U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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License No:	NPF-3
Report No:	50-346/97006
Licensee:	Toledo Edison Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5503 N. State Route 2 Oak Harbor, OH 43449
Dates:	April 14 - May 27, 1997
Inspectors:	S. Stasek, Senior Resident Inspector K. Zellers, Resident Inspector D. Jones, Reactor Inspector
Approved by:	Christopher G. Miller, Acting Chief Reactor Projects Branch 4

EXECUTIVE SUMMARY

Davis-Besse Nuclear Power Station
NRC Inspection Report 50-346/97006

This inspection included aspects of plant operations, maintenance, engineering, and plant support. The report covers a six-week period of resident inspection; in addition, it includes the results of an announced inspection by a regional specialist.

Operations

- Control room operators maintained suitable operational control of the unit throughout the inspection period. Unit cooldown and heatup evolutions were performed in a deliberate, controlled manner and in accordance with plant procedures (Section O1.1).

- Engineered safety features and important-to-safety standby systems were walked down and verified appropriately lined up and maintained in excellent material condition (Section O2.1).

- Equipment and systems responded to a May 4 plant trip as designed. Operator actions in response to the transient were timely and conducted in accordance with station procedures (Section O2.2).

- Control room personnel did not recognize that a post accident monitoring system hot leg temperature indicator was inoperable until identified by the NRC. The indicator appeared to have been inoperable for at least a week (Section O2.3).

- A pre-startup checklist verification step was signed as complete without engineering supervision recognizing that Technical Specification (TS) requirements mentioned in the procedure had been deleted from TS (Section O3.1).

Maintenance

- Surveillance testing and maintenance activities reviewed during the inspection period, were conducted in accordance with plant procedures. Clearances prepared to support maintenance work adequately protected both equipment and personnel (Section M1.1)

- During inspector review of emergency core cooling system (ECCS) leakage issues, the maximum allowed total leakrate from each ECCS train was determined to be 20 gallons per hour. This leakrate appeared to not have been verified through periodic testing. As such, plant engineering and the NRC were evaluating the need to perform additional testing (Section M6.1).

Engineering

- Requirements governing the preparation and implementation of action plans were not well delineated. The inspectors were concerned that action plans could potentially be used in lieu of approved plant procedures. Review of this matter was continuing at the end of the inspection period (Section E3.1).

- Good performance by the potential condition adverse to quality reporting (PCAQR) review board was exhibited during the inspection period (Section E7.1).

Plant Support

- The NRC identified outdated locally posted radiological survey maps in several areas of the radiologically restricted area (RRA) on one occasion. Radiation protection (RP) personnel indicated the local survey maps had not been updated to reflect plant shutdown conditions since the plant trip approximately ten days earlier due to the associated increased work load on the RP technicians with the plant shut down. Plant personnel performing activities in the RRA had also failed to recognize the local survey maps were not current (Section R1.2).

- An April 30 emergency preparedness drill was satisfactorily conducted (Section P1.1).

- The Army Corp of Engineers predicted potential local area flooding during the upcoming summer and fall. As a result, the plant initiated a review of the site emergency plan evacuation routes during the inspection period to determine plan adequacy (Section P3.1).

Report Details

Summary of Plant Status

The plant began the inspection period operating at about 100 percent power. On May 4, the main transformer deluge system inadvertently actuated and caused the transformer to trip, initiating a reactor trip. Subsequent troubleshooting of the main transformer identified a need to replace it. The unit remained shut down until May 26 when the reactor was again made critical. At the end of this inspection period, the reactor was in the final stages of plant startup, with the reactor in Mode 1, operating at low power levels, and plant personnel completing post maintenance testing activities on the main generator and main transformer.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Control room operators maintained suitable control of the unit throughout the inspection period. Unit cooldown and heatup evolutions were performed in a deliberate, controlled manner and in accordance with plant procedures. Reactor operators were knowledgeable of the bases for alarming control room annunciators and were cognizant of in-progress plant evolutions.

Equipment operators adequately monitored the status of operating equipment. Lubricating oil levels, motor and pump temperatures, fluid leaks, as well as general plant conditions were checked shiftly as required. Discrepant conditions were communicated to operations supervision in a timely manner.

Adherence to applicable programmatic and administrative controls was observed. The status of inoperable equipment was effectively tracked and managed. Startup and operating mode restraint checklists were satisfactorily utilized with one exception (reference Section O3.1).

O2 Operational Status of Facilities and Equipment

O2.1 System Walkdowns (71707)

The inspectors walked down the accessible portions of the following engineered safety features (ESF) and important-to-safety systems during the inspection period:

- emergency diesel generators #1 and #2
- hydrogen dilution system - train 1
- auxiliary feedwater - trains 1 and 2
- low pressure injection system - trains 1 and 2
- containment spray system - trains 1 and 2

No substantive concerns were identified as a result of the walkdowns. System lineups and major flow paths were verified to be consistent with the updated safety analysis report (USAR), plant drawings, and applicable procedures. Equipment material condition was excellent in all cases. Pump and motor fluid levels were within their normal bands. Very little oil and fluid leaks were noted. Auxiliary equipment necessary for system operability was properly aligned and functioning. Local and remote controllers were appropriately positioned and attendant instrumentation appeared to be functioning correctly.

O2.2 Plant Trip

a. Inspection Scope (93702, 71707)

The inspectors conducted a review of the plant trip that occurred on May 4 to assess plant equipment and personnel response. Alarm recorder and data acquisition and analysis system (DAAS) plots, and control room logs were reviewed. Additionally, the inspectors interviewed operations, maintenance and engineering personnel.

Following the trip, the plant independent safety engineering (ISE) group performed a transient assessment to ensure that plant equipment responded to the trip as designed. The inspectors subsequently reviewed the ISE post trip analysis.

b. Observations and Findings

The inspectors determined that at about 5:30 p.m., the main transformer deluge system had inadvertently actuated, resulting in a large amount of low quality water to be sprayed across the exterior surfaces of the main transformer. About five minutes later, an electrical fault caused the main transformer and main generator to trip, which in turn tripped the reactor.

The inspectors verified that, with minor exceptions, post trip equipment response, including control rod drive system, turbine protection system, turbine bypass valves, and main steam safety and atmospheric dump systems all operated as expected. All control rod groups inserted per design and within required response times. As expected, 16 of 18 main steam safety valves (MSSVs) initially lifted to control secondary side pressure. However, the lift setpoint of MSSV SP17B7 shifted and the valve subsequently began intermittently lifting at a lower set pressure. Operators decreased secondary side pressure to stop the valve from lifting. No adverse effect on cooldown rates resulted from the shift in lift setpoint.

The inspectors reviewed ISE's post trip analysis and determined that it accurately assessed the trip and post trip response. The ISE analysis appeared consistent with the NRC assessment of the event.

The plant determined that the most likely root cause for the deluge actuation involved an inadvertent trip of a temperature sensor providing input to the deluge system. The deluge initiation logic was arranged with 24 temperature sensors in

the system, with any single sensor trip or failure able to actuate a transformer deluge. At the end of the inspection period, all 24 temperature sensors providing input to the main transformer deluge system were replaced and engineering was in process of evaluating the benefits of modifying the deluge actuation system.

c. Conclusions

Overall, plant equipment and systems responded to the trip as designed. Operator response to the transient was timely and conducted in accordance with station procedures. Plant post trip analysis adequately assessed the transient and was consistent with the inspectors' conclusions. Appropriate corrective actions were implemented to replace the main transformer and deluge system temperature sensors.

O2.3 Post Accident Monitoring Instrumentation Found Inoperable

a. Inspection Scope (71707)

The inspectors conducted a routine walkdown of the control room on May 3, and reviewed instrumentation associated with the post accident monitoring system (PAMS).

b. Observations and Findings

The inspectors identified that PAMS instrument TI RC3B6, channel 1 hot leg temperature indication for reactor coolant loop 1 was indicating about nine degrees lower than the other three hot leg temperature indications. The operations shift reviewed surveillance procedure DB-SC-03180, "Remote Shutdown, Post Accident Monitoring Instrumentation Monthly Channel Check," and determined that the subject instrument was inoperable because the difference in temperature indications from TI RC3B5 exceeded the procedure acceptance criteria of eight degrees. The licensee entered TS 3.3.3.6, Limiting Condition for Operation, and generated Potential Condition Adverse to Quality Report (PCAQR) 97-0578. This LCO required that the inoperable channel be returned to operable status within 30 days or to place the unit in hot shutdown within the next 12 hours.

The inspectors thereafter conducted a review to determine how long the condition had existed. A computer point that obtained its signal from the same sensor, T754, was found to be reading about 50 degrees below the value of the control room indication. A review of the computer point value for the seven preceding days revealed that the degraded situation had probably existed for at least seven days. Procedure DB-SC-03180 had been successfully performed two weeks prior to the date of inoperability identification.

Although no logs or surveillance tests had been performed on this instrument during the two-week period of time, several control room personnel, including reactor

operators, senior reactor operators, and shift managers failed to identify this discrepancy during shift turnovers and during performance of their duties.

Subsequent troubleshooting of this instrument string during the outage resulted in the instrument string returning to an expected indication value. Both computer point T754 and the PAMS local indication agreed in value. The exact failure mechanism could not be determined, but because the instrument string returned to normal indication during a terminal strip wiring and connection check, the root cause was suspected to have been a poor wiring connection.

c. Conclusions

Control room personnel did not recognize a post accident monitoring system hot leg temperature indicator was inoperable until identified by the NRC. Since the system had not been inoperable for greater than the 30-day TS limit, and the surveillance requirements were satisfied, no violation was cited for this situation.

Because the exact root cause of the failure of the instrument string was not determined, followup inspection activities to track the performance of the instrument string is warranted, therefore this is an **inspection followup item (50-346/97006-01(DRP)).**

O2.4 Containment Walkdowns (71707)

The inspectors performed several walkdowns of equipment and structures within the reactor building (containment) during the forced outage. No substantive equipment and material condition problems were identified. Paint and coatings appeared properly applied to equipment and structures in containment with no evidence of peeling or missing spots.

In addition, a walkdown of containment with the reactor coolant system at normal operating temperature and pressure was performed on May 25. No significant equipment leakage was identified. As-left material condition and housekeeping were satisfactory.

O3 Operations Procedures and Documentation

O3.1 Pre-Startup Checklist

a. Inspection Scope (71707)

On May 26, during evaluation of plant readiness to enter Mode 2 during plant restart, the inspectors reviewed procedure DB-OP-06911, "Pre-Startup Checklist."

b. Observations and Findings

The inspectors noted that Step 7.14.1 of the procedure required that the plant engineering manager sign that surveillances conducted to meet a list of Technical

Specification surveillance requirements were current and that engineering did not have any issues that would adversely affect entry into Mode 2. Although Step 7.14.1 had been signed on May 25 as complete, the inspectors noted that two of the Technical Specification surveillance requirements, TS 4.6.1.6 and TS 4.6.5.3 had been deleted from the Technical Specifications via license amendment 205 issued February 22, 1996.

When this matter was brought to the attention of operations personnel, a temporary alteration (TA) was initiated to correct the procedure. When this was complete, engineering personnel re-completed Step 7.14.1 of the procedure prior to entry into Mode 2.

The inspectors were concerned that engineering supervision had signed Step 7.14.1 as complete without fully realizing that TS requirements had been deleted. In addition, the license amendment change process should have identified that DB-OP-06911 referenced the subject Technical Specifications that were to be deleted and should have ensured appropriate revisions were made.

Inspectors review of this matter was not complete at the end of the current inspection period. This matter is considered an **unresolved item (50-346/97006-02(DRP))** pending completion of inspector review.

O4 Operator Knowledge and Performance

O4.1 Operator Response to Auxiliary Boiler Trip

a. Inspection Scope (71707)

The inspectors observed operator response to an inadvertent trip of the plant auxiliary boiler on May 21.

b. Observations and Findings

The auxiliary boiler tripped, as indicated by a control room annunciator, while the plant was in hot shutdown (Mode 4) and in process of raising reactor temperature to enter hot standby (Mode 3). The auxiliary boiler was in service to provide steam to the main condenser air ejectors and main turbine gland seals to maintain main condenser vacuum.

Control room personnel were observed to perform immediate and supplemental actions as directed by alarm response procedures. Control room personnel notified the assigned system engineer of the trip. The engineer accompanied the outside assistant shift supervisor to investigate the cause. The plant manager and the outage director were involved in monitoring control room response and were kept apprised on the status of followup actions by the operations superintendent. Throughout the auxiliary boiler trip recovery, the primary reactor operator maintained an appropriate level of cognizance of decay heat removal means and did not appear distracted by recovery actions associated with the auxiliary boiler trip.

The operations superintendent encouraged the shift to consider planning for alternate primary to secondary heat transfer contingencies in the event that main condenser vacuum was lost. Additionally he advised the shift supervisor to maintain his oversight function after the shift supervisor had personally secured the mechanical vacuum pump at one point.

When it was evident that the auxiliary boiler would not be immediately available to maintain main condenser vacuum, operators transferred decay heat loads from the main condenser to the decay heat removal system with the anticipation of losing condenser vacuum.

The root cause of the auxiliary boiler trip could not be determined. However, the auxiliary boiler served no safety-related function nor provided any critical support function for normal power operation. The auxiliary boiler was subsequently started and placed back into service.

c. Conclusions

Control room operators effectively responded to the auxiliary boiler trip and ensured suitable decay heat removal capabilities were maintained. Overall, good management oversight of control room trip recovery activities was noted.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) Violation (50-346/95009-01(DRP)): Short Shift Turnovers. In response, operations management implemented more stringent supervisory reviews and observations of reactor operator turnover activities. The inspectors have since identified no further instances of operators performing short shift turnovers.

O8.2 (Closed) Violation (50-346/96003-01(DRP)): Emergency Diesel Generator Placed in Standby Without Operators Completing All Procedural Verification Steps. In response, procedure adherence requirements were reinforced to all operations personnel during requalification training. Subsequently, additional procedure adherence concerns were identified, both by the NRC and the plant. In response to those issues, plant management was in process of restructuring the administrative controls governing operations and surveillance procedure usage.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance and Surveillance Activities (61726, 62707)

The following maintenance and surveillance testing activities were observed or reviewed during the inspection period:

- DB-SC-03077 Emergency Diesel Generator 2 184-Day Test

- DB-SC-03115 — SFAS 18-month Interchannel Logic Test
- DB-SP-03152 — AFW Interlock testing
- DB-SP-03150 — AFP 1 Monthly Jog Test
- DB-SC-03077 — Emergency Diesel Generator 2 184-Day Test
- DB-SC-10000 — Main Transformer Backfeed Test
- MWO 1-96-0909-00 — Repair/Replace PIM402 (SFAS Channel 4 RCS pressure indication)
- MWO 3-96-4938-01 — Transformer X01 Double Testing
- MWO 1-97-0580-03 — Replacement of Main Steam Safety Valve SP17B7

Surveillance testing activities observed during the inspection period were conducted in accordance with approved station procedures. Operator adnerence to test procedures was good, test deficiencies were written when required, and procedures were reviewed before use. Equipment problems, even those minor in nature, were brought to system engineers' attention, with appropriate corrective actions taken. The inspectors verified that tested equipment could perform as described in the updated safety analysis report (USAR).

Maintenance activities reviewed during the inspection period were conducted in accordance with plant procedures and in a controlled manner. Clearances adequately protected equipment and personnel. Lifted wire logs, when required, were performed and satisfactorily tracked the status of lifted wires. Instrumentation was observed to perform adequately after initial repair activities had been performed. Equipment was verified to have been declared inoperable when needed and limiting conditions for operation action statements were observed.

M3 Maintenance Procedures and Documentation

M3.1 Inservice Inspection

a. Inspection Scope (73052, 73755)

The inspectors reviewed documents related to nondestructive examination (NDE) equipment, evaluations, and data associated with the examination of four high pressure injection (HPI) nozzles (50, 51, 58, and 59).

b. Observations and Findings

On April 21, 1997, Oconee Unit 2 experienced an event in which a through-wall crack developed at the safe end to pipe weld on the HPI makeup line. The thermal fatigue crack was attributed to a gap which developed between the safe end and the rolled area of the nozzle thermal sleeve which allowed alternate heating and cooling by hot reactor coolant which flowed through the gap and the cooler makeup flow.

At Davis-Besse, HPI connections with thermal sleeves (nozzle 58 provides normal makeup) were located on each of the four cold legs. The thermal sleeves in nozzles 58 and 59 were replaced in 1988 (reference Inspection Report 50-346/88009(DRS)) with redesigned sleeves.

The licensee decided that the conservative approach in response to the Oconee event would be to inspect all four HPI nozzles and the 16 welds connecting the dissimilar metal welds to the check valves rather than just the one nozzle and four welds subjected to makeup flows. Radiography testing (RT) was performed to identify any gap between the thermal sleeve and nozzle and for rotational movement of the thermal sleeve within the safe end. Weld buttons were located at the ends of the thermal sleeves to prevent movement of the sleeve. If the sleeve rotated, it would smear the weld button which would be visible in the radiograph. Ultrasonic testing (UT) was used to inspect the 16 welds between the check valves and the dissimilar metal welds of the nozzles for service induced flaws. The UT was performed using Electric Power Research Institute (EPRI) performance demonstrated ultrasonic equipment and personnel.

The examination data was found to be in accordance with the applicable nondestructive evaluation procedures and American Society of Mechanical Engineers (ASME) Code requirements. The RT of the four thermal sleeves showed no indication of rotational movement or gap between the thermal sleeve and safe end. The UT of 16 welds showed no indications of service induced flaws.

c. Conclusions

For the areas observed, ASME Code requirements were met and no violations or deviations were identified. The licensee's approach in the inspection implementation appeared to be proactive and conservative.

M6 Maintenance Organization and Administration

M6.1 Leakage Rate Testing of Emergency Core Cooling System

a. Inspection Scope (37551)

The inspectors reviewed a licensee identified issue where categorization of decay heat valves DH18 and DH19 and high pressure injection valves HP31 and HP32 in the plant inservice testing (IST) program required further evaluation.

b. Observations and Findings

The inspectors noted that engineering attempted to take timely and appropriate corrective actions to the self-identified issue. Engineering indicated that DH18 and DH19 would be classified as Category C tested valves and HP31 and HP32 as Category B tested valves under the plant IST program.

The inspectors noted that Category B valves, although requiring verification of full closure, did not require quantification of seat leakage. Engineering indicated that several other valves in the emergency core cooling system (ECCS) were similarly classified. The inspectors questioned whether an integrated ECCS design maximum leakage rate was specified that would encompass all individual component leakage. Engineering performed a review of initial licensing documents and identified that the maximum allowed ECCS leakage rate per train was 20 gallons per hour (GPH). The inspectors questioned whether the ECCS leakage rate was verified to ensure that it was less than the maximum allowed value.

The inspectors also noted that TS 6.8.4.a required implementation of a program to, in part, address potential primary coolant sources outside of containment, including leakage from the low pressure injection and high pressure injection systems. At the conclusion of the inspection period the licensee was evaluating the need to perform an integrated ECCS leakage rate test and whether the TS 6.8.4.a specified program was being properly implemented. Pending completion of licensee evaluation and subsequent inspector review, this matter is considered an **unresolved item (50-346/97006-03(DRP))**.

M8 Miscellaneous Maintenance Issues (92700, 92902)

M8.1 (Closed) LER 50-346/96-001-00: Inoperable Emergency Core Cooling System. This event involved the plant failure to perform a Technical Specification required surveillance to periodically vent all of the high pressure injection system high points. In response, the licensee requested and received a license amendment which allowed the facility to be in compliance with the surveillance requirement of the Technical Specification until the next refuel outage. During the tenth refuel outage completed in June 1996, a plant modification to install a manual vent valve on the high point of the associated HPI piping was completed. The NRC had previously evaluated this issue as documented in Inspection Report 50-346/96002. That review concluded with issuance of an escalated enforcement action.

M8.2 (Closed) LER 50-346/96-005-01: Inadequate Control of Heavy Loads in the Containment Building. This matter involved an event where an eight ton reactor vessel head lifting tripod was inadvertently traversed over the open reactor vessel in violation of plant procedures as well as NRC requirements associated with the control of heavy loads. The NRC had previously reviewed this matter which concluded with a violation being issued in Inspection Report 50-346/96005.

III. Engineering

E3 Engineering Procedures and Documentation

E3.1 Control of Action Plans

a. Inspection Scope (71707)

The inspectors reviewed operations night order log sheets, one of which included two action plans, during a control room walkdown on May 20.

b. Observations and Findings

Note 6 of the May 16, 1997 operations night orders indicated that the water between the containment isolation valves for the pressurizer auxiliary spray line, and the water between the containment isolation valves for the reactor coolant drain line, should be partially drained per PCAQR 96-1199. The report provided interim resolution of NRC Generic Letter (GL) 96-06 concerns involving thermally induced containment penetration overpressure concerns. The night orders included two action plans that were to be used for accomplishing the partial draining evolutions. These lines had been partially drained of water prior to the forced outage that had started on May 4, but the lines had been used to support plant cooldown and required re-draining prior to startup.

The two action plans described the specific steps for the operators to take in order to partially drain the lines. The steps described obtaining shift supervisor permission, notification of radiation protection personnel, obtaining water collection bottles, verification of equipment lineups, manipulating plant equipment, draining water from the lines, and restoring plant equipment. Additionally, there were signature blocks for the preparer, approver and performer of the action plan. The action plans appeared to direct actions similar to procedures without having undergone the same review and approval process as procedures.

The only available documentation that prescribed the requirements for the generation, control and execution of action plans consisted of a plant engineering policy, PE-05. This policy provided guidance for plant engineering personnel for the preparation of plant engineering action plans to address administrative or hardware related activities. However, the action plans for addressing the interim resolution of GL 96-06 concerns were generated, approved, and executed by operations personnel. No administrative controls to address operations use of action plans were in place.

At the end of the inspection period, the inspectors as well as plant personnel were reviewing policy PE-05 to determine if it was sufficient to control generation and use of action plans in general. Additionally, inspector review to determine if the steps performed by the two GL 96-06 action plans should have been prescribed by

plant procedures had not been completed. Pending completion of inspector review, this matter is considered an **unresolved item (50-346/97006-04(DRP))**.

E7 Quality Assurance in Engineering Activities

E7.1 PCAQR Review Board (37551)

The inspectors observed a PCAQR review board convened to assess a number of PCAQRs deemed ready for review. Good participant knowledge of corrective action program requirements was demonstrated during the meeting. The PCAQRs which had incomplete documentation and/or required followup were sent back to the evaluator for further action. Also, board members satisfactorily discussed the appropriateness of initial assessments, remedial actions, root causes and actions to prevent recurrence. A repetitive problem regarding the inadvertent blocking of fire doors was addressed by assigning a root cause and corrective action to prevent recurrence evaluation to be done.

E8 Miscellaneous Engineering Issues (92700, 92903)

E8.1 (Closed) Unresolved Item (50-346/96002-05(DRP)): Potential Inadequate Evaluation of Electrical Hot Shorts. The NRC had previously reviewed this issue as documented in Inspection Report No. 50-346/96008 with the resultant being issuance of escalated enforcement.

E8.2 (Closed) Unresolved Item (50-346/96003-07(DRP)): Reactor Coolant Pump Motor Oil Piping Not Configured as Required. The licensee subsequently submitted LER 50-346/97-004-00 that discussed the details of each of the individual issues. This matter will be reviewed as part of the inspectors' followup to the subject LER.

E8.3 (Closed) Inspection Followup Item (50-346/96010-06(DRP)): Effects of Mirror Insulation Debris. The inspectors reviewed a pre-existing engineering evaluation that addressed the potential effects of debris generated from a high energy line break in containment in areas near the emergency sump. The inspectors also performed a walkdown of the containment adjacent to the emergency sump and determined the amounts of mirror insulation utilized were consistent with the engineering evaluation. An engineering evaluation concluded that mirror insulation debris would not adversely affect proper functioning of the emergency sump screening.

E8.4 (Closed) Deviation (50-346/95004-02(DRS)): Failure to Trend Low Pressure Transmitter Performance. This issue involved a commitment the licensee had made to trend low pressure rosemount transmitter performance via weekly reviews of computer data points. During the inspection it was identified that 12 of the transmitters were not being trended via the computer. Engineering subsequently determined that the subject transmitters did not have inputs to the computer. Although the actions committed to in the licensee's response to NRC Bulletin 90-01, Supplement 1, were inaccurate for the subject 12 transmitters, performance was being trended using alternate means.

E8.5 (Closed) LER 50-346/96-007-00: Control Room Emergency Ventilation System Design Bases Calculation Error. This issue involved an identification by engineering that control room emergency ventilation system inleakage limits were based upon inaccurate assumptions and calculations. Once identified, the licensee took appropriate corrective actions to further limit the operational inleakage rates to less than the newly analyzed limits.

The NRC previously reviewed this matter as documented in Inspection Report No. 50-346/96014.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 General Comments (71750)

During the inspection period the inspectors conducted frequent walkdowns of the radiologically restricted area (RRA). Radiation, high radiation and contaminated area controls and postings were verified to be in conformance with radiation protection (RP) procedures. Walkdowns of the containment were conducted during the most recent forced outage to verify appropriate radiological postings, up-to-date and accurate surveys were conducted, and personnel adherence to RP procedural requirements were adequately implemented in containment.

R1.2 Outdated Locally Posted Radiological Surveys

a. Inspection Scope (71750)

On May 15, the inspectors conducted a tour of the auxiliary building and reviewed selected locally posted radiological survey maps.

b. Observations and Findings

The inspectors noted that several locally posted radiological survey maps were dated April 28, predating the May 4 plant trip. The areas included ECCS pump rooms #1 and #2, and the decay heat cooler pit. Although the plant was currently in cold shutdown with decay heat train #1 (in ECCS pump room #1) in service for shutdown cooling, the survey maps still reflected full power operating conditions. As such, the survey maps did not accurately reflect current radiological conditions in the subject areas. The local postings were updated very shortly after RP supervision was notified. The licensee stated in discussions that locally posted radiological survey maps were not required by the site RP program, and that the local postings were for information only. When questioned why the locally posted survey maps had not been updated following shutdown of the unit, RP personnel indicated that because of the higher than anticipated workload associated with the outage, several routine activities, including the posting of the local survey maps, had not been completed within their normal time frames.

c. Conclusion

The inspectors noted that local radiological information provided to plant workers did not accurately reflect conditions in some areas. In addition, personnel performing work activities in the RRA had also failed to recognize (or report) that the local survey maps were not current. At the end of the inspection period the inspectors had not completed the review of this matter. This is considered an **unresolved item (50-346/97006-05(DRP))** pending completion of inspector review.

R2 Status of RP&C Facilities and Equipment

R2.1 Calibration of RP Instrumentation (71750)

A sample of portable radiation survey instruments were inspected during the inspection period. The inspectors verified that the instruments were functional and properly calibrated within required time frames. Personnel contamination monitors (PCMs) located at the RRA exit point, as well as near the containment personnel hatch, were verified to be functional and appropriately calibrated. A sample of area radiation monitors (ARMs) were also evaluated and determined to be appropriately functional, with up-to-date calibrations and setpoints conservatively established.

P1 Conduct of EP Activities

P1.1 Observation of Emergency Preparedness Drill (71750)

On April 30, the inspectors observed portions of an onsite emergency preparedness drill from the simulator control room (SCR), technical support center (TSC), and emergency control center (ECC). The exercise scenario was aggressive, involved numerous equipment failures and was designed to eventually cause declaration of a general emergency. Emergency classification upgrades were performed in accordance with the emergency plan and were made in a timely manner. Good communications between the TSC, Operations Support Center (OSC) and SCR were observed. Control room shift briefs adequately communicated the status of the plant to control room personnel.

P3 EP Procedures and Documentation

P3.1 Potential Flooding Conditions

a. Inspection Scope (71750)

The inspectors evaluated information that the licensee had received from the Army Corps of Engineers concerning potential local area flooding predicted to occur during the upcoming summer and fall.

b. Observations and Findings

The inspectors noted that the projected Lake Erie water level conditions could adversely affect the approved emergency plan evacuation routes during certain extreme storm conditions. Based upon the information received from the Army Corps of Engineers, the emergency planning (EP) organization initiated a review of local area topography to evaluate whether revisions were necessary to the emergency plan evacuation routes during the times of potential flooding. In addition, EP was planning to update their site isolation procedure if conditions warranted that alternate routes be taken for relief personnel to travel to the site. Pending completion of the licensee review of this matter and determination as to whether evacuation routes could adversely be affected, this matter is considered an **inspection followup item (50-346/97006-06(DRP)).**

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 May 27, 1997. 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.

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Licensee

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G. M. Wolf, Engineer, Licensing
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INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73052:	Review of Procedures
IP 73755:	Inservice Inspection Data Review and Evaluation
IP 92700:	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Plant Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

50-346/97006-01(DRP)	IFI	Inoperable post accident monitoring hot leg temperature indicator
50-346/97006-02(DRP)	URI	Pre-startup checklist not appropriately completed
50-346/97006-03(DRP)	URI	ECCS leakage rate testing not performed
50-346/97006-04(DRP)	URI	Action plans not adequately controlled
50-346/97006-05(DRP)	URI	Outdated local radiological survey maps
50-346/97006-06(DRP)	IFI	EP evacuation routes could be affected by local flooding

<u>Closed</u>

50-346/95009-01(DRP)	VIO	Short shift turnovers
50-346/96003-01(DRP)	VIO	Emergency diesel generator placed in standby without operators completing all procedural verification steps
50-346/96002-05(DRP)	URI	Potential MOV electrical hot short due to fire induced damage
50-346/96003-07(DRP)	URI	Reactor coolant pump Appendix R concerns
50-346/96010-06(DRP)	IFI	Mirror insulation debris following a postulated HELB in containment
50-346/95004-02(DRS)	DEV	Failure to trend low pressure transmitter performance
50-346/96-001-00	LER	Inoperable Emergency Core Cooling System
50-346/96-005-01	LER	Inadequate Control of Heavy Loads in Containment Building
50-346/96-007-00	LER	Control Room Emergency Ventilation System Design Bases Calculation Error

LIST OF ACRONYMS USED

AFP	Auxiliary Feedpump
AFW	Auxiliary Feedwater
ARTS	Anticipatory Reactor Trip System
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
DAAS	Data Acquisition and Analysis System
DEV	Deviation
ECC	Emergency Control Center
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
GPH	Gallons Per Hour
GL	Generic Letter
HELB	High Energy Line Break
HPI	High Pressure Injection
ISE	Independent Safety Engineering
LER	Licensee Event Report
MOV	Motor Operated Valve
MSSV	Main Steam Safety Valve
MWO	Maintenance Work Order
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OSC	Operations Support Center
PAMS	Post Accident Monitoring System
PCAQR	Potential Condition Adverse to Quality Report
PCM	Personnel Contamination Monitor
RCS	Reactor Coolant System
RP	Radiation Protection
RRA	Radiologically Restricted Area
RT	Radiography
SCR	Simulator Control Room
SFAS	Safety Features Actuation System
SRB	Station Review Board
TA	Temporary Alteration
TI	Temperature Instrument
TS	Technical Specifications
TSC	Technical Support Center
USAR	Updated Safety Analysis Report
UT	Ultrasonic Testing
VIO	Violation