U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/84-58 (DRP)

Docket No. 50-341 License No. CPPR-87

Licensee: The Detroit Edison Company
2000 Second Avenue
Detroit, MI 48224

Facility Name: Enrico Fermi Nuclear Power Plant, Unit 2

Inspection At: Enrico Fermi 2 Site; Monroe, MI

Inspection Conducted: October 29-November 2 and November 5-9, 1984

Inspectors: C. H. Scheibelhut — 11/20/84 Date

V. J. Elsbergas — 11/20/84 Date

Approved By: R. C. Knop, Chief
Project Section 1C — 11/29/84 Date

Inspection Summary

Inspection on October 29-November 2 and November 5-9, 1984 [Report No. 50-341/84-58 (DRP)]

Areas Inspected: Routine safety inspection by regional personnel of licensee action on previously identified items and 10 CFR 50.55(e) items and evaluation of licensee action with regard to IE Bulletins and Circulars. This inspection involved a total of 128 inspector-hours onsite by two NRC regional inspectors, including 0 inspector-hours onsite during off-shifts.

Results: In three of the areas inspected, no items of noncompliance or deviations were identified. In the fourth area, one item of noncompliance was identified as explained in paragraph 2.e of this report.

Details

1. Persons Contacted

The Detroit Edison Company
J. M. DuBay, Director, Planning and Control
F. H. Sondgeroth, Engineer, Nuclear Engineering
L. P. Bregni, Engineer, Licensing
S. E. Martin, Engineer, Licensing
D. E. McKenzie, Engineer, Licensing

The inspectors also interviewed other licensee and contractor personnel during the course of the inspection.

2. Licensee Actions on Previously Identified Items

a. (Closed) Safety Evaluation Report (SER) Open Item (341/81-10-05): "Residual Heat Removal Pump Runout Test". The SER noted that the applicant used extrapolated pump performance curves to analyze the RHR system for the possibility of damage to the ECCS pumps due to runout. The analysis indicated that pump runout would not occur. Acceptance of the analysis is contingent on a successful test to demonstrate that the extrapolated data were conservative.

The Final Safety Analysis Report (FSAR) in Section 6.3.2.14.1 defines the maximum predicted accident flow rate at 14,800 gpm for less than ten minutes. In Inspection Report 50-341/84-19, the inspector's report documented the witnessing of the test of the RHR C pump at an indicated 14,800 gpm for ten minutes. The manufacturer's pump performance curves for this pump go to 14,000 gpm. Hence, the pump was operating in an extrapolated performance region.

An analysis of the data taken during the test indicated the following:

- The corrected flow rate was 14,900 gpm.
- The NPSH was above the minimum required.
- Measured vibration was lower than measured vibration at low flow rates.
- Audible signs of cavitation were not detected.
- Drive motor full load amperage is 255 amps per phase. During the test, 265 amps per phase were measured. The motor carried this overload for the test period without overheating.

The inspector reviewed the test data and concluded that an acceptable test was completed. It showed that the extrapolated data were accurate, and the pump was not in a runout condition. This item is closed.

b. (Closed) SER Open Item (341/81-10-15): "NSSS Vendor Review of Procedures". The SER requires that the NSSS vendor, General Electric Corporation, review the startup tests and Emergency Operating Procedures. The Office of Inspection and Enforcement is to verify completion of the reviews. The inspector determined that the required reviews will be made as documented in Inspection Report 50-341/84-28. The item could not be closed at that time because it was not known if the on-site GE organization was authorized to make the reviews or if they would have to be made by GE, San Jose. It has since been determined that the on-site GE organization has made the reviews. Therefore, this item is closed.

c. (Closed) SER Open Item (341/81-17-04): "Completion of Modification to Equipment and Systems Resulting from the Seismic Reassessment". A seismic reassessment of the structures, systems, and components required for safe shutdown based on currently accepted NRC design response spectra was required. The Office of Inspection and Enforcement is to verify installation of equipment modifications (Supplement 1 to the SER).

DECO letter EF2-57,885 dated May 18, 1982, summarized the analytical results of the seismic reassessment. The results showed that all piping stresses were within code allowable values. However, a total of 13 snubbers had calculated loads that exceeded their rated loads and were required to be replaced with snubbers adequate for the new loads. This was the total modification activity required on the piping systems. DECO letter EF2-65,621 dated October 4, 1983, covered the status of the seismic reassessment of mechanical equipment. It was found that the shock isolation mountings of the emergency diesel generator engine instrument panels required modification.

The inspector chose four of the new snubbers at random for a detailed inspection. These were:

B31-E215-SSA1 covered by DCN 6984
B31-E215-SSA2 covered by DCN 6981
B31-G006-SSA8 covered by DCN 7039
B31-E215-SSA6 covered by DCN 6982

An inspection of the snubbers and their new mountings was made. Since the snubbers are considerably larger than the ones they replaced (55,000 pound rating vs. 15,000 pound rating) new mountings were required as well as new snubbers. In all instances, the snubbers, pipe attachment and anchors were as required by the DCNs.

Field Modification Request (FMR) 4287 was issued to modify the emergency diesel generator engine instrument panel shock isolation mounts to provide increased clearance. Maintenance Orders (PN-21s) were used to accomplish the work because adjustment of the isolators to the required gap was possible. No rework was required. An inspection of the isolators showed that they had the proper gaps.

A review of the PN-21s and hanger checklists (E.F.-136s) showed proper QC involvement in the activities. This item is closed.

d. (Closed) Item of Noncompliance (341/83-20-04A). The item of noncompliance was issued because certain requirements of ANSI Standard N45.2.2-1972 concerning warehousing of materials were not incorporated into the licensee's procedures. Specifically, for Part A of the item, Section 4.3.3 of ANSI N45.2.2 requires all austenitic stainless steel and nickel base alloy materials to be handled in such a manner that they are not in contact with lead, zinc, copper, mercury, or other low-melting elements, alloys, or halogenated material.

The licensee placed protective sleeves on the storage racks to prevent contact between stainless steel and the racks. In addition, the licensee revised Sections 5.3.5 and 6.3.14 of Plant Operating Manual (POM) 12.000.28, "Material Handling and Storage", to incorporate ANSI N45.2.2 wording concerning segregation of stainless steel and nickel base alloys from contact with the stated harmful materials.

The inspector visited the warehouse and verified the presence of the protective sleeves on the racks. Review of Revision 7 of POM 12.000.28 dated November 9, 1984, showed that the required wording was incorporated. Inspection Report 50-341/84-19 had closed Part B of the item of noncompliance. Therefore, this item is closed.

e. (Closed) Unresolved Item (341/83-20-09). During the course of an inspection regarding allegations concerning warehouse practices, the inspectors found that two flanges released under Operational Conditional Release (CR) 39-82 may have been installed in violation of procedures. Further inspection has shown that this was the case. Therefore, this unresolved item is escalated into an item of noncompliance as delineated below. This item is closed.

Section 5.5.6 of Administrative Procedure, "General", 12.000.27 states in part, "Materials placed on hold for documentation ... may be released for installation provided the item can be readily removed if necessary ... . Pipe spools released under these conditions shall not be welded nor tack welded, only fitted". In 10 CFR 50, Appendix B, Criterion V, states in part, "Activities affecting quality shall be prescribed by documented instruments, procedures, or drawings ... and shall be accomplished in accordance with these instructions, procedures, or drawings".

The licensee documented the noncompliance in Quality Surveillance Summary 5651, dated August 2, 1984. It was found that the two flanges were issued with a conditional release because certain dimensional and liquid penetrant tests that were required had not been made. The flanges were welded into the system in violation of Procedure 12.000.27. This is considered to be an item of noncompliance with 10 CFR 50, Appendix B, Criterion V (341/84-58-01).

NCR 83-796 was issued on August 8, 1984 to rectify the lack of the required NDE and dimensional tests. The dimensional checks and liquid penetrant tests were made and the flanges found to be acceptable. Therefore, the required corrective action has been accomplished.

To determine if similar violations had occurred, the licensee reviewed the status of all open conditional releases. At the time of the review, 120 conditional releases were open. One other similar violation was found on CR 8403. The limits of work to be performed on the CR permitted installation only, no testing. The item in question was a small part inside a pump. The pump in question was subsequently tested before the conditional release was cleared. Documentation has arrived permitting the CR to be closed. Therefore, no corrective action is required for this similar violation.

To prevent recurrence, the licensee revised Procedure 12.000.27 to allow installation/testing of items under a CR provided the installation is not irreversible.

The inspector reviewed the Quality Surveillance Summary, the NCR, CR 8403, and the revised procedure and found that the steps taken properly resolve the problem. Therefore, the item of noncompliance is considered to be closed with no further action required on the part of the licensee.

f. (Closed) Unresolved Item (341/83-30-06): "Diesel-Generator Control Panel Seismic Qualifications". The NRC inspectors noted that the licensee modified the four emergency diesel-generator (EDG) control panels in 1983 by adding a surge suppressor (RC filter) to each cabinet. The inspectors asked the licensee if the cabinets had been analyzed for the effects of the added equipment on the seismic qualifications.

The licensee's response to the NRC concern is provided in an internal memo EF2-67439 dated July 11, 1984. As stated in this memo, the Field Modification Request (FMR) No. 6301 used to add the filters was reviewed by a Civil Field Engineer for any effect on seismic qualification. A Seismic Qualification Review form (SQR) No. 437, dated September 22, 1983, was generated to record the results of the review. The review determined that the addition of boxes with filters has a negligible effect on the seismic qualification of the panel to which it was to be added.

An item related to the subject concern was a finding by a licensee's Civil Engineer during an inspection that the box with filters was secured to the panel roof via tapped holes and screws, rather than with bolts and nuts as required. This finding is documented in SQR No. 557, Rev. A. Following this discovery, Operating or Maintenance Orders (PN 21s) were written to replace the screws with bolts and nuts. The work was completed as documented on related PN 21, Attachment A.

It has to be noted also that since the Selenium Surge Suppressors (CR 8 rectifiers) that protect the field from faults on the generator output bus are scheduled to be replaced with larger units in the near future, per FMR S-7533, the inspector requested the licensee to review if this change would have any effect on the seismic qualification of the panel. The results of the licensee's review are documented in memo FE2-84-0152, Rev. A, dated November 1, 1984. The review concludes that the replacement of rectifiers per FMR S-7533 is acceptable from the seismic qualification viewpoint.

To conclude, the concerns raised by the NRC are considered to be resolved. This item is closed.

One item of noncompliance was identified as explained in Paragraph 2.e above.

3. Licensee Actions on 10 CFR 50.55(e) Items

a. (Closed) 50.55(e) Item 76 (341/82-27-EE): "Limitorque Limit Switch Rotor Failures". This item involves cracking and breaking of plastic rotors in Limitorque limit switch assemblies. The item was discussed in Inspection Report 50-341/84-41 (DRP). The discussion was based on the information contained in the licensee's final report to the NRC, EF2-69662, dated August 27, 1984. The item was kept open pending completion of replacement of faulty parts.

An amended final report to the NRC, EF2-69719 was issued by the licensee on November 3, 1984. In the amended report the number of valves found with cracked limit switch rotors was corrected to 17, instead of 16 originally reported. Also, although originally the licensee planned to replace complete limit switch assemblies, subsequently procedures were developed to replace only defective rotors. This is acceptable.

As shown in Noncompliance Reports (NCR) 82-139, -140, -148, -149, -199, -324, -357, and 83-280 and -283, the replacement of defective rotors or complete limit switch assemblies has been completed. This item is closed.

b. (Closed) 10 CFR 50.55(e) Item 102 (341/83-16-EE): "Lifting of Control Rod Guides During RHR Preop Test". After preoperational test E1100.001 ("Three RHR Pump Run") while inserting the control rods, some operated sluggishly and others would not fully insert.

The licensee wrote Nonconformance Report 83-955 to document the problem, determine the cause, and provide a disposition. It was determined that at the high flow rates through the reactor core volume during the test, several control rod blade guides lifted and became misaligned. Subsequent attempts to insert control rods at three locations caused damage to two control rod blades and to two orifice fuel supports. The damaged parts were sent to the vendor for repair and spare parts put in the reactor in their place. The control rod blade guides are temporary devices that take the place of fuel bundles before fuel is loaded into the reactor. A detailed

inspection of control rod blades, orifice fuel support castings, blade guides, etc., was accomplished. An NRC inspector witnessed parts of this inspection program. The results of the inspection were documented in GE letter STO-EF2-592, dated October 6, 1983.

The inspector reviewed the NCR and its attached Operating or Maintenance Orders (PN 21s) and the inspection results. We conclude that the problem will not recur with fuel in the core, that the damaged parts have been replaced or repaired, and that all of the damage has been found. Future preop testing has been modified to prevent recurrence when control rod blade guides are installed. This item is closed.

c. (Open) 50.55(e) Item 105 (341/83-19-EE): "Thermal Separation Criteria Violations". The licensee determined that separation required per Edison Specification 3071-33 was not maintained in some cases. Initially, separation violations were identified in the main steam tunnel. Subsequently, violations of the separation criteria were also found in the drywell and other parts of the plant. As discussed in Inspection Report 50-341/84-41 (DRP) the licensee had initiated necessary corrective action. The item remained open pending completion of work.

The review of the documentation presently in the licensee's files shows that, as documented in Deviation Disposition Request (DDR) Nos. E-13079, E-13080, and E-13083, rework necessary to provide the required thermal separation inside the drywell has been completed. In the steam tunnel, the existing arrangement was found to be acceptable based on evaluation of the loads in the power cables. This is documented in Nonconformance Report 84-1516 (issued to update DDR E-12051).

Two concerns remain unresolved. The first concern pertains to thermal separation in other areas of the plant (in addition to the drywell and main steam tunnel). As discussed with the licensee's cognizant personnel, violations of the thermal separation criteria were also apparently found and were corrected. However, no supporting documentation was available to confirm this fact.

The other concern relates to temperature ratings of the cables in the drywell. Although the separation criteria are based on an ambient temperature of 158°F, the maximum ambient temperature for certain cables is recommended not to exceed 140°F by the manufacturer. The licensee recognizes this limit and has established in an internal memo EF2-64,225, dated December 7, 1983, a need for a periodic surveillance program for the cables in the drywell. Temperature monitoring in the drywell is also considered necessary. Furthermore, as stated in memo EF2-64,225, experimental and analytical work to provide backup for the drywell cables has been initiated. At the time of the inspection, however, no results of this work were available for review. Also, no information on the surveillance program or the temperature monitoring system in the drywell was available.

The licensee's cognizant personnel were informed of the NRC concerns discussed above. The item remains open pending satisfactory review of the requested information by the inspector.

d. (Closed) 50.55(e) Item 119 (341/84-11-EE): "Design Deficiency in Conduit Support Standard ED-14-3". Seismic torsional stresses were not considered for conduit supports in Edison Specification 3071-128-ED.

As discussed in Inspection Report 50-341/84-41 (DRP), the licensee reviewed the allowable loads for the supports and issued necessary design changes. The findings are documented in Nonconformance Report (NCR) No. 84-1609. A total of 21 conduit supports required rework. As shown in NCR 84-1609, the work has been completed. This item is considered to be closed.

e. (Closed) 50.55(e) Item 124 (341/84-16-EE): "Lamination in ASTM-A516 Grade 70 Steel Plate". During the installation of electrical cable tray support members, laminations were noted in an A-516 Grade 70 steel plate utilized as a diagonal bracing member. Another piece of the same material exhibited laminations during bending in the fabrication shop. The defective steel was purchased from Energy Steel and Supply Company, Rochester, Michigan and was traced to heat number 59537. DECO Engineering Research Department (ERD) tests concluded that the laminations were caused by alumina inclusions in the material.

The found deficiencies were documented by the licensee in Nonconformance Report (NCR) No. 84-0275. For hangers, where the A-516 Gr. 70 plate material was utilized as a tension brace, connection to the vertical member was reinforced. Where the material was used for reinforcing instead as a tension brace, the part was replaced. All steel plate material from heat number 59537 which has not been installed remains on hold and will be returned to the vendor. This item is considered to be closed.

f. (Closed) 50.55(e) Item 1[illegible] (341/84-27-EE): "Design Deficiency of Conduit Support Weld Details". Some welds were deleted from eight conduit support standards in Edison Specification 3071-128-ED (Electrical Engineering Standards - Conduit Supports) by DCR-E-3006 without seismic qualification review. DCR-E-3006 specified two-sided welds where originally all-around (four sides) welds were required.

To resolve the found deficiency, the licensee requested the Giffels Associates, Inc. (GAI) to review the structural adequacy of the subject welds and provide necessary design changes to the standards. As a result of the review, DCN-10592, Rev. A, was issued. A review of the existing conduit supports was carried out by the licensee's Field Engineering Conduit Design Group. A total of 130 supports were identified which were built per DCR-E-3006. GAI performed individual design calculations for these supports and determined that only one required rework. DCR-E-5090 was issued to add welds to this support.

The found deficiencies and the required corrective action were documented by the licensee in Nonconformance Report (NCR) No. 84-1263. As shown on this NCR, all required work has been completed. This item is closed.

No items of noncompliance or deviations were identified.

4. Evaluation of Licensee Action with Regard to IE Bulletins

For the IE Bulletins listed below, the inspector verified that the Bulletin was received by licensee management and reviewed for its applicability to the facility. If the Bulletin was applicable the inspector verified that the written response was within the time period stated in the Bulletin, that the written response included the information required to be recorded, that the written response included adequate corrective action commitments based on information presented in the Bulletin and the licensee's response, that the licensee's management forwarded copies of the written response to the appropriate on-site management representatives, that information discussed in the licensee's written response was accurate, and that corrective action taken by the licensee was as described in the written response.

a. (Closed) IE Bulletin 79-08 (341/79-08-BB): "Events Relevant to Boiling Water Power Reactors Identified During Three-Mile Island Incident". The Bulletin requests the licensee to review the circumstances of the Three-Mile Island plant accident and take necessary corrective action.

The subject Bulletin was issued on April 14, 1979. Subsequently, the Bulletin requirements were incorporated into or attached to NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident". To review concerns of NUREG-0660, the licensee established a Safety Review Task Force (SRTF). A report was issued to present SRTF findings and recommendations. Specific responses of the licensee to certain concerns of the Bulletin (items 2, 3, 4, 6, and 8) relating primarily to the safety-related system design and operation were included in the FSAR and evaluated by NRR in the Safety Evaluation Report. The remaining items in the Bulletin are related primarily to procedures and are covered in Emergency Operations Procedure 29.000.02, "Cooldown", Administrative Operations Procedure 21.000.13, "Emergency Operating Procedure Guidelines", Emergency Procedure 29.000.05, "Contingency for Level Restoration", and Fermi 2 Interfacing Procedure 11.000.125, "General Regulatory Reporting Requirements".

Based on the review of the above documents, it is concluded that the concerns of the subject Bulletin are adequately addressed. This item is closed.

b. (Closed) IE Bulletin 80-25 (341/80-25-BB): "Operating Problems with Target Rock Safety-Relief Valves at BWRs". The Bulletin described a number of malfunctions of Target Rock safety-relief valves at operating BWRs. These included failure to open on manual demand, failure to reclose on manual demand. The failures were laid to

solenoid actuator failures, foreign material in the main stage of the valve, and excessive pressure in the pneumatic system supplying the valves. The Bulletin listed three actions items to be taken by licensees.

1. Inspect the solenoid actuators to verify that they are free from excessive Loc-Tite material.

2. Revise operating/maintenance procedures to include the requirements for any SRV that fails to function as designed, and the malfunction cause has not been clearly identified and corrected, that the entire valve is to be removed, inspected, and tested in accordance with the periodic surveillance rehabilitation requirements for the valves.

3. Relief valve protection on the pneumatic supply is to be provided in close proximity to the SRVs. High- and low-pressure annunciation for the SRV pneumatic supply is to be provided to the control room with the sensor(s) located as close to the SRVs as practicable. Operating procedures shall include operator guidance in responding to a high- or low-pressure alarm. Consideration should be given to replacing the solenoid operators with a design that can withstand greater pneumatic supply overpressure.

Fermi 2 uses the same type/design safety-relief valves as described in the Bulletin.

In response to the first Bulletin item, the licensee sent all of the solenoid actuators back to the vendor for rework. As part of the rework procedure, an inspection for excess Loc-Tite was made. In addition, startup test phase procedures STUT.HUD.26 and STUT.020.026 have provisions for verifying that relief valves are not sticking and reset properly after operation. Also, Maintenance Procedure 35.000.14, "SRV Solenoid Valve Disassembly, Reassembly, and Inspection", requires an inspection for excess Loc-Tite.

In response to the second Bulletin item, Plant Abnormal Operation Procedure 20.00.25, "Failed Open Safety Relief Valve", Section B2.2 states: "If the cause of S/R valve failure cannot be determined, the entire valve must be removed from service, disassembled, and inspected". Procedure 35.000.95, "Safety Relief Valve Removal and Installation", requires that contracts written for work on S/R valves ensure that the contractor comply with IE Circular 79-19, IE Bulletin 80-25 and General Electric Service Information Letter 196. Plant Maintenance Procedure-Surveillance, 34.000.10, "Main Steam System ASME Section XI - Relief Valve Set Point Test", requires that the actual setpoint test will be performed by a designated test facility in accordance with their procedures. Fermi 2 periodic surveillance requirements state that 50% of the S/R valves are overhauled during each refueling outage.

In response to the third Bulletin item, the licensee reviewed the S/R valves' pneumatic supply system for overpressure protection and

found that the existing system for the normal and backup supply systems met the Bulletin requirements. The pneumatic supply system high- and low-pressure annunicator alarms are installed in the control room and the sensor locations meet the Bulletin requirements. The pneumatic supply system inboard and outboard containment isolating valves have control room annunciator alarms to indicate closure. Alarm response procedure 8D70 provides the control room operator with proper response to high or low S/RV pneumatic supply system pressure.

The inspector reviewed the above referenced procedures and the licensee overpressure protection review and found that they met the requirements of the Bulletin. This item is closed.

c. (Closed) IE Bulletin 84-03 (341/84-03-BB): "Refueling Cavity Water Seal". The Bulletin describes the failure of a refueling cavity water seal at a nuclear reactor with the refueling cavity flooded in preparation for refueling. The Bulletin required licensees to evaluate the potential for and consequences of a refueling cavity water seal failure and provide a summary report of the findings.

The licensee prepared the required report and sent it to the NRC Regional Administrator by means of letter EF2-70038, dated November 5, 1984.

The inspector reviewed the report and two drawings: 5M721-2169, "Refueling and Drywell Seal Bellows for Primary Containment" and 6M721-2048, "Diagram, Fuel Pool Cooling and Clean Up System". The report concluded that the Fermi 2 seal does not contain active components (i.e., inflatable seals) and is not susceptible to the type of failure described. After reviewing the drawings that show the seal details and the leakage detection systems, an emergency operating procedure that requires that fuel bundles in transit be placed in the reactor vessel or spent fuel pool if there is any indication of leakage, and the report, we conclude that the Fermi 2 seal is not susceptible to the type of catastrophic failure described in the Bulletin. This item is closed.

No items of noncompliance or deviations were identified.

5. Evaluation of Licensee Action with Regard to IE Circulars

For the IE Circular listed below, the inspector verified that the circular was received by the licensee management, that a review for applicability was performed, and that if the Circular was applicable to the facility, appropriate corrective actions were taken or scheduled to be taken.

(Closed) IE Circular 80-05 (341/80-05-CC): "Emergency Diesel-Generator Lubricating Oil Addition and On-site Supply". The item was discussed in Inspection Report 50-341/84-47 (DRP). At that time, the available information was judged as insufficient to resolve two concerns of the circular.

In response to the NRC request, the licensee has provided additional information. Discussion pertaining to the first of the remaining concerns, that there is a sufficient lube oil supply for the emergency diesel generators (EDGs), is provided in the licensee's internal memo EF2-72297, dated October 29, 1984. The discussion addresses the EDG sump and lube oil tank capacities, lube oil consumption, alarms, provisions for adding oil, and oil supplies on-site. The inspector's review of the above memo concludes that it can be reasonably assured that an adequate supply of lube oil is always available for proper operation of the EDGs, as required by the circular.

The second concern was a need for a review to determine if there is other safety-related equipment that consumes lube oil and may require addition of oil while performing a safety function. This review now has been completed by the licensee and the results are discussed in internal memo EF2-72292, dated October 31, 1984. The review concludes that even though there is other equipment that uses lube oil, only limited replenishment or change-out at periodic intervals is required and the equipment truly does not consume oil.

Based on the NRC review of the above documents and the previously available information, the concerns of the subject circular are considered to be resolved. This item is closed.

No items of noncompliance or deviations were identified.

6. Exit Interview

The inspectors met with the resident inspectors and licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on November 9, 1984. The resident inspectors summarized the scope and findings of the inspection. The licensee acknowledged the inspectors' findings.