U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-346/78-01

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

Licensee: Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, OH 43652

Facility name: Davis-Besse Unit 1

Inspection at: Davis-Besse Site, Oak Harbor, OH

Inspection conducted: December 13-15, 1977 and January 11-13, 1978

Inspector: T. N. Tambling 1/27/78

Approved by: R. C. Knop, Chief 1/27/78
Reactor Projects Section 1

Inspection Summary

Inspection on December 13-15, 1977 and January 11-13, 1978 (Report No. 50-346/78-01)

Areas Inspected: Review of licensee event reports, review of design, design changes and modifications, witnessing of a transient test of 40% power and followup on licensee's evaluation on Auxiliary Feedwater Pump problems. The inspection involved 56.5 inspector-hours onsite by one NRC inspector.

Results: Of the four areas inspected, no items of noncompliance were found in three areas; one apparent item of noncompliance was found in one area (infraction - failure to properly implement and maintain a procedure - Para. 3).

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DETAILS

1. Personnel Contacted

*T. Murray, Station Superintendent
B. Beyer, Maintenance Engineer
*L. Stalter, Technical Engineer
L. Grime, Reliability Engineer
*J. Buck, Operations Quality Assurance Engineer
*W. Green, Assistant to Station Superintendent
*T. Hart, Quality Assurance Engineer
*W. Schultz, Power Engineering and Construction
C. Daft, Quality Control Supervisor

The inspector also talked with and interviewed other licensee employees, including members of the technical and engineering staff, operations staff and radiation protection.

*Denotes those attending exit interview.

2. Review of Nonroutine Events Reported by the Licensee

The inspector reviewed licensee actions with respect to the following listed nonroutine event reports to verify that the events were reviewed and evaluated by the licensee as required by Technical Specifications, that corrective action was taken by the licensee, and that safety limits, limiting safety system settings, and limiting conditions for operation were not exceeded. The inspector examined selected Station Review Board minutes, the licensee investigation reports, logs, and records, and inspected equipment and interviewed selected personnel.

Inadvertant boron dilution due to improper adjusted control valve (NP-33-77-02)

Loss of Decay Heat flow during surveillance test of SFAS (NP-32-77-05)

Momentary loss of core flow in Mode 4 due to valve switching sequencing problem (NP-32-77-06)

Failure of Decay Heat check valve DH-77 to close when Core Flood Tank was unisolated (NP-32-77-08)

Both Auxiliary Feedwater Systems inoperable (NP-32-77-11)

Electrolyte level above maximum level indication mark on 125 volt batteries (NP-32-77-13)

Loss of Makeup Pumps due to blown control circuit fuse (NP-32-77-14)

Cooling water to Auxiliary Feedwater Pump bearing improperly aligned (NP-32-77-17)

Auxiliary Feedwater Pump turbine governor valve vibrated closed (NP-32-77-18 and Part 21 report, L. E. Roe to J. G. Keppler, RIII dated November 16, 1977)

Inadvertant SFAS actuation in Mode 5 coupled with loss of 120 VAC Y2 bus (NP-33-77-06)

Failure of two control room emergency dampers to meet response time during testing (NP-33-77-09)

Loss of SFAS channel 1 due to loss of Y1 distribution panel (NP-33-77-14)

Containment internal pressure not being monitored (NP-33-77-16)

Emergency Ventilation System 1-1 inoperable (NP-33-77-34)

Electrolyte level in station batteries high (NP-33-77-29)

Containment personnel air lock interlock inoperable (NP-33-77-35)

Steam generator level indication outside differential tolerance (NP-33-77-44)

Decay Heat isolation valves DH-11 and DH-12 declared inoperable (NP-33-77-50)

The inspector noted that the licensee had identified and corrected four items related to these events.

No other items of noncompliance or deviations were identified.

During the exit interview the inspector requested supplemental reports for NP-32-77-18, NP-33-77-06 and NP-33-77-14 to cover the final corrective actions, concurred with the licensee's interpretation the post-LOCA flow path through DH11 and 12 (NP-33-77-50), discussed interpretations regarding surveillance testing and noted improvement in the closing out of recent DVR's associated with non-routine reports.

The following licensee event reports were reviewed and closed out on the bases of an inoffice review and evaluation:

Loss of Shield Building integrity (NP-33-77-32)

Containment isolation valve CV-5074 inoperable (NP-33-77-38)

Main Steamline hydraulic snubber SR17 and SR11 inoperable (NP-33-77-70)

Makeup Pump 1-2 removed from service to allow maintenance (NP-33-77-73)

Decay Heat Valve Pit opened to perform wiring change (NP-33-77-77)

RPS channel 1 accidently de-energized (NP-33-77-78)

Governor valve closed on AFP 1-2 (NP-33-77-80)

EVS train 1-2 delta pressure controller malfunction (NP-33-77-94)

Diesel generator trip on overspeed (NP-33-77-96)

Control Room isolation damper would not completely close (NP-33-77-95)

Failure of AF 3872, Stop valve between AFT 1-2 and Steam Generator (NP-33-77-83)

3. Reduction of Discharge Cycle Limits on Pressurizer Relief Valves, LER NP-32-77-12

The inspector reviewed the licensee's report dated August 8, 1977 to determine whether the corrective action was being implemented. Within this review on December 14, 1977 the inspector found that a log of operational transients was being maintained in the control room. However,

a. An unapproved copy of AD 1839.01, Documentation of Allowable Operating Transient Cycles, was being used in the field for implementation.

b. The actuation of the pressurizer relief valve RC-2A on September 24, 1977, had not been logged.

c. The current revision (Revision 1) of AD 1839.01 limitation for the number of cycles and temperatures limits on the pressurizer relief valves was not in agreement with PT 5164.03, (Revision 0) Pressurizer Power Relief Valve Periodic Test.

The failure to use an approved copy of AD 1839.01 for field implementation, to log the transient cycle on September 24, 1977, and to revise AD 1839.01 to reflect current limitations is considered to be an item of noncompliance with the requirements of Section 6.8.1 of the Technical Specification.

The licensee immediately replaced the unapproved copy of AD 1839.01 with an approved copy. In the exit interview the licensee committed to revise AD 1839.01 to reflect the correct limits, review the transient log to insure that items are being recorded properly, to record the temperature at the time of pressurizer relief valves actuations, and to evaluate how they will assess pressurizer relief valve actuations at different temperatures.

4. Containment Isolation Valve RC 240A Inoperable, LER NP-33-77-40

The inspector reviewed the licensee's report dated August 26, 1977, to determine the status of the corrective action. Although the initial corrective action had been completed, DVR 74-1 had not been closed out and the investigation by the architect/engineer was not available for review by the inspector. This item remains unresolved pending the close out of the DVR and the investigation.

5. Circular 77-13, Reactor Safety Signals Negated During Testing

The inspector reviewed with the licensee his conclusion and actions on Circular 77-13. The licensee had concluded that their procedures and management control systems were adequate to prevent a similar occurrence, however, the referenced incident was being used as a basis of a special training secession for operating and plant personnel to demonstrate the possible consequences of not adhering to procedures and controls.

6. <u>Design, Design Changes and Modifications</u>

The inspector reviewed implementing procedures and selected Facility Change Request forms to determine whether changes to the facility were made in accordance with 10 CFR 50.59, the Technical Specifications and established QA/QC and administrative controls. The review included the following implementing procedures.

Quality Assurance Procedure 2030, Design Control.

Administrative Procedure 1845.00, Changes, Test and Experiments, Rev. 0.

Power Engineering Instruction DBI, 320, Design Changes, Tests, and Experiments, Rev. 3.

Power Engineering Instruction DBI-334, Safety Review/Evaluation/ Accident Analysis, Rev. 1.

Power Engineering Instruction DBI-331, Work Package, Rev. 1.

At the exit interview the inspector discussed several areas in AD 1845.00 that should be clarified. These were:

Section 2.3 - The processing of non-nuclear safety related design changes should be defined.

Section 4.2 - The definition of Nuclear Safety Related should be expanded to includ: other chapters of the FSAR. 10 CFR Part 50.59 addresses the FSAR and not only Chapter 15. Also, ASME items are not addressed.

Section 3 - Section 6.5.1.6 of the Technical Specifications states that SRB will review tests, experiments, changes or modifications that affect nuclear safety. Nuclear safety as used there has a much broader scope than implied in section 4.2.

Section 6.9 and 6.10 - These sections need to be clarified as to who is responsible for preparation of test procedures, the information required in the procedures and the format to be used.

Section 9.5 - This section should be clarified that a delegated representative can act for the Project Engineer to insure that testing and acceptance criteria are met before returning the affected equipment to service.

The following selected Facility Change Requirements were reviewed.

FCR 77-221	AFPT Speed Control Relay Modification
FCR 77-199	Modification of Wiring on AFPT Control Relays CW/AUX and CCW/AUX
FCR 77-154	Pressure Doors 215, 601 and 602
FCR 77-417	CRDCS - Trip Breakers
FCR 77-026	Proposed Technical Specifications change for DH11 and DH12
FCR 77-34	Modification to DH12 Control Circuit
FCR 77-47	Proposed Technical Specification Change to EVS
FCR 77-072	Change EVS delta P Cell Range from 10 inches to 2 inches H_2O
FCR 77-078	Proposed Technical Specification Change on Steam Generator Level Transmitters
FCR 77-120	Separation of NI-1 and NI-2 Cable Runs
FCR 77-152	Modification to EVS Blowout Panel Setpoints

The inspector found these facility change requests in various states of completion with the master copies in circulation for review and approval. In the exit interview the inspector discussed the general status of closing out completed FCR's and stated that the final review of FCR's would be unresolved pending an updating of the licensee's files. This was due in part that the licensee was engaged in an effort to close out outstanding FCR's.

7. Licensee Internal Audits

While reviewing the licensee's implementation of design changes and modification (Facility Change Requests, Paragraph 6), the inspector noted that the licensee had identified an item associated with the implementation of procedures QAP 2030 and AD 1845.

8. Auxiliary Feedwater Systems

As a result of an inoffice review of problems with the auxiliary feedwater systems, a telephone conversation on December 12, 1977, was held with representatives of the licensee to determine what special action they were taking to insure the reliability of the systems. On December 13, 1977, the licensee stated they would:

Complete an engineering evaluation of the auxiliary feedwater pump control system to determine if the system is adequate.

Would increase the surveillance testing from monthly to weekly.

Would not proceed above 75% power if any more problems developed until both the licensee and the NRC could evaluate the problem(s).

On January 12, 1978, the licensee completed an engineering evaluation of the auxiliary feedwater system problems including a failure on December 28, 1977. This evaluation and the associated system modifications were reviewed by the inspector. The results of the evaluation and modifications were discussed in the exit interview.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance or deviations. Unresolved items disclosed during inspection is discussed in Paragraphs 4 and 6.

10. Power Ascension Testing, TP 800.14

The inspector witnessed the performance of TP 800.14, Reactor/Turbine Trip performed December 15, 1977 from 40% power. Overall performance was evaluated, including adherence to test procedure and meeting of acceptance criteria.

No items of noncompliance or deviations were identified.

11. Exit Interview

The inspector met with licensee representatives (denoted in Paragraph 1) on December 15, 1977 and January 13, 1978. The inspector summarized the scope and findings of the inspection. The licensee representative made the following remarks in response to certain of the items discussed by the inspector.

Acknowledged the statement by the inspector with respect to the items of noncompliance (Paragraph 3).

Acknowledged the inspector's request for supplementary closeout reports to specified licensee event reports (Paragraph 2).

Acknowledged the inspector's statement that to closeout of LER NP-33-77-40 would remain unresolved pending the closeout of the investigation (Paragraph 4).

Reaffirmed their review and action on Circular 77-13 (Paragraph 5).

Stated that they would review AD 1845.00 with regards to the inspector's comments and would continue their efforts for the timely closeout of facility change requests. They also acknowledged the inspectors statement that closeout of facility change requests would remain unresolved pending updating of the licensee files (Paragraph 6).

Stated that they would continue weekly surveillance testing on the auxiliary feedwater pump. This surveillance frequency would continue until confidence and reliability had been reestablished, but would continue for a minimum of 8 weeks before a reduction to normal frequency (Paragraph 8).