U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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Licensee:	Toledo Edison Company
Facility:	Davis-Besse Nuclear Power Station
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Dates:	April 10 - June 11, 1996
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EXECUTIVE SUMMARY

Davis-Besse Nuclear Power Station
NRC Inspection Report 50-346/96003

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers an approximate 8-week period of resident inspection activities as well as the results of announced onsite inspections by two regional radiation specialists, two regional engineering specialists, and one regional security specialist.

Operations

- Overall, operational activities were conducted in a controlled conservative manner.

- Shutdown risk controls were well implemented during the refuel outage conducted April 8 - June 2, 1996.

- On two occasions operators failed to follow operating procedures, which resulted in issuance of a violation. In the first case, following an inadvertent start of emergency diesel generator (EDG) 2, operators failed to complete the procedure for performing the engine shutdown and to place it in a standby condition as written (Section 03.1). In the second case, operators inadvertently drained 200 gallons of reactor coolant system inventory to the reactor coolant drain tank during a valve manipulation due to not following a precaution in the associated operating procedure, and the event was influenced by weak communications (Section 03.2).

- Operational requirements during shutdown conditions with the core off-loaded were not fully understood by operations personnel (Sections 04.1, 04.2).

Maintenance

- Overall, maintenance activities observed and/or reviewed during the inspection period were satisfactorily conducted.

- During the refuel outage, a heavy load was lifted over the open reactor vessel in violation of licensee procedures (Section M4.1). The safety consequences associated with this matter were under review at the end of the inspection period. The results of that review will be documented in a future inspection report.

- Overall, equipment material condition and housekeeping during the outage were satisfactory.

- During inservice inspection activities, the inspectors noted the following:

ISI personnel effectively detected and evaluated SG tube degradation and complied with applicable licensee procedures and program requirements. An eddy current testing (ET) probe used for detection of potential circumferential degradation in SG tubes was qualified on thicker walled steam generator (SG) tubes. Differing opinions of this probe's effectiveness, for detection of degradation in thinner walled SG tubes used in once through steam generators (OTSGs), reflect a need for development of industry qualified ET equipment (Section M3.1).

Recent industry experience with SG tube degradation found in OTSGs (groove intergranular attack and tube support plate dent cracking), was used for training and testing ET personnel and demonstrated good safety focus (Section M5.1).

Engineering

- Overall, engineering adequately addressed and resolved several issues that were identified during the refuel outage. These included problems associated with a displaced fuel assembly spacer grid (Section E1.1), cracked bearings found in a control rod drive mechanism (Section E1.2), and EDG problems that necessitated submittal of a license amendment request (Section E2.1).

Plant Support

- The radiation protection program, including the as low as is reasonably achievable (ALARA) program, was effective in keeping outage dose low. Improvements in the radiation work permit process and in the content of ALARA briefing packages were noted (Section R1.1, R8).

- Improvements in the control of shutdown chemistry resulted in lower radiation levels in various areas of the plant (Section R1.2).

- The radioactive material (RAM) control program was effective, as indicated by the low number of RAM found outside of specified areas during the refueling outage (Section R1.3).

- The implementation of the security program appeared to be satisfactory during the inspection period.

- Use of radiant heat energy shielding in the containment annulus was identified as a concern by the NRC, and during the inspection period, the licensee implemented compensatory measures (Section F1).

- The licensee identified that an oil collection lip was not installed on the reactor coolant pump (RCP) No. 2-1 motor housing as required to meet 10 CFR Part 50, Appendix R, requirements (Section F.2).

Report Details

Summary of Plant Status

At the beginning of the inspection period, the unit was in cold shutdown with the Tenth Refueling Outage (10RFO) in progress. The outage was completed on June 2, 1996, when the main generator output breakers were shut. At the end of the inspection period the unit was nominally at 100 percent power.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted ongoing reviews of plant operational activities. Overall, operational activities were conducted in a controlled, conservative manner. Except as noted below, personnel accomplished their tasks in accordance with procedural and regulatory requirements. The control of plant risk during shutdown conditions was, in particular, well controlled.

O1.2 Core Map Verification

Following completion of core reload activities, the inspectors reviewed the core verification videotape and compared fuel assembly serial numbers and fuel location/orientation to the approved core map for the current operating cycle to independently verify proper fuel placement. No discrepancies were noted during the review.

O1.3 Review of Shutdown Risk Controls

The inspectors reviewed the licensee's shutdown risk analysis for 10RFO and verified it conformed with administrative procedure NG-DB-116, "Outage Safety Control". During the outage, the inspectors verified that ongoing work activities were conducted in compliance with the shutdown risk analysis, and that appropriate compensatory actions were developed and implemented where they diverged. The inspectors identified no substantive concerns during the review in this area.

O3 Operations Procedures and Documentation

O3.1 Emergency Diesel Generator Inadvertent Start and Return to Standby

a. Inspection Scope

On May 15, while technicians were connecting test leads for the Data Acquisition and Analysis System (DAAS) to Emergency Diesel Generator (EDG) No. 2 in preparation for Integrated Safety Features Actuation System (SFAS) testing, an inadvertent start of EDG 2 occurred. An emergency shutdown of the EDG was then immediately performed.

Subsequently, the licensee valved out the starting air to the EDG and attempted to reconnect the DAAS leads. The result was the generation of a second EDG start signal although the engine did not roll with starting air unavailable. The inspectors independently reviewed the event and subsequent licensee followup actions.

b. Observations and Findings

The licensee determined the cause of the inadvertent start to be low input impedance associated with the particular DAAS circuit card used for the monitoring of the EDG start circuit. The inspectors reviewed the subject electrical drawings and confirmed the licensee's root cause determination.

Following the first start event, operations personnel performed an emergency shutdown of the engine using system operating procedure DB-OP-06316 (revision 01), Section 5.8, "Emergency Shutdown or Operation Following an Automatic Trip of EDG 2". Following the second event where a start signal only was generated, operators again entered DB-OP-06316. The intent was to shut down the engine and to place the EDG in a proper standby condition in both cases. However, the inspectors determined that operators failed to properly complete all steps of Section 5.8. The last step in the section specified that Section 3.10, "Stopping EDG 2", was to be performed. Section 3.10 included steps to both stop the EDG and to verify it to be in an appropriate standby condition. The last step in Section 5.8 was not performed as specified nor was Section 3.10 entered in either case. Additionally, the EDG room ventilation system was shut down and placed in standby although the procedure did not specify those actions in the steps that were completed. When the inspector questioned operators, they responded that they had returned the EDG to a standby status based upon their knowledge of the equipment in lieu of completing the procedure as written.

In addition, administrative procedure DB-OP-00000, "Conduct of Operations," specified that for other than simple, routine evolutions, procedures were to be in hand and steps signed off as they were performed. Although an EDG emergency shutdown following an inadvertent start would not be considered a routine, simple evolution, the steps in DB-OP-06316 associated with the emergency shutdown actions were not signed off as they were performed.

Subsequently, the EDG did autostart during the integrated SFAS test.

c. Conclusions

The inspectors concluded that the licensee's root cause determination for the EDG 2 start was appropriate, and that EDG 2 had been returned to a standby condition following the inadvertent start events based upon the fact that the EDG functioned normally when called upon to start and run during the Integrated SFAS test. However, the approved operating procedure was not properly utilized to assure the EDG had been properly left in a standby condition. As such, the failure to adequately adhere to procedural requirements is considered one example of a **violation (50-346/96003-01)** of 10 CFR Part 50, Appendix B, Criteria V.

O3.2 Inadvertent Transfer of Reactor Coolant System Inventory

a. Inspection Scope (71707)

A review of an inadvertent transfer of about 200 gallons Reactor Coolant System (RCS) Inventory to the Reactor Coolant Drain Tank (RCDT), during the 10th refueling outage, as documented in Potential Condition Adverse to Quality Report 96-0822, dated May 22, 1996, was performed. Inspector followup of this event consisted of verifying that affected piping and components were not over-pressurized, determining the apparent root cause(s) of the event, and verifying the adequacy of licensee corrective actions.

b. Observations and Findings

On May 22, 1996, at about 1:00 am, the plant was in mode 5 and operations personnel were making preparations to enter mode 4. Decay Heat Removal (DHR) Train 2 was operating, removing decay heat from the core with DHR Train 1 in the process of being placed into the Low Pressure Injection (LPI) standby mode. An Equipment Operator (EO), who was also a licensed Reactor Operator (RO), was conducting the valve lineup verification to place DHR Train 1 into LPI standby mode using Attachment 21 of procedure DB-OP-06012, Revision 2, Decay Heat and Low Pressure Injection Operating Procedure.

During the step that verified that motor operated valve DH 830 was closed, the EO observed that an information tag was hanging on the Control Room (CR) remote operator for DH 830 indicating that the valve was manually seated. He noted that the valve needed to be reclutched to its motor operator, however, the valve was located in a contaminated area (CA) in the DHR Heat Exchanger room. DH 830 provided cross connection between DHR Trains 1 and 2 pump discharges.

After consulting with the "Outside" (of the Control Room) (OCR) SRO, it was agreed to reclutch DH 830 later in the procedure. This decision was not communicated to the Shift Supervisor or the CR SRO.

The EO, having completed a majority of Attachment 21, received concurrence from the RO to reclutch DH 830. The RO did not inform the Shift Supervisor or CR SRO of the EO's intentions. Because of difficulties in reclutching DH 830, the EO cracked open DH 830 to reclutch the operator.

Cracking open DH 830 established a flow path from the DHR Train 2 pump discharge, through relief valve DH-1508 in the Containment Emergency Sump suction piping, to the RCDT. This transfer of water raised the level in the RCDT, and caused the RCDT high level alarm annunciator to alarm in the CR.

The Reactor Operator, aware that the EO was in the process of reclutching DH-830, observed that DH 830 did not indicate closed on the remote indicator. The RO then closed DH 830 using the remote operator, stopping the inadvertent transfer.

The EO, who was unaware that the inadvertent transfer had occurred, exited the CA, and called the CR to inform them that DH 830 had been reclutched. The CR informed the EO of the event at that time.

The inspectors reviewed RCDT level computer points and determined that DH 830 had been cracked open for about a 5 minute period, with about 200 gallons of Reactor Coolant transferring to the RCDT. Because this was a relatively minor loss of RCS volume, it did not result in any degradation of the DHR system's ability to remove Decay Heat from the core.

Other immediate actions that the licensee took were to check that DH-1508 had reseated itself. Subsequent corrective action was to conduct a visual inspection of affected components and piping for over-pressurization effects. No over-pressurization effects were observed.

Other corrective actions were to counsel the EO and the OCR SRO for performing an inappropriate valve manipulation despite repetitive training that emphasized potential loss of RCS inventory when operating DHR cross-connect valves. Additionally, negative performance letters were placed into their personnel records.

The licensee also conducted a technical review of the event and determined that theoretical component and piping pressures did not exceed previously analyzed maximum operating parameters. The evaluation of whether maximum operating parameters had been exceeded was facilitated by licensee documents, Field Problem Resolution 88-0794, and Bechtel Letter BT-16352, which addressed questions concerning potential over-pressurization of DHR suction lines.

The inspectors visually inspected that components and piping that were inadvertently pressurized by this event had no leaks or apparent deformities. The inspectors also verified that the responsible operators were counselled. Additionally, the inspectors evaluated the licensee's technical review of the event, with no concerns noted.

The inspectors independently reviewed operator performance leading to the event. An interview of the Shift Supervisor determined that the Shift Supervisor and CR SRO were aware that DH 830 was manually seated, and had discussed among themselves risks associated with reclutching DH 830 while DHR Train 2 was in operation providing cooling to the core. This information had not been communicated to the rest of the shift.

A review of the lineup procedure determined that if the operator had reclutched DH 830 when its position was verified in the procedure, no path for an inadvertent transfer would have been present.

Additionally, an informational placard located below the CR hand switch for DH 830 stated, "PRIOR TO OPERATING XVER VALVES, ISOLATE MINIMUM FLOW VALVE IN OPPOSITE LOOP (DH10 OR DH26)". If this had been done, no inadvertent transfer would have occurred.

The DHR procedure included precaution 2.1.5, "Whenever the DH cross-over line is used (DH 830 & DH 831), the suction valve to the disabled pump must be closed to prevent over pressurization of the DH pump suction line and lifting PSV (DH) 1508 or PSV (DH) 1509." This precaution was not observed and would have prevented the inadvertent loss of inventory if it had been followed.

The EO, when questioned, did not think that the precautions towards opening the valve applied towards reclutching the valve to its motor operator. Originally, it was not his intention to manipulate the valve. However, when he was in the process of reclutching the valve, he cracked it open because of difficulties in clutching the valve when the valve was on its shut seat.

c. Conclusions

Immediate and follow on corrective actions were evaluated as adequately addressing personnel and equipment concerns.

The EO and OCR SRO displayed a lack of sensitivity toward the operation of the DHR cross-connect valves despite prior training, an informational placard, and a procedural precaution that explicitly explained the consequences of improper valve sequencing.

There were two cases of poor communications. One case was that the OCR SRO did not inform the rest of the shift of his instruction to the EO to place DH 830 on the motor operator. The other case was that the RO who obtained notification that the EO was going to place DH 830 on the motor operator acknowledged the EO without informing the CR SRO. Either one of these communications potentially could have prevented this event from occurring.

Although not assessed as poor communications, the Shift Supervisor and the CR SRO could have prevented the event from happening if they had communicated their concerns about reclutching DH 830 to the rest of the shift.

This event involved a failure to follow procedure precaution 2.1.5 of DB-OP-06012. As such, this is considered a second example of a **violation (50-346/96003-02)** of 10 CFR Part 50, Appendix B, Criterion V.

04 Operator Knowledge and Performance

04.1 Condition Not in Conformance With the Updated Safety Analysis Report

a. Inspection Scope

On April 30, 1996, following an inadvertent start of EDG No. 2, operators noted that room temperatures were increasing more than expected with the EDG running. Following shutdown of the EDG, the licensee identified that a ventilation supply damper for the EDG cubicle was not functioning properly. EDG No. 2 was declared inoperable, however, this condition coupled with EDG 1 also being inoperable with the core offloaded to the spent fuel pool (SFP), and the associated effect on Decay Heat Removal System operability was not recognized as a condition not allowed by the Updated Safety Analysis Report (USAR).

b. Observations and Findings

A similar situation was identified near the beginning of the refuel outage and is further discussed in Section E2.1 of this report. However, in the earlier case, the licensee recognized the condition was not allowed by the USAR and submitted a License Amendment Request (LAR) to the NRC for prior approval. The license amendment was granted on a one time only basis. In the later case, operations personnel verified that technical specifications were not applicable but did not contact support personnel to identify whether other licensing requirements were applicable. Once the configuration was determined to be in non-conformance, corrective actions were expeditiously pursued to repair the ventilation damper.

c. Conclusions

The inspectors were not able to definitively determine the reason operations personnel did not recognize the April 30 event was similar to the configuration and consequences associated with the recent license amendment (which they had received training on prior to implementation). Plant management counselled personnel associated with this event to review all applicable documentation (not just technical specifications) and/or request assistance to perform those reviews.

04.2 Spent Fuel Pool Operability

a. Inspection Scope

Following the inadvertent start of EDG 2 as discussed above and in Section 03.1 of this report, operations personnel failed to fully understand the correlation between EDG inoperability and spent fuel pool Emergency Ventilation System (EVS) operability. At the time EDG 2 was declared inoperable, EDG 1 was also inoperable due to ongoing maintenance. Initially, operations personnel determined that the EVS system remained operable with both EDGs inoperable. Later, that decision was reversed and the EVS was declared inoperable and an

Emergency Notification System (ENS) call to the NRC headquarters operations officer made. Following additional evaluation, the licensee again reversed their position, and determined that the EVS system was operable even with both EDGs inoperable.

b. Observations and Findings

The inspectors reviewed the licensee's final determination of EVS operability and verified its conformance with applicable licensing and design basis documentation.

c. Conclusions

The inspectors concluded that operational requirements during shutdown conditions, especially during times the core was offloaded to the SFP, were not fully understood by all necessary personnel. The licensee concluded the same and initiated actions to better delineate those requirements via additional reviews of the governing documents, and the communicating of those results to plant personnel. At the end of the inspection period, those efforts were continuing.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Activities

a. Inspection Scope (62703)

The inspectors observed all or portions of the following work activities:

MWO 3-96-0683-01	Clean & Inspect Class 1E Bus F1
MWO 3-96-0668-01	Clean & Inspect Class 1E Bus D1
MWO 1-96-0438-00	Furmanite of source valve for S/G 1-1 Pressure Transmitter SP12B1
MWO 3-95-0229-01	Motor Operated Valve Testing/Preventive Maintenance on Component Cooling Water Discharge Isolation Valve CC 5096
MWO 7-91-0253-31	Main Feedwater Stop Valve, FW 601, Stem Replacement

b. Observations and Findings

The inspectors found that maintenance performed on equipment was performed with properly authorized Maintenance Work Order packages (MWOs). MWO work history sheets were updated in the field to provide documentation of maintenance observations. More significant problems were documented for resolution on Potential Condition Adverse to Quality Reports (PCAQRs). Quality control check points were observed to be conducted and documented at appropriate steps in work procedures.

When questioned, maintenance workers were knowledgeable of their job activities and utilized appropriate controls to minimize the entry of foreign material into Foreign Material Exclusion (FME) areas. No instances of inadequate FME controls were noted by the inspector.

The inspectors review of a sample of MWOs affecting technical specification related equipment determined that appropriate "Inoperable Equipment Tracking Log" entries were made. These entries imposed mode restraints on the plant pending a return of the affected equipment to an operable status.

M1.2 Inservice Inspection (ISI) Unit 1 - Review of Program

a. Inspection Scope (73051)

Inspectors reviewed the inservice inspection program and its implementation for compliance with technical specifications, ASME Code, and NRC requirements.

b. Observations and Findings

The licensee's Steam Generator (SG) Eddy Current Testing (ET) inspection scope exceeded technical specification requirements and met GL 95-03 commitments.

c. Conclusions

No violations or deviations were identified. The SG ET scope met GL 95-03 commitments.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Reactor Building Walkdowns

a. Inspection Scope (71707, 62703, 92901, 92902, 92903, 92904)

During the refueling outage the inspectors conducted several tours of the reactor (containment) building. In addition, "closeout" inspections were conducted to independently verify that the condition of the reactor building was appropriate to support unit restart. In addition, a final walkdown was conducted by the inspectors at full pressure/temperature conditions to independently verify proper equipment operation and absence of liquid and steam leaks.

b. Observations and Findings

The inspectors noted, during one walkdown early in the outage, that compressed gas cylinders were being stored adjacent to, and in contact with, instrumentation associated with steam generator level indication, control, and protective functions. Once identified, the licensee removed the compressed gas cylinders and verified that the associated instrumentation would be calibrated and/or functionally checked to ensure that the cylinders had not affected their calibration due to inadvertent impact. Subsequently, no calibration problems were identified. The licensee committed to include compressed gas cylinder storage as an item in the post-outage critique to ensure that future controls would be appropriately specified and followed.

Several junction box and electrical cabinets were noted with loose or missing access cover/door latches and/or closure screws. The inspectors informed the licensee, and followup actions were taken to correct the examples noted.

A walkdown of the containment emergency sump was also conducted by the inspectors at a time when all work activities were complete in that area. The inspectors noted overall good housekeeping and material condition. However, the inspectors questioned the appropriateness of storing large test flanges in the emergency sump area. This matter is further discussed in Section E1.6 of this report.

c. Conclusions

Overall reactor building material condition and housekeeping appeared to be satisfactory. Specific discrepancies that were noted were corrected in a timely manner. No concerns were identified that directly impacted on operability of equipment.

M3 Maintenance Procedures and Documentation

M3.1 Inservice Inspection - Procedure Review

a. Inspection Scope (73052)

NRC inspectors reviewed SG data analysis guidelines, ISI procedures, and lists of equipment used during observed ISI activities for compliance with ASME Code and NRC requirements.

b. Observations and Findings

ISI procedures reviewed by the inspectors were approved by the ANII and met ASME Code Section XI, 1986 Edition, requirements.

An unshielded 0.115" diameter rotating pancake coil (MRPC) probe was used for detection of circumferential cracking in SG tube expansion transition areas. This probe was qualified for detection of circumferential modes of cracking per PWR Steam Generator Examination Guidelines (EPRI NP 6201, Appendix H, Revision 3) and met GL 95-03 commitments. Inspectors noted that this probe was demonstrated and qualified on thicker walled tubes used in recirculating type steam generators. The lead ET analyst performing the SG inspections, considered the 0.080" diameter MRPC probe to be more effective at detecting degradation in thinner walled tubing used in once through steam generators (OTSGs). Licensee personnel reported that industry efforts under consideration included qualification of the 0.080" diameter MRPC probe for detection of circumferential cracking. Licensee personnel considered the 0.115" diameter MRPC probe to be proven effective at detecting degradation in OTSGs based on results from other OTSG ET inspections.

c. Conclusions

No violations or deviations were identified. The SG ET scope met GL 95-03 commitments. An ET probe used for detection of potential circumferential degradation in SG tubes was qualified by the industry on thicker walled SG tubes. Differing opinions of this probes effectiveness, for detection of degradation in thinner walled SG tubes used in OTSGs, reflected a need for development of industry qualified ET equipment.

M4 Maintenance Staff Knowledge and Performance

M4.1 Heavy Load Lifted Over Open Reactor Vessel

a. Inspection Scope

On April 16, 1996, the licensee identified that a heavy load as defined by NUREG-0612 was lifted over the open reactor vessel in violation of licensee procedures. At the time of the event the reactor vessel head was removed with the vessel internals plenum still installed and irradiated fuel in the reactor vessel. The heavy load was the Reactor Vessel Head Lifting Tripod (RVHLT), which was raised near the reactor building polar crane's uptravel stop, and was then traversed laterally across containment to support incore monitor work.

b. Observations and Findings

The licensee subsequently submitted licensee event report (LER) 96-005-00 describing the event, its root causes, and corrective actions taken and/or planned. Initial inspector review of the LER determined that weaknesses in communication between workers in containment, outage central, and the control room, coupled with weaknesses in understanding heavy load lift requirements by personnel involved with the lift contributed to the event.

This event is currently under NRC review. The licensee had retained a contractor to perform computer modelling and analyze whether fuel damage would have resulted from a postulated drop of the RVHLT. This matter is considered an **unresolved item (50-346/96003-03)** pending NRC review of the contractor's analysis and consequences of a postulated drop on the fuel.

M4.2 Maintenance Tracking

a. Inspection Scope (62703)

An inspector review of the effectiveness of maintenance controls during the 10th refueling outage was performed. Outage shift briefs were attended to determine if management was aware of maintenance and plant status, Shift Managers were questioned to determine whether maintenance conditions were permitted by technical specifications, and the Inoperable Equipment Tracking Log was reviewed to determine its utilization and effectiveness.

b. Observations and Findings

Outage shift briefs were conducted before each 8 hour shift and communicated to outage management changing radiological conditions, maintenance status of important equipment, and important Potential Condition Adverse to Quality Reports (PCAQRs). Additionally, the shutdown risk advisor communicated the shutdown risk concerns to outage management to heighten their attention to higher risk plant conditions.

Shift Managers maintained an up-to-date status of maintenance in progress and were able to determine if maintenance activities were active, field complete, required post maintenance testing, or closed.

Multiple mode restraint barriers were observed to be effectively implemented such as comprehensive reviews of the Inoperable Equipment Tracking Log by shift management, independent reviews of mode restraint PCAQRs by quality control personnel, and independent reviews of maintenance activities by maintenance and engineering personnel. These reviews of the operability status of technical specification required equipment were effective in preventing the plant from entering prohibited modes without the required equipment being operable.

M4.3 Inservice Inspection - Observations of Work Activities and Data Review

a. Inspection Scope (73753 and 73755)

The NRC inspectors observed ISI personnel and reviewed data recorded during ISI activities to determine compliance with ASME Code and NRC requirements. The NRC inspectors observed the following activities:

- Framatome Technologies, BWI, Verner & James and General Public Utilities personnel performing eddy current testing (ET) of SGs.
- Framatome Technologies personnel performing magnetic particle testing of steam generator B inlet nozzle weld and ultrasonic equipment calibration checks for core flood piping weld examinations.

b. <u>Observations and Findings</u>

Permeability variations (potentially masking flaw indications), were recorded on SG ET data for a majority of the 97 SG tubes inspected with a non-magnetically biased bobbin probe. The affected portions of these tubes were reinspected with a magnetically biased bobbin probe, that effectively eliminated the permeability variations.

The licensee reinspected SG tubes using an MRPC probe where ET indications had been detected by bobbin coil. Three SG A tubes in which bobbin coil indications, confirmed by the MRPC probe, were plugged.

MRPC probe inspections of SG A, detected a single 0.15" long axial crack-like indication in a stress-relieved tube expansion transition (588-119). A portion of this tube was removed for further analysis and characterization of the indication. The ET scope was expanded in accordance with technical specification requirements (C-2 classification) for SG A (nine percent of the SG tube expansion transitions were examined with the MRPC probe). In addition three percent of SG B tube expansion transitions were examined with the MRPC probe. No additional flaws were detected.

c. <u>Conclusions</u>

No violations or deviations were identified. ISI personnel effectively detected and evaluated SG tube degradation and complied with applicable licensee procedures and program requirements.

M5 Maintenance Staff Training and Qualifications

M5.1 <u>Inservice Inspection - Qualifications of NDE personnel</u>

a. <u>Inspection Scope (73753)</u>

Inspectors reviewed Framatome Technologies, BWI, Verner & James and General Public Utilities personnel qualifications and certifications for compliance with ASME Code, SNT-TC-1A and applicable NRC requirements.

b. <u>Observations and Findings</u>

Written and practical site specific tests, which included questions on recent OTSG experience with groove intergranular attack and tube support plate dent cracking, were passed by all ET personnel performing SG inspections.

c. <u>Conclusions</u>

No violations or deviations were identified. The use of recent industry experience with SG tube degradation found in OTSGs (groove intergranular attack and tube support plate dent cracking), for training and testing ET personnel demonstrated good safety focus.

III. Engineering

E1 Conduct of Engineering

E1.1 Displaced Fuel Assembly Spacer Grid

a. Inspection Scope (37551, 92903)

During fuel inspection activities conducted during the refuel outage, the licensee identified that one spacer grid on one fuel assembly was displaced several inches lower than intended. Spacer grids are used for lateral alignment of the individual fuel rods within an assembly. The spacer grid was returned to its proper position and the fuel manufacturer, Framatome Technologies, Inc. (FTI), was contacted to assess the significance of the slippage.

b. Observations and Findings

FTI determined that a mispositioned spacer grid could adversely affect the fuel coolable geometry in a post Loss of Coolant Accident (LOCA) environment. However, no viable mechanism could be identified that would cause a downward movement of a spacer grid during power operations. Because of the flow dynamics in the core, the spacer grids would tend to move upwards in the direction of core flow. However, upward movement of spacer grids are prevented by the fuel assemblies physical construction. FTI postulated that the spacer grid had moved as the assembly was being lifted from the core during refueling operations. FTI felt that the assembly must have come into contact with at least 2 other adjacent assemblies, due to their becoming bowed during power operations, to cause the observed spacer grid slippage. The licensee recognized that the potential could exist for similar slippage of spacer grids to occur during future refuelings and, at the end of the inspection period, engineering personnel were evaluating what additional inspections of the fuel may be prudent. The inspectors reviewed FTI's analysis and noted no concerns.

However, while reviewing the licensee's spacer grid problem, FTI identified an additional generic issue associated with this matter. Specifically, a review of the Mark-B fuel assembly horizontal faulted condition analyses revealed that only core periphery fuel assemblies had been evaluated for consequences of grid plastic deformation. Initial engineering followup determined that the 10 CFR 50.46, 2200 degrees F limit for maximum cladding temperature could be exceeded, but application of NRC approved leak before break analysis method would show acceptable deformation levels only. FTI communicated this information to the B&W Owners Group and initiated discussions with NRC headquarters.

c. Conclusions

Licensee followup action to the slipped spacer grid was appropriate and timely. No operational concerns were outstanding at the time the unit was returned to power. The licensee was evaluating what additional inspections may be needed during future refuelings. However, the generic concern with the B&W analyses remained and was being addressed with NRC HQ. Pending licensee determination of what future inspections may be needed to detect spacer grid slippage and the resolution of the generic issue with the B&W analyses, this is considered an **inspection followup item (50-346/96003-04)**.

E1.2 Control Rod Drive Mechanism Bearing and Leaf Spring Cracks

a. Inspection Scope (92903)

During disassembly and inspection activities of Control Rod Drive Mechanism (CRDM) N-12, personnel identified cracks in the radial bearing, synchronizing bearing, and the anti-rotation leaf spring.

b. Observations and Findings

To determine extent of condition, two additional CRDMs were disassembled and inspected with no further bearing problems noted. In addition, all CRDMs leaf springs were inspected with a borescope with up to six initially determined to possibly have similar cracking in the leaf springs. Following changeout of the subject springs with indications of possible cracks, only one spring was verified to have an actual crack.

A failure analysis was performed by FTI which determined that the bearing cracks were caused due to one or more physical impacts, i.e., that the bearings were cracked during assembly or disassembly of the CRDM, and did not occur during operation. In addition, FTI determined that both bearings would have continued to operate acceptably with the cracks present. No further degradation would have been anticipated, and even had further degradation occurred, the radial bearing served no safety function. Degradation of the synchronizing bearing would be detected by difficulty in maintaining its coupling to its control rod prior to affecting any safety function.

The inspectors inspected the subject components and reviewed the licensee's corrective actions. An inspector review of the FTI report determined that the evaluation had adequately addressed the pertinent issues.

No prior CRDM bearing cracking problems had been identified, either onsite or at other B&W units.

c. Conclusions

The inspectors concluded licensee review was comprehensive and adequately characterized the extent of condition and safety consequences associated with the specific components found degraded as well as more generic implications. The inspectors had no further concerns on this matter.

E1.6 Containment Emergency Sump Walkdown

a. Inspection Scope (71707, 92903, 37551)

As discussed in Section M2.1 of this report, the inspectors performed a walkdown of the containment emergency sump and associated areas to independently verify that the area had been returned to an acceptable condition following completion of outage work activities.

b. Observations and Findings

The inspectors noted that large test flanges had been stored within the emergency sump grated area. When questioned on what evaluation had been done to allow the flanges to be stored there, the licensee was unable to produce engineering documentation that addressed the acceptability of this configuration. The licensee was able to produce a memorandum of a telephone call that was made from maintenance to engineering personnel in 1982 that addressed this issue. The telephone documentation was made by maintenance personnel following completion of a telephone call with engineering. However, no formal engineering documentation justifying the configuration was available.

The inspectors then questioned the evaluations conducted for other materials stored in containment during power operations, including scaffolding materials, slings, and rigging. The licensee indicated that each item was reviewed on a case-by-case basis and that the depth of the review varied over time. Specifically, the licensee indicated recent reviews were very detailed and conducted in accordance with current engineering practice. However, reviews conducted earlier than 1990 were not as comprehensive.

c. Conclusions

As a result of discussions with engineering and operations personnel, the inspectors determined that the storage of the test flanges in the containment emergency sump grated area was acceptable. However, the inspectors remain concerned that appropriate documentation of the engineering justification did not exist to support this conclusion. In addition, older engineering evaluations to justify the storage of other materials in containment may have been weak. Further inspector followup of this matter is needed to assess whether the engineering justifications for the storage of materials in containment meet regulatory requirements. Pending completion of that review, this matter is considered an **unresolved item (50-346/96003-05)**.

E2 Engineering Support of Facilities and Equipment

E2.1 DHR License Amendment

a. Inspection Scope

During troubleshooting of EDG 1 during the refuel outage, the licensee determined that substantial overhaul of the generator was needed. The generator was removed to an offsite shop facility to be rewound. However, the licensee had planned to perform work on Train 2 of the Decay Heat (DH) System. Subsequent review by the licensee as to whether the work could still proceed with EDG 1 inoperable, revealed that with the core offloaded to the Spent Fuel Pool (SFP), DH Train 1 was required to be operable as the "qualified" makeup source to the SFP. However, by reference, the USAR specified that for DH Train 1 to be operable, both its normal and emergency electrical supply be available. DH Train 1's emergency electrical supply was EDG 1. Subsequently, the licensee prepared and submitted a license amendment request (LAR) on April 18, 1996, to allow the station blackout diesel generator (SBODG) to be substituted for EDG 1 as the emergency electrical supply for DH Train 1. The inspectors reviewed the licensee's submittal concurrently with the NRC Office of Nuclear Reactor Regulation (NRR).

b. Observations and Findings

The inspectors independently verified that the necessary temporary equipment lineups, and compensatory actions to support the proposed alternate configuration could be adequately performed. A review of the electrical loads associated with the SBODG, and associated interconnecting buses was performed. In addition, the inspectors verified that the proposed compensatory actions were appropriately incorporated into procedural requirements.

c. Conclusions

No substantive concerns were identified during inspector review of the licensee's proposed actions. Subsequently, NRR approved the license amendment on a "one-time only" basis. The licensee thereafter entered into the alternate lineup, implemented the compensatory actions, and completed the maintenance on DH Train 2.

E8 Miscellaneous Engineering Issues (92902)

E8.1 (Closed) Inspection Followup Item (50-346/94013-02): Auxiliary Feedwater (AFW) Steam Supply Check Valves MS-734 and MS-735 Chattering During Normal Operation. Subsequent licensee analysis determined the cause of the chattering to be small pressure perturbations across the valve seats/discs caused by small changes in steam flow. A combination of steam traps in the subject lines were opened to incrementally increase steam flow across the valves and thereby reduce the pressure fluctuations. When this was done, the chattering phenomenon substantively stopped. The decision was then made to continue maintaining the subset of steam traps open throughout the operating cycle. Appropriate procedures were revised to direct this action each plant startup. To recover the steam that was routed through the steam traps, installation of a modification was ongoing during the inspection period to direct the steam flow through the steam traps back to a feedwater heater.

The inspectors inspected the internals of MS-734 and MS-735 which were disassembled during the most recent refuel outage. Some degradation was noted on both sets of valve internals. However, the rate of degradation was such that routine disassembly and inspection during each refuel outage would be adequate to assure that any needed refurbishment would be done in a timely manner. The inspectors had no further concerns on this matter.

E8.2 (Closed) Inspection Followup Item (50-346/94004-02): Inadequate Engineering Evaluation of Temporary Lead Shielding. The licensee subsequently reviewed the current revision to the temporary lead shielding control program procedure and determined the procedure to be adequate. A memorandum was subsequently issued by RP management to appropriate parties specifying that once temporary shielding was placed, no change in its configuration was to be allowed unless additional engineering evaluation was conducted. No further examples of inadequate placement of temporary lead shielding was thereafter identified.

E8.3 (Closed) Inspection Followup Item (50-346/93013-02): Loss of Service Water to the Turbine Plant Cooling Water Heat Exchangers. The licensee subsequently determined that with the ambient air temperature greater than the setpoint for the transfer of the Component Cooling Water (CCW) train while in standby, a loss of the normal service water flowpath would occur. The licensee subsequently replaced the Temperature Indicating Controllers (TICs) with a type that was ambient temperature compensated. This was deemed an acceptable resolution to ensure this matter would not recur.

IV. Plant Support

F1 Control of Fire Protection Activities

During the inspection period, the Office of Nuclear Reactor Regulation (NRR) pursued a concern regarding the adequacy of radiant heat energy shields installed in the containment annulus area. Following discussions with the NRC, the licensee initiated appropriate compensatory measures to address the specific areas of concern. However, it appeared that previous licensee actions were not adequate. Pending completion of inspector review of this matter, this is considered an **unresolved item (50-346/96003-06)**.

F2 Status of Fire Protection Facilities and Equipment

During a walkdown in the reactor building, engineering personnel identified that a metal lip had not been installed on reactor coolant pump (RCP) 2-1 motor housing to collect and direct any oil leakage from the motor to the oil collection box. The motor had been replaced during the eighth refuel outage and the lip was not welded onto the replacement motor upon its installation. No substantive oil leakage was noted during the ensuing two operating cycles. However, 10 CFR Part 50, Appendix R required such an RCP oil collection system be in-place and functional. Inspector review of this matter was not complete at the end of the inspection period. Pending completion of that review, this matter is considered an **unresolved item (50-346/96003/07)**.

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Outage Radiological Controls

The inspectors reviewed radiological performance and the As Low As Reasonably Achievable (ALARA) program during the tenth refuel outage (10RFO). The inspectors observed that the licensee planned effectively for 10RFO. The selected pre-job briefings attended by the inspectors were thorough and included discussions on worksite radiological conditions. ALARA initiatives, such as the use of water shielding for selected work activities and the use of wireless remote dosimetry and closed circuit televisions for job coverage, effectively minimized personal dose. The total dose expended during 10RFO was approximately 154 manrem (1.54 Sv). Although this exceeded the outage goal of 120 manrem (1.20 Sv), emergent work on high dose jobs accounted for a majority of the dose expended above the goal. The final dose for these jobs was reasonable considering the increased work scope. The inspectors concluded that radiological performance improved since the previous refueling outage and that dose was effectively minimized.

R1.2 Shutdown Chemistry (84750)

The inspectors reviewed the licensee's shutdown chemistry process to determine effectiveness. The licensee used a higher concentration of hydrogen peroxide than in previous refueling outages to induce a CRUD burst during shutdown. To enhance cleanup, in addition to the normal letdown flow, a spent fuel pool system demineralizer was also used. Implementation of these changes was effectively accomplished through cooperation between operations, chemistry, and engineering departments. These changes resulted in a faster rate of removal of cobalt-58 which in turn resulted in lower radiation levels at the reactor coolant pumps and inside the steam generators.

R1.3 Radioactive Material Control (83750)

The inspectors reviewed the licensee's radioactive material (RAM) control program to assess staff awareness of RAM controls. The RP department's sensitivity to RAM controls was evidenced through the number of Potential Conditions Adverse to Quality Reports (PCAQRs) initiated in the last 6 months, suggesting program improvements and questioning attitudes. The overall program was effective during this assessment period, as indicated by the low number of RAM found outside of specified areas during the refueling outage.

R2 Status of RP&C Facilities and Equipment

The inspectors performed routine tours of the Auxiliary Building, Turbine Building and containment. During a system walkdown of liquid and gaseous radioactive waste (radwaste) systems, the inspectors found no problems with material condition. Minor housekeeping problems noted by the inspectors were addressed in a timely manner by the RP department.

R5 Staff Training and Qualifications in Radiological Controls

The inspectors reviewed selected procedures and personnel qualification statements for Contract Radiation Protection Technicians (CRPTs) hired to supplement the radiation protection departmental staff during the refueling outage. The inspectors noted that implementing procedures used for categorizing qualifying experience of CRPTs were consistent with the recommendation found in American Nuclear Standards Institute (ANSI) 18.1 (1971). Overall, the methods used by the licensee to verify CRPT qualifications and experience were being effectively implemented.

R7 Quality Assurance in RP&C Controls

The inspectors reviewed several Potential Conditions Adverse to Quality Reports (PCAQRs) generated by RP during the last 6 months, as well as those generated by other departments and assigned to RP. Overall, the PCAQRs initiated were consistent with the thresholds described in administrative procedure NG-NA-00702, Revision 01, "Potential Condition Adverse to Quality Reporting." Corrective actions to PCAQRs were inclusive and thorough.

R8 Miscellaneous Radiological Controls and Chemistry Issues

R8.1 (Closed) Violation (50-346/94010-01): Failure to perform adequate evaluations of radiological conditions associated with two 1994 refueling outage jobs. The corrective actions for these events were documented in inspection report (50-346/95005). The implementation of the corrective actions resulted in improved radiation work permits and ALARA briefing packages for the tenth refuel outage. Worksite radiological conditions were properly evaluated. No recurring problems were noted.

R8.2 (Closed) Violation (50-346/94010-02): Failure to follow written procedures governing radiological controls. The corrective actions for programmatic improvements were documented in inspection report (50-346/95-005). Effectiveness in the implementation of the programmatic changes was determined through reviews of licensee PCAQRs and direct observations during plant tours. The inspectors noted that radiation workers were exercising good radiological control practices. Heightened awareness of the general radiation worker force was noted regarding the appropriate response to electronic dosimeter alarms.

S1 Conduct of Security and Safeguards Activities

Personnel adherence to security program requirements continued to be satisfactory. Overall, plant personnel used security badges and keycards in accordance with plant procedures. Security officer performance also appeared satisfactory. Selected compensatory measures required for certain areas and/or activities were verified. Security officers in the Personnel Processing Facility (PPF) conducted themselves in a professional manner. Officers operated equipment in the PPF appropriately. Officers also appropriately controlled the issuance of security badges. No substantive concerns were noted in this area.

S2 Status of Security Facilities and Equipment

Personnel Processing Facility (PPF) search and access control equipment was verified to be operable by the inspectors. Selected security barriers, alarm stations, and protected area lighting were verified operable as well. The inspectors noted no substantive concerns in this area.

S8 Miscellaneous Security and Safeguards Issues

S8.1 (Closed) Unresolved Item (Report No. 50-346/95010)): Application of 5 working day time limit for reporting results to the MRO, as identified in Section 2.7(g)(1), of Appendix A to 10 CFR Part 26, to split specimens. Evaluation concluded that Section 2.7(g)(1) of Appendix A as it pertained to time limits for reporting test results did not apply to split specimens.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on June 8, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X2 Pre-Decisional Enforcement Conference Summary

On May 23, 1996, a pre-decisional enforcement conference was held at the NRC Region III office to discuss potential enforcement issues identified in Inspection Report (50-346/96002). Details of the conference are further discussed in inspection report (50-346/96004).

PARTIAL LIST OF PERSONS CONTACTED

Centerior Energy/Toledo Edison

J. Stetz, Vice President, Nuclear
J. Wood, Plant Manager
J. Lash, Director, Engineering & Services
T. Myers, Director, Nuclear Assurance
L. Dohrmann, Manager, Quality Services
D. Eshelman, Manager, Operations
J. Rogers, Manager, Maintenance
R. Zyduck, Manager, Nuclear Engineering
J. Michaelis, Manager, Nuclear Support
J. Freels, Manager, Regulatory Affairs
J. Moyers, Manager, Quality Assurance
E. Bergner, Manager, Training

Framatome Technologies

W. Boudreaux, Lead analyst

Factory Mutual

T. Lapps, ANII

The NRC inspectors also contacted and interviewed other licensee and contractor employees.

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillances
IP 62703:	Maintenance Observation
IP 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 73051:	Inservice Inspection - Review of Program
IP 73052:	Inservice Inspection - Review of Procedures
IP 73753:	Inservice Inspection - Review of Work Activities
IP 73755:	Inservice Inspection - Review of Data
IP 83729:	Occupational Exposure During Extended Outages
IP 83750:	Occupational Radiation Exposure
IP 84750:	Radioactive Waste Treatment, and Effluent and Environmental Monitoring
IP 92901:	Followup - Operations
IP 92902:	Followup - Engineering
IP 92903:	Followup - Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-346/96003-01	VIO	One example of a failure to follow procedure during emergency shutdown and restoration of EDG 2 to standby.
50-346/96003-02	VIO	One example of a failure to follow procedure during manipulation of a LPI valve.
50-346/96003-03	URI	Heavy load lifted over open reactor vessel.
50-346/96003-04	IFI	Licensee followup of displaced fuel assembly spacer grid.
50-346/96003-05	URI	Adequacy of engineering evaluations to support storage of materials in containment during operation.
50-346/96003-06	URI	Acceptability of radiant heat energy shields in the containment annulus.
50-346/96003-07	URI	Oil collection lip found not installed on RCP motor housing.

Closed

50-346/94010-01	VIO	Failure to perform adequate evaluations of radiological conditions associated with the performance of two outage jobs during the 1994 refueling outage.

50-346/94010-02	VIO	Failure to follow written procedures, specifically, those governing radiological control issues.
50-346/94013-02	IFI	AFW steam supply check valve chattering.
50-346/94004-02	IFI	Engineering evaluation of temporary lead shielding.
50-346/93013-02	IFI	Loss of service water to turbine plant cooling water heat exchangers.

LIST OF ACRONYMS USED

10RFO	10th Refueling Outage
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
ANSI	American Nuclear Standards Institute
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CNRB	Company Nuclear Review Board
CR	Control Room
CRDM	Control Rod Drive Mechanism
CRPT	Contract Radiation Protection Technologies
DAAS	Data Acquisition and Analysis System
DBOTP	Davis-Besse Operational Transient Procedure
dpm	disintegrations per minute
DH	Decay Heat
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ENS	Emergency Notification System
EO	Equipment Operator
ET	Eddy Current Testing
EVS	Emergency Ventilation System
FME	Foreign Material Exclusion
FTI	Framatome Technologies, Inc.
IFI	Inspection Followup Item
IR	Inspection Report
ISI	Inservice Inspection
LAR	License Amendment Request
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
Msv	milli-Sievert
MWO	Maintenance Work Order
MRPC	Rotating Pancake Coil
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
OCR	"Outside" the Control Room
OTSG	Once Through Steam Generator
PCAQR	Potential Condition Adverse to Quality Report
PDR	Public Document Room
PPF	Personnel Processing Facility
QA	Quality Assurance
RAM	Radioactive Material
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refuel Outage
RO	Reactor Operator
RP	Radiation Protection

RP&C	Radiological Protection and Chemistry
RPA	Radiologically Protected Area
RVHLT	Reactor Vessel Head Lifting Tripod
SBODG	Station Blackout Diesel Generator
SFAS	Safety Features Actuation System
SFP	Spent Fuel Pool
SG	Steam Generator
SRO	Senior Reactor Operator
Sv	Sievert
TIC	Temperature Indicating Controller
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation