U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-346/38006(DRS)

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: Oak Harbor, Ohio

Inspection Conducted: February 22-26 and March 7-11, 1988

Rolf A. Westberg

Inspectors: Rolf A. Westberg 4/11/88
Team Leader Date

Rolf A. Westberg for

Winston C. Liu 4/11/88
Date

David S. Butler

David S. Butler 4/11/88
Date

Ronald N. Gardner

Approved By: Ronald N. Gardner, Chief 4/12/88
Plant Systems Section Date

Inspection Summary

Inspection on February 22-26 and March 7-11, 1988 (Report No. 50-346/88006(DRS))
Areas Inspected: Special safety inspection of activities with regard to review of allegations and resultant review of QA implementing procedures (51061B) and quality records (51065B); design changes and modifications (37700, 37701, 37702); Licensee action on previously identified items (92701); and training (41400).
Results: Of the four areas inspected, no violations or deviations were identified in two areas; two violations were identified in the remaining areas (failure to install oil sightglass in accordance with approved procedures (Paragraph 3.c.(4)(f)) and failure to apply design control measures to a specification change (Paragraph 3.c.(5)(b)).

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DETAILS

1. Persons Contacted

Toledo Edison Company (TED)

L. F. Storz, Plant Manager
P. C. Hildebrandt, Engineering General Director
L. O. Ramseth, Quality Assurance Director
S. C. Jain, Nuclear Engineering and Independent Safety Engineering Director
D. S. Knaszak, Engineering Services Manager
J. C. Sturdavant, Licensing Principle
G. Honma, Compliance Supervisor
T. W. Anderson, Maintenance Planning and Outage Maintenance Superintendent
G. L. Tillman, Design Process Supervisor
M. J. Knaszak, Design Engineer

U.S. NRC

P. M. Byron, Senior Resident Inspector
R. N. Gardner, Chief, Plant Systems Section
J. J. Harrison, Chief, Engineering Branch

The preceding personnel attended the exit meeting at the Davis-Besse site on March 11, 1988. Other personnel contacted as a matter of routine during the inspection are documented in Attachment A to this report.

2. Licensee Action on Previous Identified Items

(Closed) Unresolved Item (346/86019-02): Review of clarification and procedural revisions to the Design Change Program. See Section 3.c.(2) of this report for details.

3. Facility Change Request (FCR) and Modification Review

a. Inspection Scope

This three-week special safety team inspection of the FCR and modification program reviewed the following areas: FCR closeout commitment to NRC; design control program; audits; program implementation relative to completed FCRs; and review of modifications scheduled for the fifth refueling outage. In addition to document reviews and interviews, limited walkdowns and verification of completed maintenance work orders (MWOs) and surveillance activities were accomplished. Team members also reviewed calculations in the civil/structural, mechanical, and electrical/instrument and control disciplines.

b. Summary

In general, the design change program was being effectively implemented; however, a number of program strengths and some program weaknesses were identified. In addition, violations of NRC requirements were identified. Details of the violations are explained in the body of this report. The identified program strengths and weaknesses are summarized below.

(1) Strengths

- 10 CFR 50.59 safety reviews were good.
- FCR closeout was vastly improved since 1985.
- Good improvement was noted in the control of Measuring and Test Equipment (M&TE).
- Excellent instrument and control (I&C) instrument data string packages.
- Implementation of the motor operated valve reliability program and MOVATS testing is a positive step in the resolution of MOV problems.

(2) Weaknesses

- Quality Assurance audits of Design were nontechnical, not activity oriented, and did not assess assurance QA activities.
- Assurance QA was understaffed and overworked on nondesign related activities.
- Too many procedures were required to implement the design process.
- Examples were found which show inadequate design verification; however, the overall designs were acceptable.

(3) Conclusions

The licensee has been aggressively attacking the FCR backlog and should accomplish their goal. In addition, the design process appears to be improving with the advent of the new modification system. Finally, the introduction of more technical design audits should solve the design verification weakness.

c. De. iled Inspection Find'ngs

(1) FCR Closeout Commitment Review

As a result of NRC inspection findings in 1985 (50-346/85031-02; FCR system is ineffective and 50-346/83035-05; MWO and FCR systems require further evaluation and improvement), Toledo Edison Company (TED) committed to the development of an action plan to reduce the FCR closeout backlog. The subsequent action plan identified 448 FCRs for closeout by the end of the fifth refueling outage (March 1988). Since the original commitment, more FCRs were added to the listing and the total reached 547. During this inspection, of the 547 FCRs, only 85 remained open. Since this inspection preceded the outage, it appears certain that the FCR closeout commitment will be met.

The responses to the 1985 NRC inspection findings also included commitments regarding procedure revisions, generation of a new procedure to control interdivisional FCR activities, and an audit of the FCR system by a third party contractor. The inspectors reviewed Procedure Nos. NEP-010, "Processing Facility Change Requests;" NEI-010.2, "FCR Closeout Instructions;" NG-NE-0301, "Plant Modifications;" and the January 16, 1986 audit of the FCR system by Stone & Webster. The inspectors also interviewed key members of the FCR closeout organization relative to FCR processing and prevention of recurrence of a backlog. The aforementioned reviews and interviews produced the following information:

- Personnel directly responsible for FCR closeout have increased from six in 1985 to the present thirteen.
- Design Process Management reports to the Engineering General Director This is a higher level of management than the reporting chain which existed in 1985.
- Management personnel with FCR responsibility have generally risen in level, that is, coordinator to supervisor and supervisor to manager.
- The depth of definition in the procedures has increased.
- Coordination between participating organizations has improved with the development of "Motherhood Procedure," NG-NE-0301.
- Training has improved. Where it was previously reading only, now classroom training is provided.
- Better procedural control of revisions and supplements to FCRs will help to prevent future backlog of FCR closeouts.

- Increased personnel dedicated to FCR closeout and the creation of a field closeout organization will also help to prevent future backlogs.

The inspectors concluded that the FCR closeout program had improved greatly between 1985 and the present and that it was being implemented effectively.

(2) Design Control Program

The inspectors reviewed the TED design control program to verify that it was in conformance with the QA program commitments and regulatory requirements. This review assessed the program procedures against the requirements of 10 CFR 50, Appendix B, Criterion III, and the TED QA manual commitment to the good work practices recommended by ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants." The review of the program procedures, which are listed in Attachment C to this report, was completed with acceptable results. The review produced one program weakness comment, that a large number of procedures were used to implement the program. This comment was discussed with the Engineering General Director who indicated that this fact had been recognized and was the subject of a third party assessment.

The inspectors also reviewed the unresolved item from Inspection Report No. 50-346/86019-02 and concluded that the previous concerns had been adequately addressed.

(3) Design Audits

The inspectors reviewed the audit program relative to design control to verify programmatic commitments and technical merit. The items considered during this review included independence of audit personnel, personnel qualifications, schedule, corrective action, and technical content.

Thirteen internal design audits (TED) and five external design audits (Architect/Engineer organization) conducted in 1985 through 1987 were reviewed. In addition, six audits conducted in 1987 relative to test control, training, and corrective action were also assessed. For a complete listing of these audits, see Attachment D to this report.

This review indicated that TED was implementing their commitment to audit design control; however, only two of the eighteen design audits reviewed were activity oriented or assessed technical items. The two audits were No. AR-87-BECHT-01 and No. AR-87-BECHT-02.

That the design audits were nontechnical, programmatic audits was identified as a program weakness and discussions were held with TED QA. As a result of these discussions, the inspectors reviewed the qualifications of the personnel who performed the design audits relative to their previous design experience. Of eleven TED auditors and lead auditors, three had previous design experience. Of seven technical specialists who participated in the audits, two had verifiable previous design experience. The inspectors also discussed the activities which were not verified by the TED audits, such as: calculations, 10 CFR 50.59 evaluations, implementation of safety class boundaries at transitions, and post modification test results.

As a result of the above discussions with QA, discussions were also held with Design Engineering and Engineering Assurance QA relative to assessment of the technical content of the design packages. Subsequently, the inspectors reviewed a draft procedure, No. EN-DP-01203, "Engineering Design Evaluation," and two QA Procedures, No. QA-EA-01102.03, "Quality/Technical Reviews," and No. QA-EA-01105.03, "Review of Facility Change Requests and Plant Modifications." In addition, two recently completed design evaluations conducted under Procedure No. EN-DP-01203 were reviewed.

The inspectors concluded that although the procedures and evaluations were acceptable, there was insufficient independence of the involved personnel to independently assess the technical content of the design packages.

Further discussions and interviews were conducted and the plant personnel indicated that the evaluations and reviews by Assurance QA were not being presented as an audit program but as a means of assessing and improving the design process. The inspectors agreed and concluded that the evaluations were an excellent method for providing another independent design verification. These discussions did, however, point to another program weakness. Since Assurance QA is performing an in-line review function of design packages, the QA organization should be assessing them in their audits of design. Further, the interviews with Assurance QA personnel and their manager and a review of the scope of their intended function indicated that they were understaffed. Nine people were performing procurement activities while only five were designated for design activities. Of these five, several were involved with various plant "task force" duties which take time away from design activities. This appeared to be another program weakness.

(4) Review of FCRs in Closeout

To assess implementation of the design process relative to completed FCRs, the team selected nine FCRs from the October 17, 1985 Confirmatory Action Letter (CAL) (No. CAL RIII-85-13, Item 1.a(4)) relating to safety-related piping system operability and twenty-one others selected at random (see Attachment B for a complete listing). The FCRs were primarily reviewed for completion of maintenance work orders (MWOs), the revision of procedures and the completion of required training. The team also reviewed design documentation to assess design verification, 10 CFR 50.59 safety evaluations, calculations, records, and interface information. In addition, items were walked down in the plant to verify correct installation per the FCRs. This review produced the following results:

(a) FCR 78-126: Modification to the drain lines from the steam generators to the condenser. Design changes were made to allow their use at normal operating pressures and temperatures for feed and bleed during start-up and shutdown to maintain water chemistry. The inspectors reviewed documentation associated with this package including a final design calculation filed with the package identified as Calculation C-ECS-063-002, Revision 0, dated May 17, 1984.

The stated purpose of the calculation was to determine horsepower and torque switch settings for motor operated valves MS 603 and MS 611 which are containment isolation valves in the steam generator blowdown lines. The inspectors identified a number of deficiencies in the calculation:

- No substantiation or references were provided for any of the design input used in the analysis.

- The methodology used to calculate horsepower had no apparent technical basis.

- There was no basis provided for equations used to calculate valve stem factor or opening torque for the valve.

- The source of the opening torque used to calculate horsepower was not provided and was not based on any apparent results of the equation identified to determine torque.

- The inspectors were unable to comprehend the methodology used since the originator was no longer employed at TED.

Despite these inadequacies, the calculation stated, "check performed in accordance with Exhibit VI, checking of calculations; results are satisfactory." The inspectors concluded that the results of this analysis were not meaningful and the analysis was not consistent with the requirements of ANSI N45.2.11 relative to design verification and design input.

In response to the inspector's concern, Toledo Edison informed the team that subsequent to the June 9, 1985 event (loss of all feedwater), Davis-Besse had initiated a comprehensive Motor Operated Valve Reliability Program. In accordance with this program, all safety related valve motor operators were tested using the MOVATS testing system. In addition, calculations for all safety related valves were performed to determine worst case design differential pressures. The results of these calculations were filed with the design data for each valve.

Toledo Edison provided the summary of the calculations (CME 3.01-204) for MS 603 and MS 611 which indicated that the worst case design differential pressure was based on the steam generator relief valve set pressure, 1050 psig. Valve testing (MOVATS) was based on this differential pressure.

The inspectors had no further questions relative to the design differential used for these MOVs. However, the inspectors were concerned that the design basis calculation still on file (C-ECS-063-002) may be used in the future by engineers as a source of design basis data in the performance of design modifications or other safety related work. Further, it is not known whether other similar calculations may exist which have been superseded by those in the Davis-Besse MOV reliability program files. These files have been updated to reflect new calculations and the results of MOVATS testing. Toledo Edison was unable to provide a resolution to this question during the inspection. Consequently, this item remains open pending TED resolution and subsequent NRC review (346/88006-01).

(b) FCR 79-308: Facility change request which added a second independent and redundant miniflow recirculation line from the High Pressure Injection (HPI) pumps to the Borated Water Storage Tank (BWST). Previously, the recirculation line from each pump was cross connected into a single line returning to the BWST. In their review of the Safety Evaluation for this FCR, the Safety Review Board (SRB) noted a concern that the modification could result in reduced HPI flow to the reactor coolant system. The response indicated that HPI flow would not be significantly altered. However, it was not obvious from the response to this concern that the change in system resistance would not result in reduced HPI flow. The SRB indicated that the response was inadequate and that more detail was required. Although further elaboration was provided, it was not clear that the issue had been addressed. The team could find no documented analyses to confirm that HPI flow would not be reduced.

In response to the inspector's concern, Toledo Edison provided a draft calculation which indicated that the additional path provided for recirculation would not significantly reduce HPI flow since the orifice in both lines is the major resistance to flow.

This item is not safety significant since the draft calculations provided to the team demonstrate that adequate HPI flow is provided to the reactor coolant system with the modified system. However, the item indicates a weakness of the part of the licensee in the documentation of design analyses relative to facility change requests. This item remains open pending completion of a finalized calculation confirming adequate HPI flow to the reactor with the modified system and subsequent NRC Review (346/88006-02).

(c) FCR 85-160: Modification to install a drain line from the pressurizer power operated relief valve (PORV) loopseal to the pressurizer surge line. The FCR package included a Design Review Checklist as part of design verification to assure that appropriate design considerations had been made and documented for the FCR. The inspectors reviewed the Design Verification Checklist and found that the reviewer had annotated several items with the following comments:

1 "Assume that loads developed as a result of pzr spray actuation has been considered. (This would be a water slug rushing up the drain line when a delta-P is developed between steam space pressure and RCS pressure." (Item 6)

2 "Assumed to have been performed." (Hydraulic analyses Item 11)

3 "Assumed" supporting calculations completed, checked, and approved. (Item 27)

The inspectors were concerned that the purpose of the checklist as a design verification tool was being circumvented by these assumptions.

In response to the inspector's concern (TED memo to file dated March 10, 1988, NED 88-20156), Toledo Edison determined that a seismic analysis was performed for the line (Item 6) which adequately addressed the issued raised by the checklist and that appropriate calculations had been performed (Items 11 and 27). The inspectors had no further questions on this issue. However, this observation contributes to the team's concern that design activities were not adequately documented for FCR's performed at Davis-Besse.

(d) FCR 85-0204: Main Feed Pump Turbine (MFPT) high discharge pressure trip. This FCR added a 0.1 second time delay relay to the nonsafety-related pump discharge pressure switch PSH-506 (MFPT 1-1) and PSH-582 (MFPT 1-2) trip logic. The time delay was added to prevent spurious trips due to high discharge pressure transients. The PSHs trip their respective MFPT on a pump discharge pressure of 1500 psig. The setpoint was based on the design pressure of the high pressure feedwater heaters (1500 psig) which are located downstream of the pumps.

The Instrument Information Sheet, for both PSHs, gave the PSH setpoint as 1500 psig ± 15 psig and suggested the measuring and test equipment (MTE) be equivalent to a 0 to 2000 psig Heise gauge, ± 2 psig. The "As Left" setpoint obtained from the instrument calibration records was 1500 psig for PSH-506 and 1495 psig for PSH-582. The Instrument Information Sheet provided for setting the PSH higher than the feedwater design pressure. The inspectors discussed with the licensee the need to factor in all the setpoint calibration errors to prevent exceeding the feedwater design pressure. Pending further review, this is considered an open item (346/88006-03).

(e) FCR 85-293: Change setpoint for pilot-operated relief valve (PORV) actuation to 2450 psig. The PORV is a safety grade valve with a non-safety grade actuator. Its safety function is to retain pressure boundary integrity of the Reactor Coolant System (RCS). Operability of the PORV was intended to minimize lifts of the safety-grade pressurizer code safety valves. Credit was not taken in Technical Specification (TS) Bases 3/4.4.3. for opening the PORV during any anticipated transients.

The TS requirement for the safety valves was a lift setting of $\leq$ 2525 psig corresponding to the ambient conditions of the valve at normal operating temperature and pressure. The PORV opening setpoint in TS was $\geq$ 2390 psig. The licensee determined the margin between the opening of the PORV and the safety valves based on the hot setting of the safety valve lift setpoint and the total PORV setpoint error (FCR 85-293). The licensee uses procedure MP 1401.02, "Pressurizer Code Relief Valve Removal, Disassembly, Repair, Assembly, Installation, Testing, and Reinstallation," in verifying the lift setpoint. The procedure provides both a hot and cold setpoint method. At the time of this FCR, the safety valves were set with the hot method. The licensee has developed a graphical signature (valve bonnet operating temperature vs. lift pressure) for each safety valve. The graph represents a linear function. The lift pressure decreases as the bonnet temperature increases (conservative direction). The licensee is now using the cold method (three lifts at $\leq$ 2525 psig) to provide the cold lift setpoint. The hot setpoint can be determined from the graph for a safety valves' nominal operating temperature.

The margin previously determined in this FCR has changed do to using the cold setpoint methodology. The inspectors discussed this item with the licensee including the need to determine the setpoint margin for all required TS Modes of operation, and for each installed PZR code safety valve and the PORV opening setpoint. The licensee was also requested to include in the determination all instrument/calibration errors (PORV instrument string) and the pressure gauge error used in calibrating the safety valves. They were also requested to develop a method to document the determination each time a safety valve is tested. Pending further review, this is considered an open item (346/88006-04).

The inspectors reviewed the last code safety setpoint calibration and determined that the safety valves in use were operable with a setpoint $\leq$ 2525 psig.

(f) FCR 78-024: Containment Spray (CS) pump bearing oil sightglass installation. Prior to June 1977, the licensee noted that during the operation of CS pumps 1-1 and 1-2 a difficult time was encountered in maintaining the proper oil levels. As a result, the oil levels were being kept too low and bearing damage occurred. To make it easier to monitor the oil levels, the licensee decided to install oil sightglasses on each pump. This occurred in May 1977, under MWO 77-0798.

The inspectors reviewed the documents contained in the above FCR package. The inspectors noted that the licensee performed the installation of the oil sightglass on the CS pumps without licensee commitments and NRC requirements being met in that:

1 No design drawings or detailed drawings were used.

2 No procedures/instructions were found for installation and inspection.

3 No documented design criteria/instructions were utilized for seismic qualification evaluation.

The above findings were discussed in detail with licensee representatives. No dissenting comments were received from the licensee. The inspector informed the licensee that these findings were examples of a violation of 10 CFR 50, Appendix B, Criterion V (346/88006-05).

In addition to the aforementioned findings, the inspectors noted that in the time period between 1977 and 1986, the installed oil sightglass assemblies were not seismically qualified although the plant was in operation. The NRC inspectors held discussions with licensee representatives regarding the potential reportability and operability requirements. At the time of this inspection, the licensee was not able to complete the evaluation of reportability and operability. Pending further review, this matter is identified as an unresolved item (346/88006-06).

The inspectors reviewed seismic qualification Calculation No. C-ME-61.01-076, dated February 19, 1986, and the revised Calculation No. C-CSS-61.01-102, dated March 4, 1988, for the installation of CS pump oil sightglass assemblies. The inspectors noted that the calculated stresses were well below the allowable stresses set forth by the applicable ASME code. Consequently, the installed oil sightglass assemblies were seismically qualified by the above calculations. The inspectors found a numerical error in the pump mass ratio calculation. However, it does not affect the outcome of the calculation. The inspectors concluded that the installed oil sightglass assemblies could perform their intended function during a seismic event; however, the error in the calculation was another example of inadequate design verification.

(g) FCR 85-010: Support modification on the Auxiliary Feedwater system. The inspector's review determined that the material identification for Item No. 8 was missing on Support Drawing No. GC-EBB-4-H11, Sheet 1 of 4, Revision T2. This was a further example of inadequate design verification.

This FCR modified a snubber support on the Auxiliary Feedwater piping to steam generator 1-2. NCR 85-003 identified two concerns. One was the interference of the snubber on support GC-EBB-4-H11 with its end bracket. The other was the support assemblies that were not installed in accordance with the applicable design drawings. The as-built support assemblies were reviewed by Bechtel. It was determined that the support assemblies were to be acceptable in terms of interim operation. For long term operation minor modifications were required. The modifications contained in this FCR involved rotating the two snubber end brackets 90 degrees and the rework of support members so as to permit snubber position to be horizontal during normal operation. In addition to the above support modification, the pipe local stresses were also evaluated due to the complex support assemblies.

On the basis of the above review, the inspectors concluded that the support modified could perform its intended function.

In general, the team found the review of documentation provided with the FCR packages to be cumbersome and not well organized. Although the team was able to assess the closeout of maintenance work orders for the FCRs reviewed, the team could not assure that any FCR package contained all required documentation, or that appropriate checklists had been executed to assure completion of the closeout process. Specifically, the team could not confirm that all procedures had been revised or training conducted as required. However, the team was encouraged by the initiation of a modification procedure which seeks to correct many of the problems contributing to the cumbersome closeout process.

The team found that the implementation of MOVATS testing and the motor operated valve reliability program was a positive step in the resolution of MOV problems at Davis-Besse.

(5) Review of New FCRs/Modifications

To assess implementation of the design process relative to new FCRs and Modifications, the team selected three design change packages at random from those scheduled for the fifth refueling outage. Two of the packages were FCRs No. 84-002, "Reactor Vessel Head to Hot Leg Vent Line Piping," and No. 86-0432, "Feed and Bleed Enhancements." The other package was a modification, No. 87-1107, "Recommended Improvements to Steam and Feedwater Line Rupture Control System (SFRCS)." In addition, the scheduled replacement of Station Batteries No. 2P and No. 2N was reviewed even though it was neither an FCR nor a modification. The reviews of FCR No. 84-002 and modification No. 87-1107 were completed with acceptable results. The reviews of FCR No. 86-0432 and the battery replacement generated the following results:

(a) FCR 86-432: Enhancements to feed and bleed capability

Toledo Edison committed to the NRC (Serial 1382, dated June 23, 1987) to enhance the present feed and bleed capability through a modification to the makeup system. Feed and bleed is not part of the design basis for Davis Besse. However, the upgraded makeup system would provide increased flow and add independent flow paths for each pump. Subsequent to these improvements, feed and bleed could accommodate failure of either makeup pump or the PORV, in addition to multiple failures of steam generator cooling systems which would necessitate the initiation of feed and bleed operations. All new equipment is to be purchased and installed as nuclear safety-related, Seismic Class I. Where possible, existing equipment will be upgraded to meet nuclear safety related requirements. These modifications are to be made in two phases during the fifth and sixth refueling outage. Feed and bleed enhancements are being made under FCR 86-432; however, the design is not yet complete.

The inspectors reviewed correspondence contained in the correspondence section of the FCR file relative to feed and bleed enhancements to be made under FCR 86-432. A listing of equipment located in the makeup pump room which identified qualification requirements was included in this package. The listing indicates that the startup lube oil pump motors for the makeup pump will not be qualified for the environment in the makeup pump room. The reason given for this was that the startup pumps are used only during startup of the makeup pumps and, at that time normal temperatures prevail. However, in Bechtel Calculation 540-72-22501, Revision 0, dated September 25, 1987 (see discussion in design Analysis, below), credit was taken for shutting down one of the makeup pumps to minimize the heat load in the room.

The inspectors concluded that shutting down one of the makeup pumps essentially eliminates the proposed feed and bleed enhancements as committed to the NRC in Serial 1382. A failure of the redundant pump would render the plant with no means of feed and bleed since the shutdown pump could not be restarted in the high temperature environment with the unqualified startup lube oil pump motor.

In response to the team's concerns, Toledo Edison advised that they were investigating the possibility of qualifying the startup lube oil pump motor for the temperatures determined in the pump room heat up analysis. If this is determined not to be feasible, the room heat up analysis will be revised to consider both pump motors operating, and the effects on temperature will be determined. The licensee stated that all required equipment will be evaluated for operation in this environment and procedures will be written to reflect the need to start (and continued operation of) both pumps if required.

This item is not safety significant since the feed and bleed enhancements are outside the design basis for the plant. However, the observation indicates that the NRC commitment to enhance feed and bleed capabilities may not be achieved if the startup lube oil pump motor cannot be qualified for its operating environment. Further, this observation indicates a weakness in the coordination of design basis information developed in design analysis with interfacing groups (e.g., equipment qualification). This item is considered unresolved pending completion of these evaluations and review by the NRC (346/88006-07).

The inspectors reviewed design documentation related to the feed and bleed enhancements made in FCR 86-432 including a number of design analyses. The team found that these calculations contained flawed methodologies, did not always consider worst case operating modes, and were based on unverified or unsubstantiated assumptions. The following were examples of these deficiencies:

1 MPR Associates, Inc., calculation "Options for Increasing Makeup System Capacity at Davis-Besse," dated December 30, 1986, was performed to determine the makeup system flow capacity for both the existing and modified systems. The results of this calculation (flows at various reactor pressures) were used as input for the B&W analysis (32-1168039-00) of feed and bleed capability for Davis-Besse. The inspectors identified the following concerns with the MPR calculation:

a The methodology used to determine the makeup flow to the reactor coolant system consisted of determining the system resistance curve for the existing and modified system for several different cases, and algebraically modelling the makeup pump curve. Recirculation flow to the BWST was assumed to be 35 gpm and a flow of 32 gpm was assumed to the reactor coolant pump seals. Based on these assumptions, the flow to the reactor coolant system was calculated in each case. However, the inspectors found that flows in this type of flow network must be determined based on an analysis of the resistance in each flow path. Depending on the relative resistance of each path, flow to the reactor coolant system could vary significantly.

The inspectors noted that flow to the reactor coolant pump seals is determined by a flow control valve. Thus, flow in this path should be based on the worst case control valve differential pressure plus other valve and line losses and for the modified system, recirculation flow is to be isolated.

b There was no basis given for the 10 psi suction pressure used.

c The modelling used for the modified system (second line) was not clear. The team was advised that the same piping losses were assumed for the second line as the existing makeup line which implicitly assumed the same line lengths for the new line.

In response to the inspector's concerns, TED indicated that a final calculation of flows in the upgraded system has been recently completed, and the calculation was being reviewed. The results of this calculation indicated higher reactor coolant system flows than those used as input for the B&W analysis. TED also indicated that this calculation would evaluate NPSH provided to the makeup pumps to assure that vendor NPSH requirements are satisfied during feed and bleed operations since this has not been done.

The inspectors had no further questions on this calculation. However, this item remains open pending final review and issue of the TED calculation of feed and bleed flows for the enhanced system (346/88006-08).

2 Bechtel Calculation 540-72-12501, Revision 0, dated September 25, 1987, "Analysis of the Makeup Pump Room (Rm 225) Heat Up," is a calculation to determine the temperature response of the makeup pump room during feed and bleed operations. The results of this calculation serve as the basis for equipment qualification temperatures in this room since the room coolers for the room are not safety grade and are not being upgraded to safety grade equipment. The inspectors had the following concerns with this calculation:

a The calculation assumed 30% of the room and 10% of the adjoining vestibule were congested (i.e., occupied with equipment, piping, ductwork, etc.). No basis was given for these assumptions. If the congestion were actually more than assumed, room temperatures could be higher than calculated. In response to this concern, Bechtel provided data which indicated that actual volume occupied by equipment and piping was less than 20% of the room volume.

b A Bechtel memorandum attached to the calculation indicates that a 1.15 multiplier (applied to pump motor heat load), originally used to account for piping and lighting heat loads, was deleted. Instead, actual piping in the room was modelled as a heat sink. However, no consideration of heat loads due to lighting in the room was included.

c The calculation took credit for heat transfer to cooler piping in the room via natural convection. The team found that heat transfer to cool piping via natural convection is not conservative and may not be effective.

ANSI N45.2.11 requires that assumptions should be verified or adequate substantiation provided to justify the assumptions. This item remains open pending resolution of these concerns and revision of the calculation to document the basis for the assumed room congestion and the other undocumented assumptions (346/88006-09).

3 Bechtel Calculation 34.34, Revision 0, dated September 25, 1987, was performed to "determine the pressure in the makeup pump while operating the pumps piggyback with the decay heat removal pumps." The

calculation determined the maximum pressure imposed on the makeup pump suction and discharge piping. The maximum suction piping pressure was based on static head due to the BWST at maximum level and decay heat removal pump shutoff head. To determine the maximum makeup pump discharge pressure, 2691 psig, the makeup pump head at 170 gpm (approximate feed and bleed flow) was added to the maximum suction pressure. Based on these pressures, the calculation concluded that maximum pressures were less than design. However, the inspectors found that the worst case discharge pressure would result from operation of the makeup pump at shut off head (e.g., against a closed pump discharge valve) while in the piggyback mode. The inspectors independently determined that maximum pressure could reach approximately 2900 psig in this case which is significantly greater than the pressure determined in the calculation. Since the design pressure for this line is 3050 psig, there is no safety concern.

The inspectors concluded that none of the concerns related to these calculations are safety significant. However, these observations contributed to the team's concern that design verification of analysis for facility changes is a weakness at Davis-Besse.

(b) Battery Replacement

The inspectors elected to look at the scheduled replacement of Station Batteries No. 2N and No. 2P because of their important function in the safe shutdown of the plant. The batteries were scheduled for replacement because they were approaching the end of their service life.

Station Batteries No. 2N and No. 2P are 60 cell, 1500 amp hour, lead calcium batteries. These batteries, together with D.C. Motor Control Center 2 and the battery chargers make up Train B of the 250/125 volt bus, as described in the Davis-Besse Technical Specifications.

Prior to entering the plant, the inspectors reviewed Purchase Order No. EN1Q-010783ST to GNB Batteries Inc., a synopsis of MWOs 1-87-1182-00 and 1-87-1184-00 for the battery replacement, and Specification Change Notice No. 01-03 to Specification No. 12501-E-19Q, Revision 1, "Technical Specification for Operational Phase for 250/125 Volt Station Batteries." The change notice changed the load profile for the battery performance discharge test.

The inspectors interviewed key personnel responsible for the battery replacement and the performance discharge test. This interview led to the review of Test Procedure No. ST 5084.02, "Station Battery Service and Performance Discharge Test." The inspectors noticed that the load profile in the test procedure was more conservative than the load profile required by the specification change. This information was given to Engineering and they were asked to explain the difference.

Engineering subsequently issued a memorandum which documented why the specification was changed and issued procedure change request No. 88-0336 to correct the procedure. However, it did not explain why the procedure was not changed when the specification change was processed, a time difference of six months. Discussions were held with Engineering and the following sequence of events was constructed:

1 Surveillance Report Number 84-36 and NRC Open Item No. 50-346/82-21-14 questioned the battery capacity.

2 Calculation C-EE-002-004, Revision 1, dated December 1984 determined the capacity was acceptable.

3 FCR No. 82-0029, Revision E, changed the DC distribution Technical Specification to adopt the Standardized Technical Specifications.

4 The revised Technical Specification was approved by the NRC on March 12, 1987, in Amendment 100 to License No. NPF-3.

5 Specification Change 01-03 to Specification No. 12501-E-18Q, dated August 3, 1987, should have required changes to the following interfacing documents:

   a DB-ME-09200
   b DB-ME-09201
   c DB-ME-3000
   d DB-ME-03002 (ST 5084.02)
   e DB-ME-03001

6 NRC inspection on February 26, 1988, determined that the above procedures had not been changed.

7 Procedure Change request for DB-ME-3002 (ST 5084.02) was processed on March 3, 1988.

This sequence of events made it obvious that Procedure No. NEP-021, "Specifications," was violated, in that, the required "Interfacing Document Worksheet" which would have accomplished the required changes was filled out but never processed. Further discussions were held with engineering management relative to the use of NEP-021 as a means for changing the specification. It then became clear that the above problem was an intermediate cause. The root cause was that the procedure was misused, in that the battery load profile change was treated as if it were a minor change, such as correcting a typo, without applying full design change controls. Since the change involved technical matters, an FCR should have been processed for the change to ensure that the proper controls were implemented. Failure to apply design control measures to a specification change is a violation of Criterion III of 10 CFR 50, Appendix B (346/88006-10).

Within the areas inspected, two violations, two unresolved items, and six open items were identified.

4. Allegation (RIII-87-A-0170) (Closed)

On December 30, 1987, Region III received an anonymous allegation concerning the manner in which backlogged Field Change Requests were resolved under the Course of Action submitted to the NRC in the wake of the June 9, 1985 event. The allegation stated that actions were taken which did not follow established administrative procedures and in which documentation was mishandled and fabricated. Three broad problems and one specific problem were identified:

a. FCRs were submitted for closure which did not meet the criteria. Closure was based on work requests which were voided or which were listed as open and work in progress.

b. FCRs were modified or consolidated which caused a delay in their implementation.

c. Unrelated FCRs were consolidated and given an implementation priority not commensurate with their safety significance.

d. FCRs which modified the Containment Spray Pump Oil sightglass isolated it from the oil reservoir for extended periods of time with plant operational and surveillance being conducted on the system. This was not evaluated for LER reportability by the Licensing Department as it should have been.

NRC Review

To assess these problems, the inspectors reviewed the FCR program procedures, selected and reviewed a sample of FCRs closed between the present and implementation of the Course of Action Program in 1985, and reviewed the Containment Spray Pump oil sightglass FCR. The results of the inspectors review are as follows:

(1) FCRs were submitted for closure which did not meet the criteria. The inspectors reviewed the FCR program procedures to determine the criteria for FCR closeout. The following procedures were reviewed: NEP-010, "Processing Facility Change Requests;" NEI-010.2, "FCR Closeout Instructions;" and AD-1845.03, "Facility Change Request Implementation." The inspectors also interviewed key members of the FCR closeout organization relative to criteria for FCR closeout.

To assess whether FCRs were closed without meeting the established criteria thirty-three FCRs out of the original 448 FCRs in the FCR closeout commitment to the NRC were reviewed in detail with special attention paid to the use of Maintenance Work Orders (MWOs) in the closeout process.

The inspectors were not able to substantiate this concern. No examples were found where the closeout criteria were not met. Specifically, no cases were found of FCRs closed by MWOs that were later voided.

(2) FCRs were modified or consolidated which caused delays in their implementation. The inspectors were able to substantiate this concern but only in the time period preceding the 1985 closeout commitment. The alleger was referring to the practice of continually adding supplements to FCRs so that they became so large that final closeout was difficult. This fact was recognized by the NRC and resulted in a violation in Inspection Report No. 50-346/85031-02 which stated "continually adding supplements to some FCRs (changing scope of work) partially contributed to some nonconforming or deficient conditions not being corrected in a timely manner." This violation was corrected and the violation was closed in Inspection Report No. 50-346/86004. The alleger was referring to a problem identified by the NRC whose correction was tracked by the NRC. Therefore, this item has no safety significance.

(3) Unrelated FCRs were consolidated and given an implementating priority not commensurate with their safety significance. The inspectors review of the program procedures documented in (1) above indicated that FCRs were not allowed to be consolidated. However, unrelated supplements; such as; mechanical, electrical, and I&C were routinely combined under a common FCR number. This item was previously identified by the NRC as a violation and tracked to closure. See Section (2) above.

The inspectors determined that this item was not safety significant.

(4) Containment Spray Pump Oil Sightglass. This concern was substantiated. See Section 3.c.(4)(f) of this report for details. It will be tracked until closure as Unresolved Item No. 346/88006-6 pending NRC review of the licensee's evaluation of reportability and operability.

5. High Pressure Injection (HPI) Direct Current (DC) Lube Oil Pump Review

During a configuration management walkdown, the licensee discovered that the thermal overload relay was installed upside-down for each HPI pump DC lube oil pump. Also, it was discovered that the installed control circuit fuses were rated at 1 Amp while the elementary drawing (E52B - Sheet 64, "Reactor Cooling System HPI Pump DC Lube Oil Pump") specified a 10 Amp fuse; and the installed supply fuses were rated at 15 Amp while the one line diagram (E-7, "250/125V DC and Instrumentation AC") specified a 10 Amp fuse. The licensee conducted a search of maintenance history records which showed no equipment replacement, system modifications, or control circuit changes. The licensee believed this condition had existed since initial equipment installation.

The inspectors reviewed the licensee's corrective actions. Tests were conducted to determine the motor and control circuit inrush and nominal operating current. These are listed as follows:

Circuits	In-Rush Current	Time of In-Rush Current	Operating Current
DC Motor	12.8a	0.6 second	2.6a
Control	0.7a	Not Determined	0.22a

These values support the original design. The supply fuse should be 10 Amp and the control fuse should be 1 Amp (determined from fuse curves and size availability). The licensee issued Design Change Notice E52B-442 which changed the 10 Amp fuse to 1 Amp on drawing E52B, installed the 10 Amp fuses in the plant, and correctly positioned the overload relay. Further review of drawing E52B revealed the overload relay contacts were not connected to the control circuit. The inspectors verified these items were correct per the "As-Built" plant design. The Electrical Superintendent indicated this was a common practice at Davis-Besse, that safe shutdown equipment supplied with overload relays would not have the relay contacts wired into the circuit.

The inspectors reviewed the initial findings and corrective actions to determine HPI operability. The overload relay had no affect on the HPI operability. The inspectors determine the supply fuse (15 Amp) installed since initial equipment installation provided adequate motor (overload) and wiring (fault) current protection. The 15 Amp fuses were coordinated with the upstream fuse (800 Amp). Review of the 10 Amp fuse melting time-current curve indicated the fuse would open in approximately 60 seconds on a 128% overload current. This is greater than the inrush current time of 0.6 seconds and will assure the fuse will not open during the oil pump start cycle.

The inspectors have no further questions on this item and have concluded that both HPI pump DC lube oil pumps were operable from initial plant startup.

6. Followup on Items from OSTI

The inspectors performed a follow-up review on two items from the OSTI (50-346/87-24) as follows:

a. Procedures are not updated in a timely manner following plant modifications. The inspectors reviewed this program weakness and found that the OSTI report was based on only one example. The review of 30 previously closed FCRs during this inspection did not support this concern. This item is considered closed.

b. A large backlog of engineering responses to Corporate Nuclear Review Board questions to 50.59 evaluations exists, primarily related to plant modifications. The number of safety evaluations requiring response at the time of the OSTI was 120. As of March 1, 1988, this number has been reduced to 27. This item is considered closed.

7. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involves some action on the part of the NRC or licensee or both. Open items are discussed in Paragraphs 3.c.(4)(a), (b), (d), and (e); and 3.c(5)(a) 1 and 2.

8. Unresolved Items

An unresolved item is a matter about which more information is required in order to ascertain whether it is an acceptable item, an open item, a deviation, or a violation. Unresolved items are discussed in Paragraphs 3.c.(4)(f) and 3.c.(5)(a).

9. Exit Interview

The inspectors met with licensee representatives denoted in Paragraph 1 during and at the conclusion of the inspection on March 11, 1988. The inspectors summarized the scope and results of the inspection and discussed the likely content of this inspection report. The licensee acknowledged the information and did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.

Attachments:
A. Personnel Contacted
B. Facility Change Request Review
C. Procedure Review
D. Audits Reviewed

Attachment A: Personnel Contacted

Toledo Edison

T. W. Anderson	Maintenance Planning and Outage Management Superintendent
M. L. Borysiak	Senior Instrument and Control Engineer
D. R. Breese	Facility Modification Supervisor
W. S. Delicate	CNRB Administrator
R. C. Elfstrom	Nuclear Specialist
G. N. Ferguson	Senior Engineer
D. J. Harris	Quality Systems Manager
P. C. Hildebrandt	Engineering General Director
G. Honma	Compliance Supervisor
T. R. Isley	Lead Instrument and Control Engineer
S. C. Jain	Nuclear Engineering and Independent Safety Engineering Director
J. J. Johnson	Operations Engineering Supervisor
J. R. Kasper	Operations Superintendent
J. B. Keagler	Associate Nuclear Engineer - Electrical
D. S. Knaszak	Engineering Services Manager
M. J. Knaszak	Design Engineer
D. R. Lightfoot	Facility Modification Department Superintendent
D. J. Mominee	Engineering Assurance Design Supervisor
J. E. Moyers	Quality Verification Manager
J. A. Nevshemal	Mechanical Engineering Manager
L. O. Ramseth	Quality Assurance Director
T. B. Ridlon	Nuclear Technologist

R. R. Rinderman	Quality Verification Supervisor
E. M. Salowitz	Planning Superintendent
E. D. Schock	Assistant Nuclear Engineer - Electrical
R. W. Schrander	Nuclear Licensing Manager
L. F. Storz	Plant Manager
J. C. Sturdavant	Licensing Principal
R. J. Swain	Assistant Nuclear Engineer
T. S. Swim	Civil/Structural Engineering Manager
G. L. Tillman	Design Process Supervisor
V. M. Watson	Design Engineering Director
A. G. Weedman	Engineering Assurance Manager
J. A. Wells	Engineering Assistant Analyst/Modifications
B. L. Wrightman	Engineering Assistant
A. K. Zarkesh	Independent Safety Engineering Manager

<u>Bechtel</u>

C. A. Leskovar	Design Engineer
M. L. Murphy	Senior Engineer

Attachment B: Facility Change Request Review

FCR Number	Description
78-0024	Containment Spray Pump Oil Sightglass
78-0126	Modify Drain Lines for Feed and Bleed Use During Startup and Shutdown
79-0308	Add Mini Recirculation Line From HPI Pumps to BWST
80-0221	Relief Valves SW3962 and SW3963 Setpoint Change
83-0063	HPI Lube Oil Pump A.C. Motor Replacement
83-0136	Replacing Existing AFW Pump Turbine Governors
84-0002	Reactor Vessel Head to Hot Leg Vent Line Piping
84-0111	SW Pump Pressure Switch Setpoint Change
85-0010	AFW Piping System Support Modification
85-0086	MS Piping System Support Modification
85-0126	Removal of Core Drill Sealant Material and Replacement with Non-Shrink Grout
85-0159	SFRCS Low Steam Pressure Trip Logic
85-0160	PORV Loopseal Drain Line Installation
85-0164	AFPT Overspeed Trip Mechanism Replacement
85-0167	SFRCS Full Trip Alarm
85-0201	ICS Steam Generator Low Level Setpoint
85-0204	MFPT High Discharge Pressure Pump Trip
85-0224	MS Line 'A' Snubber Addition
85-0239	EDG Room High/Low Temperature Alarm
85-0242	LPI Flow Transmitter Replacement
85-0263	RCP Seal Leakage Instrumentation Removal
85-0293	Raise PORV Opening Setpoint

85-0328	Makeup and Purification Flow Instrumentation
86-0002	RPS Fan Noise Suppression
86-0016	SFRCS Pressure Switches
86-0093	EDG Overspeed Trip Modification
86-0148	AFW Press to Test Lights
86-0162	Auxiliary .eedwater Controls
86-0174	Essential Steam Generator Level Control
86-0235	Flow Transmitter FT-4909 Replacement
86-0300	SFRCS/AFW Integrated Test
86-0432	Feed and Bleed Enhancements
87-0015	EDG Thermostat Setpoint Change

Attachment C: Procedure Review

Procedure Number	Title	Revision
AD 1844.02	Control of Work	3
AD 1844.03	Facility Change Request Implementation	0
AD 1844.03	Facility Change Request Implementation	1
AD 1845.04	Facility Change Request Closeout	0
DB-ME-3002	Station Battery Service and Performance Discharge Test	8
EN-DP-01203	Engineering Design Evaluation	Draft A
NEI-010.2	FCR Closeout Instructions	3
NEI-020.1	Instructions for PICA Forms	0
NEI-200.1	MOD Processing Instructions	0
NEP-010	Processing Facility Change Requests	1
NEP-011	Conceptional Design	0
NEP-020	Design Work Packages	0
NEP-200	Processing Plant Modifications	1
NG-NE-0301	Plant Modifications	1
QA-EA-1102.03	Quality/Technical Reviews	0
QA-EA-1105.03	Review of Facility Change Requests and Plant Modifications	2

Attachment D: Audits Reviewed

Year	Audit Report Number
1985	1349 1371 1344 1439
1986	1486 1511 1516 1572 AR-86-DESIGN-01
1987	AR-87-DESIGN-01 AR-87-DESIGN-02 AR-87-DESIGN-03 AR-87-DESIGN-04 AR-87-BECHT-01 AR-87-BECHT-02 AR-87-BWNPD-01 AR-87-B&ROE-01 AR-87-SWENG-01 AR-87-TESTC-01 AR-87-TRAIN-02 AR-87-TRAIN-03 AR-87-TRAIN-04 AR-87-CORAC-01 AR-87-CORAC-02