical equipment was operating at voltage levels that slightly exceeded plant approved limits (Section 1.2). Also, a non-cited violation was identified where the interim configuration of the containment hydrogen analyzer was inadequate (Section 1.4). A unit transient was caused by manipulation of a supposedly disabled circuit on the control room electro-hydraulic control (EHC) panel (Section 1.3). Further NRC review of the process used to disable the circuit is warranted.

Performance within the area of **MAINTENANCE** was very good overall. Maintenance and surveillance activities reviewed this inspection period were typically conducted in accordance with licensee procedures and regulatory requirements. One exception was a failure to follow a procedure on the part of electrical maintenance personnel that resulted in a safety-related service water valve being left in a degraded condition. This matter was determined to be a violation (Section 2.2).

Performance within the area of **ENGINEERING** was very good overall. Engineering support for the day-to-day operation of the plant was determined to be generally excellent. Engineering activities related to the spent fuel dry cask storage project continued to progress in an orderly fashion (Section 5.0). However, engineering followup of concerns related to service water system temperature monitoring was initially slow (3.1).

Performance within the area of **PLANT SUPPORT** was excellent. Personnel adherence to the radiation protection and security programs was excellent. Security protection for the dry cask storage area was adequate. Additional security measures, if used for cask movement to the storage area, may require inclusion in an appropriate plan or procedure. Reduction in the security door false alarm rate continued to offer a challenge to the security staff. Onsite emergency facilities continued to be well maintained in a good state of readiness.

Performance within the area of **SAFETY ASSESSMENT/QUALITY VERIFICATION** was excellent. The Station Review Board and Company Nuclear Review Board provided good assessment of technical issues with participants actively engaged in those discussions (Section 3.3). Plant management was visible in the plant equipment spaces as well in the control room on a routine basis (Section 1.1).

### Summary of Open Items

Violations: Identified in Section 2.2

Unresolved Items: Identified in Sections 1.2 and 3.1

Inspector Follow-up Items: Identified in Sections 1.3, 3.2, 4.2, and 5.6

Non-cited Violations: Identified in Section 1.4

## INSPECTION DETAILS

### 1.0 OPERATIONS

NRC Inspection Procedure 71707 was used in the performance of an inspection of ongoing plant operations. No violations or deviations were identified and overall performance in this area was considered excellent. However, a concern related to the control of electrical equipment operating voltage limits was identified that will necessitate further review. Also, a non-cited violation was identified relating to inadequate control of a containment hydrogen analyzer interim equipment configuration that occurred during the previous inspection period.

#### 1.1 Control Room Activities Were Performed Well.

Control room activities continued to be well controlled and conducted in a conservative manner. Operators' adherence to procedures was very good with no substantive concerns noted. Control room personnel were cognizant of panel indication and equipment status. Shift briefs adequately communicated necessary information to the oncoming shift.

Plant management remained involved in the day-to-day operation of the unit with routine visits to the control room and plant tours being made on a continuing basis. During this inspection period, an initiative was begun where one reactor operator was assigned primary responsibility for monitoring control room panels at any one time with no additional ancillary duties. This position was rotated between the reactor operators during their shift.

#### 1.2 Electrical Equipment Operating at High Voltage Conditions.

While observing plant electrical panels, the inspector noted that certain equipment was operating at higher than nominal voltage levels. In particular, essential inverter YRF3 as well as non-essential busses YAU and YBU were determined to be operating at voltages that exceeded either their nameplate rating and/or higher than analyzed voltage limits.

In 1987, the licensee had prepared and issued Request for Action (RFA) 88-1042 establishing acceptable voltage operating bands which were consistent with ANSI/IEEE Standard 141-1986. However, although those voltage limits were defined by the RFA, ANSI/IEEE standards, and specified in vendor manual literature for the equipment, as well as the equipment nameplate rating, YRF3 as well as YAU and YBU were found to be operating at voltage levels higher than analyzed. The licensee was evaluating this issue and extent of condition, and was establishing what corrective actions if any may be necessary to better control voltage operating conditions. Potential Condition Adverse to Quality Report (PCAQR) No. 95-0676 was initiated to ensure resolution of this issue. YRF3 was subsequently readjusted to a lower operating voltage. Thus, no specific safety-related equipment concern existed at the conclusion of the inspection period. Pending completion of the licensee actions on

this matter and subsequent NRC review of those actions, this is considered an unresolved item (50-346/95007-01(DRP)).

1.3 Power Transient Due to EHC Failure.

On August 7, with the plant at full power, the unit experienced a 15 percent power transient that occurred over an approximate 45-second period. During performance of a lamp test of the control room electro-hydraulic control (EHC) panel, one of the indicating lights on a portion of the panel failed the lamp test. This particular portion of the panel was a main turbine inlet throttle pressure limiter which had previously been abandoned in place by an engineering change incorporated in 1987. The light that failed the lamp test was the "on" light for that control circuit. A decision was made by shift personnel to replace the faulty light bulb. While replacing the light, the turbine control valves started to rapidly shut initiating the plant transient. The operator took immediate action by pressing the "off" button for the limiter, and the turbine control valves thereafter reopened to their normal position. As a result of the turbine control valve rapidly shutting, a primary to secondary heat imbalance caused reactor pressure to increase from a nominal value of 2155 to 2245 psig. Steam generator pressure increased from a nominal value of 870 to 950 psig causing the turbine bypass valves to open. The duration of the transient, from initiation until plant parameters returned to normal values, was about 45 seconds. All primary and secondary systems responded to the transient as designed.

Since it was found that the throttle pressure limiter control push buttons were not fully disabled, "Do Not Operate" information tags were placed on the control panel to identify that fact. In addition, at the conclusion of the inspection period, the licensee was evaluating further followup actions that may be necessary to adequately address this situation. A review was also being conducted to establish how the Field Change Request (FCR) had been implemented. Pending completion of that review, this matter is considered an inspection follow-up item (50-346/95007-02(DRP)).

1.4 Follow-up on Previously Opened Items A review of previously opened items (violations, unresolved items, and inspection follow-up items) was performed per NRC Inspection Procedure 92901.

(Closed) Unresolved Item (50-346/95005-01(DRP)): Containment Hydrogen Analyzer door found unlatched and partially open. Although the door latch could only be engaged by use of a screwdriver or similar tool, the door was not tagged with either an operational information tag (OIT) or a material deficiency tag. The absence of tags appeared to contribute to some confusion on the part of maintenance personnel as to the acceptability of leaving the door unlatched/open.

Administrative procedure DB-OP-00010, Operational Information Tags, specified in Section 1.2.1 that an OIT was to be used to provide precautionary information regarding the operation or status of a device, component, or system. Section 4.1 specified again that an OIT be



installed strictly as a source of information regarding the operation of a device, component, or system. Section 6.1.3.a.1 also specified that for equipment in need of repair, a material deficiency tag be placed, if the equipment is accessible, in accordance with procedure DB-PN-00007, Control of Work. Since neither an operational information tag nor a material deficiency tag was placed, this is considered a violation of administrative procedure DB-OP-00010. However, based on the subsequent engineering evaluation that determined that no adverse safety consequences resulted from the door being open, and licensee followup actions being adequate to return the door latch to service, this matter met the criteria specified in Section IV of the "General Statement of Policy and Procedures For NRC Enforcement Actions, and a Notice of Violation were not issued.

## 2.0 MAINTENANCE

NRC Inspection Procedures 62703 and 61726 were used to perform an inspection of maintenance and testing activities. Overall, performance in this area was very good. Maintenance and surveillance activities reviewed this inspection period were typically conducted in accordance with licensee procedures and regulatory requirements. However, VOTES testing of a service water valve was not appropriately accomplished and is considered a violation of the electrical maintenance procedure used for the activity. In addition, a non-cited violation was identified during followup of an issue from the previous inspection period.

### 2.1 Maintenance Work Order (MWO) Procedural Concerns During Emergency Diesel Generator Preventive Maintenance

While observing preventive maintenance activities associated with Emergency Diesel Generator (EDG) No. 1, the inspector noted that steps in the maintenance work order were being completed out of sequence from that specified, and that the signoffs for completed steps were not done at the time the particular steps were completed. Subsequent review determined that no restrictions limiting workers from performing MWO steps out of sequence existed nor were signing off steps after the fact prohibited. Discussions with maintenance management resulted in management expectations being developed and communicated to the craft specifying that maintenance work order steps be completed in sequence if required (and specified in the MWO), as well as step signoffs to be completed immediately following completion of each step. Revisions to several associated administrative procedures were in process at the end of the inspection period. Completion of these actions adequately addressed inspector concerns on this matter.

### 2.2 Service Water Valve Not Properly Restored Following Completion of VOTES Testing

During a walkdown of the Service Water system, the inspector noted that the limitorque operator limit switch compartment cover on valve SW 1399 was not properly installed. The bolts used to attach the compartment to the limitorque housing had not been properly engaged and had resulted in

an up to 1/2 inch gap between the cover and the housing. Additionally, at least two of the bolts had not been installed. Subsequently it was determined that SW 1399 had just undergone VOTES testing the same day, operations was preparing to return the valve to service, and no further inspection of the valve was planned prior to declaring it operable. Since the limit switch compartment was designated as a spray shield under the licensee's program to protect the internals from water intrusion, the compartment's as-left condition adversely affected the valve's operability.

Electrical maintenance procedure DB-ME-09301, Preventive Maintenance For Type SMB and SMC Limitorque Valve Operators, was used to perform PMs and VOTES testing of SW 1399. Step 8.3.13, was erroneously signed off during restoration from testing indicating that the cover had been adequately reinstalled.

Since the procedural step was signed off as complete even though the activity had not been adequately accomplished, and would have resulted in SW 1399 being left in a degraded condition if the degraded condition had not been identified by the NRC, this is considered a violation of 10 CFR Part 50, Appendix B, Criterion V (50-346/95007-03(DRP)).

Actions were immediately initiated to reinstall the cover, and a PCAQR (No. 95-0605) initiated to track resolution of this issue.

### 3.0 ENGINEERING

NRC Inspection Procedure 37551 was used to perform an onsite inspection of the engineering function. Engineering support for the day-to-day operation of the plant was determined to be generally excellent. No violations or deviations were identified. However, an unresolved item was identified relating to the service water intake temperature monitor.

#### 3.1 Service Water System Intake Temperature Monitoring Problems

During walkdowns of the Service Water (SW) system piping within the service water tunnel area of the plant in late July, the inspector noted that two local temperature indicators monitoring SW inlet temperature, although indicating consistent readings between themselves, were approximately 4 degrees higher than that provided by plant computer point T413. T413 measured intake forebay temperature and was used to monitor for compliance with the Technical Specification 85°F limit. The inspector was concerned that the accuracy of T413 may not have been adequate to ensure compliance to the Technical Specifications.

A detailed licensee review of the accuracy of the T413 indication was promulgated based upon, in part, NRC questions concerning the divergent indications, and as part of an additional system review due to increasing SW inlet temperatures that were approaching the Technical Specification limit. Several issues resulted: 1) T413 was not included in the licensee's calibration program; 2) The output signal from monitor

3) T413 was placed in such a manner as to allow it to be partially or completely covered by silt buildup; 4) T413 at one time apparently had been part of the calibration program but at some point had been removed from the program.

Once the problems associated with T413 were identified, appropriate compensatory measures were initiated. Attempts were made to recalibrate the T413 string, adequacy of the two local indicators were verified, and a high accuracy temperature monitor was placed in one of the local SW temperature monitor wells to monitor SW and UHS temperatures in lieu of using T413.

However, as described above, several issues remained pending at the end of the inspection period. Pending resolution of those issues, this is considered an **unresolved item (50-346/95007-04(DRP))**.

### 3.2 Containment Air Cooler (CAC) 1-3 Lineup/Testing Issues

During review of a CAC 1-3 lineup, the inspector noted that during realignment of the CAC to either train 1 or train 2 power, the DC power supply alignment was not directly indicated. CAC 1-3 was considered an installed spare which could be aligned as either train via manipulation of four AC breaker switches. When this was done, auxiliary contacts were also supposed to automatically realign the DC supply power to the appropriate train. Plant engineering personnel indicated that periodic testing and inspection of the breakers were sufficient to verify that the DC supply contacts were properly functioning. These inspection activities occurred during breaker preventive maintenance activities conducted once every 18 months.

Current operability of the CAC was not at issue during this review. Rather the acceptability of the testing practices to verify continued operability was in question. At the end of the inspection period, NRC review as to the acceptability of those testing practices was ongoing. This matter is considered an **inspection followup item (50-346/95007-05(DRP))** pending completion of that review.

### 3.3 Station Review Board (SRB) and Company Nuclear Review Board (CNRB)

The inspector attended several SRB (onsite review committee) meetings and a CNRB (offsite review committee) meetings during the inspection period. Participants were observed to be actively involved in the discussions held during those meetings. The specific technical issues appeared to be adequately addressed in all cases. No substantive concerns were noted.

### 3.4 Follow-up on Previously Opened Items NRC Inspection Procedure 92903 was used to perform a review of previously opened items (violations, unresolved items, and inspection follow-up items).

**(Closed) Violation (50-346/94016-01(DRS)):** The check valve modifications made on the auxiliary crossover main steam system lacked a

documented safety evaluation. NRC management reviewed the licensee's response to the violation, dated March 10, 1995, and discussed the issue with the licensee. Subsequently, the violation was rescinded.

(Closed) Open Item (50-346/92010-01(DRS)): The licensee lacked technical justification for grouping MOVs for the purpose of DP testing. Grouping was to be performed using a mathematical computer model capable of predicting the required stem thrust at any stroke position. The computer model had undergone verification and validation and the program was available for use. This item will be further evaluated per TI 2512/109 in an upcoming inspection. This item is closed.

#### 4.0 PLANT SUPPORT

NRC Inspection Procedures 71750 and 83750 were used to perform an inspection of Plant Support Activities. Overall, performance in this area continued to be excellent. No violations or deviations were identified.

##### 4.1 Radiological Controls

Personnel adherence to the radiation protection (RP) program continued to be excellent this inspection period. Personnel use of dosimetry and the radiation work permit process was very good as well. RP survey documentation that was reviewed adequately reflected actual plant conditions with no substantive discrepancies noted by the inspectors.

##### 4.2 Security and Safeguards

Plant personnel adherence to the security program continued to be excellent this inspection period. Security department self assessment efforts were varied, offered flexibility, and appeared adequate to identify adverse trends. Maintenance support for security equipment was generally very good with respect to timeliness of repair and reduction of compensatory measures for equipment failure. Security procedures were excellent in scope, depth, and detail. Appropriate reviews were performed and procedures reflected the work practices observed.

During the inspection period, it was noted that the security staff had been successful in meeting most goals established for trend analysis and other performance indicators. However, one exception was in reducing the number of security door false alarms and the necessary response to such alarms. Goal attainment for this performance indicator was not achieved for the past 18 months. The current goals established by the security staff appeared realistic based on system operational performance over the past 5 years. This matter is considered an inspection followup item (346/95007-06(DRSS)) pending further inspector review of this matter.

The inspector also noted that a security procedure allowed alarm station operators excessive time to acknowledge a certain type of alarm that could adversely affect assessment capability for subsequent alarms (The



details of the alarm system, assessment capabilities, and time involved are considered safeguards information and exempt from public disclosure. The details have been discussed with the licensee staff). The time limit identified in the procedure exceeded the time needed for initial assessment purposes and if the time to acknowledge an alarm identified in the procedure was consistently used, assessment capability for subsequent alarms could be unnecessarily degraded. Subsequent to the inspection, the licensee revised the subject procedure to limit the particular alarm response times. The inspector had no further concerns on this matter.

#### 4.3 Emergency Preparedness

Tours of the onsite emergency response facilities were periodically conducted during the inspection period. In all cases, the facilities appeared to be well maintained and in a good state of readiness.

On July 19, and August 16, the inspector observed portions of the licensee's emergency preparedness drills from the technical support center (TSC) and the emergency control center (ECC). No substantive concerns were noted.

#### 4.4 Follow-up on Previously Opened Items NRC Inspection Procedure 92904 was used to perform follow-up inspection of the following item:

**(Closed) Inspection Followup Item (50-346/95002-01(DRSS)):** This item pertained to a request that the licensee provide the NRC a copy of any investigative report they completed in reference to the granting of unescorted access in October 1994 to an individual before all requirements for such access were completed. The licensee had not prepared a consolidated formal investigative report of the incident. However, the documents pertaining to the incident and requested by the NRC during an onsite visit between May 10-12, 1995, have been provided to the NRC for review. The documents pertaining to the concern are still being reviewed. This item is considered closed.

**(Open) Unresolved Item (50-346/95002-02(DRSS)):** This item pertained to "ascertaining" the activities of some personnel who had been granted unescorted access and had not been under a behavior observation program (BOP) for periods of 30 days or more. This issue was still being reviewed by the NRC and is considered an open item until resolved. No additional actions pertaining to this issue are required by the licensee's staff.

#### 5.0 DRY CASK STORAGE OF SPENT FUEL

Inspections were performed of onsite activities conducted in association with the project for dry cask storage of spent fuel. Overall, performance in this area was good. Onsite activities appeared to be progressing in an acceptable manner. Licensee oversight of vendor activities had been substantial. No violations or deviations were identified.

5.1 Dry Shielded Canister Radiographs Were Acceptable.

The inspector examined a sampling of radiographs of welds from the dry shielded canisters (DSC) that had been made at the vendor's facility and forwarded to the licensee for review and approval. Sample size consisted of radiographs of one pressure retaining weld on each of the three DSCs. Based on the sample size, the inspector determined that the radiographs were acceptable and the weld quality met code requirements.

5.2 Storage Pad Concrete Testing Was Satisfactory.

The inspector reviewed results of concrete testing for the dry fuel storage pad. Slump tests performed during concrete placement were acceptable and within design specifications. Air entrainment ranged between 3.3 and 9.5 percent. Specifications were 3 to 6 percent air. However, checks of compression strength test records indicate that break tests exceeded the 4000 psi design strength. Therefore, although air entrainment of the concrete slightly exceeded the established specification, overall strength of the concrete remained acceptable.

5.3 Review of Licensee Audits Conducted at Vendor Facilities

The inspector reviewed licensee audit reports relative to audits performed at VECTRA and sub-vendor facilities. Audit AR-95-VECTRA-03 performed at VECTRA, San Jose, CA and Utility Vault Company, Pleasanton, CA identified an audit finding that the identification, documentation, and implementation of process changes were not controlled. Examples were:

Certificate of Compliance 1004 required changes, tests, and experiments to be evaluated in accordance with Section 9 of the Certificate of Compliance. Procedures for complying with Section 9 had not been issued.

The Engineering Change Notices (ECN) for 23 transfer cask ECN's had not been evaluated. No plans were in place to perform these evaluations.

At the facilities of the vendor supplying the Horizontal Storage Modules (HSM) the licensee's auditors identified the following:

The HSM vendor was not complying with all specification requirements. Concrete material changes were made without revising the specifications.

Exception was taken to the American Concrete Institute (ACI) standard for curing and protecting concrete without documented justification.

The water bath for curing concrete test specimens was not controlled to ensure water temperature was maintained as specified in ASTM C31.

Exposed surfaces of roof slabs, before and after form removal, had not been protected against moisture loss.

Use of a fibrous admixture in the concrete that was not part of the approved mix design.

Changing aggregate size from specified size.

Certificates for aggregate were not current and did not contain all the required test results.

Form removal and moving concrete via lifting lugs prior to the concrete attaining minimum strength required by specifications.

The licensee was continuing to followup with the appropriate vendor facilities to ensure the identified issues were adequately resolved. Additional NRC inspection will also be conducted to assure the subject issues were adequately addressed.

5.4 Onsite Roadway Modifications Adequately Completed

The inspector performed a walkdown inspection of roadway modifications between the spent fuel pool and the storage pad. The licensee had tested the roadway subbase, resurfaced unacceptable areas, replaced old culverts, and built and surfaced sections of road between the existing road and the concrete storage pad.

5.5 Auxiliary Building Crane Load Test Was Completed Satisfactorily.

The inspector observed auxiliary building bridge crane testing and reviewed associated test results. The crane lifted a load of 134 tons, held the load suspended for 10 minutes, then transversed the trolley with the load. Based on the test results reviewed and the load of a full transfer cask, the crane appeared to perform satisfactorily.

5.6 Security Program For Cask Storage Area Was Adequate.

Security protection for the Spent Fuel Dry Cask Storage (SFDCS) area was reviewed and was adequate. The level of protection for the SFDCS area was equivalent to protected area protection. Existing procedures addressed contingencies that could constitute a security threat to the SFDCS area.

At the end of the inspection period, the licensee's staff was evaluating the need for additional specific security measures during cask movement to the storage area. Pending further NRC review, this matter is considered an inspection followup item (50-346/95007-07(DRSS)).

## 6.0 PERSONS CONTACTED AND MANAGEMENT MEETINGS

The inspectors contacted various licensee operations, maintenance, engineering, and plant support personnel throughout the inspection period. Senior personnel are listed below.

At the conclusion of the inspection on August 23, 1995, the inspectors met with licensee representatives (denoted by \*) and summarized the scope and findings of the inspection activities. The licensee did not identify any of the documents or processes reviewed by the inspectors as proprietary.

- \*J. P. Stetz, Vice President, Nuclear
- \*S. C. Jain, Director, Nuclear Services
- \*J. K. Wood, Plant Manager
  - T. J. Myers, Director, Nuclear Assurance
- \*J. W. Rogers, Manager, Maintenance
- \*B. Donnellon, Manager, Plant Engineering
  - J. E. Moyers, Manager, Quality Assessment
- \*J. Michaelis, Manager, Nuclear Support
  - D. Crouch, Superintendent, Mechanical Maintenance
- \*D. P. Ricci, Supervisor, Operations
  - P. Smith, Supervisor, Compliance
  - J. Barron, Supervisor, Test/Performance
  - D. Schreiner, Supervisor, ISE
- \*A. VanDenabeele, Supervisor, Quality Analysis
- \*J. Syrowski, Supervisor, EP
- \*R. C. Zyduck, Manager, Nuclear Engineering
- \*G. Skeel, Manager, Security
  - W. T. O'Connor, Manager, Regulatory Affairs
- \*R. Scott, Manager, Radiation Protection
  - J. Lash, Manager, Plant Operations
- \*L. M. Dohrmann, Manager, Quality Service
- \*C. Hawley, Manager, NSI
- \*D. Eshelman, Superintendent, OPS
- \*N. Peterson, Sr. Engineer, Licensing
- \*M. Borysiak, Supervisor, DEEC
- \*G. W. Gillespie, Supervisor, Chemistry
- \*R. B. Coad, Supervisor, RP
- \*K. C. Prasad, Supervisor, Safety Engineering
- \*P. Mainhardt, Senior Engineer
- \*G. A. Bradley, Nuclear Specialist
- \*A. V. Antrassian, Engineer, Licensing
- \*M. Leisure, Sr. Engineer, Licensing
- \*M. Riemer, Professional Practice Student